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Advances in fast reactor technology

Proceedings of the 30th meeting of the International Working Group on Fast Reactors held in Beijing, China, 13–16 May 1997

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FOREWORD

Individual States were largely responsible for early developments in experimental and prototype liquid metal fast reactors (LMFRs). However, for development of advanced LMFRs, international co-operation plays an important role. The IAEA seeks to promote such co-operation. For R&D incorporating innovative features, international co-operation allows pooling of resources and expertise in areas of common interest. Information on experience gained from R&D, and from the operation and construction of fast reactors, has been reviewed periodically by the International Working Group on Fast Reactors (IWGFR). These proceedings contain updated and new information on the status of LMFR development, as reported at the 30th meeting of the IWGFR, held in Beijing, China, from 13 to 16 May 1997.

EDITORIAL NOTE

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SUMMARY

1. INTRODUCTION

The 30th Annual Meeting of the IAEA International Working Group on Fast Reactors (IWGFRs) was held in Beijing, China, from 13 to 16 May 1997. The meeting was attended by IWGFR members from Brazil, China, France, Germany, India, Japan, the Republic of Kazakhstan, the Republic of Korea, the Russian Federation, the United Kingdom, as well as by an observer from Switzerland.

The primary purpose of the meeting was to co-ordinate an exchange of information on the status of fast reactor development and operational experience, including experience with experimental types of reactor.

2. IWGFR ACTIVITIES

Present trends in LMFR design are aimed at providing design which can compete economically with other reactor types. Because of the relatively high capital cost of LMFRs, this implies the need for both high availability and plant lifetimes in excess of 40 years. Reduction of over-conservative design, while avoiding failure over lifetimes of at least 300 000 hours is therefore a motivation for the development of improved design standards. To this end, two TCMs (on Creep Fatigue Damage Rules To Be Used in Fast Reactor Design (June 1996); and on Evaluation of Radioactive Materials Release and Sodium Fires in Fast Reactors (November 1996)) were held in the United Kingdom and Japan respectively, providing an important opportunity to review developments in design rules for creep-fatigue conditions, codes and method for evaluation of radioactivity release and discussion of safety design principles against sodium fires.

The Co-ordinated Research Programme on Intercomparison of Analysis Methods for Seismically Isolated Nuclear Structures started in 1996. The purpose of this CRP is to validate reliable numerical methods used for both detailed evaluation of dynamic behavior of seismic isolators and isolated nuclear structures of various NPP types. The first Research Coordination Meeting (RCM) was held in May 1996 in St. Petersburg, Russia to discuss in detail the experimental data on seismic isolators and perform a comparison of the results of calculations based on such data. The CRP on Harmonization and Improvement of FR Thermomechanical and Thermohydraulic Codes and Relationship using Experimental Data was started in 1996. The first RCM, which was held in November in Lyon, France, discussed the status and scope of contributions to the CRP from participating organizations, reviewed the case setup data provided by FRAMATOME and discussed first results of thermohydraulic calculations provided by the participants. A joint IAEA/EC benchmark exercise for hypothetical accident (unprotected loss of coolant flow) in a BN-800 reactor with near zero void core has been continued during 1996. Two consultancies were held: one in June 1996 in Brussels and another in December 1996 in Vienna, in order to review and concur the results of preboiling transient calculations provided by organizations from France, Germany, India, Italy, Japan, Russian Federation and United Kingdom.

In addition, a consultancy was held in October 1996 in Vienna to harmonize the international assistance to ensure stable operation during the remaining lifetime and the development of an effective decommissioning programme of the BN-350 fast reactor in Kazakhstan.

3. NATIONAL FAST REACTOR PROGRAMMES

Brazil. Fast reactor activities in Brazil are mostly research-oriented and have the objective of establishing a consistent knowledge base which can serve as a support for a future transition to the activities more directly related to design, construction and operation of an experimental fast reactor. Due to the present economic difficulties and uncertainties, the programme is modest, with a potential to grow; however, it is affected by the present "eye on the market" philosophy to permeat everything which concerns nuclear energy in Brazil. Under a cooperation agreement with the Instituto de Engenharia Nuclear (IEN, Nuclear Engineering Institute), neutronics calculations are being continued with different methodologies in order to choose the best route for more refined calculations if necessary. The objective here is to understand the influence of various nuclear data sets, etc. A natural flow circuit using water as the circulating fluid has been recently installed in the Department of Mechanical Engineering of the Instituto Tecnolólogico de Aeronáutica (ITA, Aeronautic Technological Institute), as part of a formal agreement between the Advanced Studies Institute and ITA. Using a 8 kW(th) heat source, the circuit will be used for studying basic phenomena and also for computer code (passive system analysis) validation. In the area of heat transfer/removal, an exchange between the researchers at the Indira Ghandi Centre for Atomic Research (IGCAR, India) has been initiated in order to identify topics of common interest for a future collaboration. As in the ALMR (USA), HT-9 is being considered as our reference for cladding material. In 1996, a sample was experimentally obtained and first analyses have indicated the need for minor adjustments in its composition, which are presently being done. Under a formal cooperation agreement with the Instituto de Pesquisas Energéticas de Nucleares (IPEN, Nuclear Research Institute, San Paulo), metallic uranium recovery by electrorefining is being investigated, in order to understand the basic processes involved. A first experiment performed in 1996, in which a small amount of uranium was electrodeposited, has been successful in the sense that it allowed the identification of problems which deserve further attention. Among those can be included the uncertainties in the elecrodeposition current to be applied, whose values in the literature are conflicting. A small electromagnetic pump (EMP) has been recently built and successfully tested at the Instituto de Estudos Avançados (IEAv, Advanced Studies Institute). The pump uses an iron C-type magnet with a direct-current coil to obtain the required magnetic induction. Present research involves extensive testing and also the utilization of rare-earth permanent magnets ($SmCo_5$) in place of the C-magnets in order to reduce the pump's geometric dimensions. R&D work on alternating current EMPs is planned to be initiated in 1997. After many years "on the shelf", the first of the three sodium loops which were part of the contract with ANSALDO/NIRA (Italy) will finally be assembled at the Institute de Engenharia Nuclear (IEN, Nuclear Engineering Institute, Rio de Janeiro), with completion expected by mid-1997. The loop (named SS-10) is to be used as a bed for experiments related to sodium purification and transferring.

China. During the ninth 5-year plan (1996-2000), four nuclear power stations consisting of eight reactors with 6620 MWe installed will be constructed. Within the framework of the High Technology Programme, the Chinese Experimental Fast Reactor (CEFR) with a capacity of 65 MWth/25MWe is under progress. The experimental reactor CEFR-25 may use UO_2 fuel for the first core; subsequent core will use MOX fuel. It is a sodium-cooled pool type reactor. There are two primary sodium pumps and four intermediate heat exchangers (IHX) in the main vessel, two secondary loops, single steam/water circuit and one turbine-generator set. The decay heat removal is provided by two independent passive residual heat removal systems.

France. The CAPRA project, initiated in February 1993, aimed at demonstrating the feasibility of a fast reactor to burn plutonium at as high as possible rate. The first two-year phase of the CAPRA project studies was completed. Complementary studies, now in the framework of the CAPRA programme, were performed in 1996-97. EDF maintains interest and support for FR development, as a contribution to the studies on the different possible options for the management of the back end of the fuel cycle for getting best value from SPX operating experience and to preserve expertise and previous investments. In that contract EFR programme studies aims notably to provide a "paper work house"(reference EFR) for assessment of the economic or safety interest of R&D results. The EFR programme studies have been recently adjusted to take account of the limitation of resources. In 1996 the EFR programme studies have allowed significant progress in the definition of the reference EFR for which full updated safety and economics documentation is now expected for 2001. The main outstanding issues are identified and the feedback from operating experience of FRs worldwide is obtained through systematic reviews of all EFR plant areas. Core safety and containment studies have confirmed the effectiveness of the design, notably partially closed "plan table area over the roof in case of contaminated sodium fire. Significant progress has been obtained also in the repair domain, in the definition of advanced specifications, codes and standards and computing tools. Innovative concept studies have assessed the interest in features with potential for adoption in the reference EFR, i.e. the design of strong back resting on the main vessel bottom which reduces inspection requirements of the core support structures.

Germany. Fast reactor activities at Forschungszentrum Karlsruhe (FZK) include neutron physics and safety calculations of selected core configurations for burner fast reactor (CAPRA type) and post-irradiated examination of structural and cladding materials. They are embedded in the European fast reactor development programme. First calculations of the initiation phase of an oxide core melt-down behavior were performed with the newest version of the SAS4A code. To determine swelling and in-pile creep of FR cladding material, a pressurized tube experiment was carried out in the PFR reactor. Samples of various alloys (model plain Fe-15Cr-15Ni, commercial German 1.4970, etc.) were tested in the temperature range between 420^{o}C and 600^{o}C and doses were reached between 61 and 106 dpa. After shut-down of the PFR, the samples were transported to FZK for PIE and non-destructive measurements of diameter and length were completed. In collaboration with CEA, a ferritic-martensitic wrapper material (W.Nr. 1.4914) was irradiated in the PHENIX reactor for 105 dpa and impacts on service life and integrity should be examined by testing the mechanical properties and microstructure. Particular attention was paid to the welds connecting the martensitic wrapper material with the austenitic (AISI 316) top and bottom sleeves.

India. Operation of fast breeder test reactor (FBTR) is continuing. The maximum fuel burnup reached is 32 000 MW·d/t without any failure. One fuel subassembly was taken out at 25 000 MW·d/t burnup and post-irradiation examination has been completed. The fuel performance is very good and the targeted fuel burnup is 50 000 MW·d/t. Turbine was rolled up to synchronous speed of 300 rpm several times during the year and the operation was found to be smooth. Synchronization with the grid could not be done because of the problems with oil leak in the bearing, failure of auxiliary oil pump motor and problems in getting the required vacuum. Regulatory authority has approved the proposal to raise the power from 10.5 to 12.5 MW(th) with the addition of fuel subassemblies in the core. Design and R&D works on 500 MW(e) Prototype Fast Breeder Reactor is continuing. The main options for the reactor are sodium coolant, pool type primary sodium circuit, MOX fuel, two primary sodium pumps, two secondary sodium loops with 4 SG in each loop. The important design activities carried out during the year are plant dynamic studies, decay heat removal analysis, design of pump to diagrid pipe, scram parameters, location of secondary sodium pump in the secondary circuit and design of the fuel handling machines. The important experimental R&D work carried out during the year are testing of prototype primary sodium pump in water, operation of a large sodium test rig to study heat and mass transfer in the cover gas, fabrication of six annular linear induction pumps, testing of dummy fuel subassembly on the water test rig for pressure drop and vibration measurements, testing of the full length SG sector model for flow distribution, pressure drop and vibration and testing of 1/3 scale diagrid model in air for pressure drop and flow distribution measurements. 30 kW(th) U-233 fueled KAMINI reactor was made critical in October 1996 and will be used for activation analysis and neutron radiography. Post-irradiation examination of experimental fuel pins and one full scale fuel subassembly has been completed and the results are very encouraging. The non-destructive examinations carried out include eddy current inspection of cladding tubes and ultrasonic examination of the secondary sodium pipe welds of FBTR. In the field of materials development, tensile, creep and creep-fatigue interaction effects on SS 316 LN and 9 Cr-1Mo, microstructural characterization and environmental effects on SS 316 LN and 9 Cr-1Mo, weldability of SS 316 LN and inconel 718, aluminizing of inconel 718, thermal ageing of trimetallic joints (SS 316 LN/Alloy 800/9 Cr-1Mo) and fracture properties of 9 Cr-1Mo welds have been studied.

Japan. During the period from April 1996 to March 1997, the 30th duty cycle operation has been started on the experimental fast reactor Joyo. The cause investigation on sodium leak incident has been completed and safety examination is being performed on the prototype fast reactor Monju.

The design study for demonstration FBR plant optimization has been completed by JAPC for three years since fiscal year 1994. Related research and development are underway, being discussed and coordinated by the Japanese FBR R&D Steering Committee, which was established by PNC, JAPC, JAERI and CRIEPI.

Kazakhstan has a nuclear scientific-industrial complex which was set up as a part of a nuclear infrastructure of the former USSR. More than 50% of the uranium resources of the former Soviet Union are in Kazakhstan, with seven uranium mines. Two UO_2 plants produced up to 35% of the total uranium in the USSR in 1990. There are extensive facilities for producing UO_2 pellets for VVER fuel elements from Russian enriched uranium. Kazakhstan has several research reactors and one operating nuclear power plant, the BN-350 fast reactor, which started operation in 1973 with a design life of 20 years. Work on its lifetime extension has the intention of bringing it into compliance with current safety standards. 1995 and 1996 were devoted to this work. In October 1996, experimental investigation on 'accident-proof' decay heat removal by natural circulation was carried out. The reactor BN-350 was restarted in February 4, 1997 at a power level of 420 MW(th).

Republic of Korea. There is a significant dependence of electricity generation on nuclear energy in the ROK, and this dependence raises the issue of the need of efficient utilization of imported uranium and of spent fuel storage. A fast reactor is thought to be a practical solution to these concerns, and the Korea Atomic Energy Commission decided to develop and construct a liquid metal reactor of 150 MW(e) by 2011. KAERI is developing KALIMER (Korea Advanced Liquid Metal Reactor), which will be the first of its kind in the country. To develop KALIMER, design methodology research is underway. The main features of the KALIMER concept are to be formally determined in July 1997. To develop this concept, a comparative study is being conducted of various possible design alternatives. The study is focused on making KALIMER more safe, economic, with less impact on the environment and more resistant to nuclear proliferation. The safety is to be enhanced by implementing an additional passive shutdown system which works on the Curie point temperature of a magnet, a primary system based on a pool, a passive reactor vessel cooling system whose structure is very simple and which is reliable. To improve the KALIMER's economics, the system is using seismic isolation device, electro-magnetic pumps, simplified safety grade reactor heat removal system (RHRS) and metal fuel. The use of metal fuel will also make KALIMER more resistant to nuclear proliferation. The high thermal efficiency of KALIMER also improves the economics and makes for less discharge to the environment. After determining main features of KALIMER, the basic design work will be carried out over the next five years, until 2002. During the first three years, the KALIMER's system design will be framed out quantitatively, mainly for NSSS and the BOP systems physically connected to the NSSS. In the remaining two years, the finished design will be evaluated for its overall performance, safety and economics. Based on the overall evaluation results, the design will be refined further and the remaining systems which are not covered during the first three years will also be designed. There is a lot of equipment and technology which require R&D for implementation of KALIMER. To name a few, the submerged electro-magnetic pump, a seismic isolator, high temperature structure design, metal fuel, passive Curie point shutdown system and the steam generator. R&D works on these items require a massive investment to the facility and it would therefore be desirable to carry out this work through an international collaboration. The collaboration on the systems engineering and methodology development is also needed to make up the lack of the Republic of Korea's experience in liquid metal reactor systems engineering.

Russian Federation. The BN-600 reactor operation is successful, its load factor being equal to 76.32%. Since its first start-up, the power unit has operated as electricity generating plant during 76.5% of time, 21.1% of time having been spent on refuelling and planned repair work, and 2.3% of time was taken by unscheduled reactor shutdowns. Currently, the BN-800 reactor design undergoes examination according to the regulations now in force in Russian Federation, in order to obtain the license to continue its construction on both Beloyarsk and South-Urals sites. In order to fulfill the construction of the 4th power unit of the Beloyarsk NPP, a public joint-stock company was established, incorporating several large enterprises and institutions whose financial potential is sufficient for accomplishing the 7-year construction. The economic analysis of the BN-800 reactor design, including the revision of some expense items is carried out, aimed at a cost reduction. Much work on the fuel and structural materials irradiation was carried out on the BOR-60 reactor. Fuel burnup of 30.5% has been achieved in some of the pilot vibro-packed MOX fuel elements. Experimental studies are under way on the BFS critical facilities, modelling various fast reactor core designs for plutonium utilization and minor actinides burning. Work on uranium-thorium fuel cycle is being carried out on the COBRA test facility. Analytical studies were conducted on the former weapons plutonium utilization in the BN-600 and the BN-800 reactors. Experimental studies have been focused on the issues of sprayed sodium burning and sodium leaks under thermal insulation of the pipelines. Also, sodium boiling and several aspects of sodium technology were studied experimentally.

Switzerland. The Fast Reactor activities are the results of two small research programmes, on liquid metal thermohydraulics and fast reactor physics, and a general evaluation of the development of advanced reactors including fast reactors. The mentioned two research programmes which had been supported by the utilities and Swiss Department of Energy were terminated at the end of 1996 due to budgetary reasons. However some of the scientific know-how from the abandoned programmes are now incorporated in the growing research activities for actinide transmutation and waste incineration. The liquid metal thermohydraulics programme centered around the investigation of mixing layers of two overlaying horizontal streams of liquid metal of different temperatures and velocities. First series of analogue experiments was performed with water. The experiments with sodium are starting now. The reactor physics work was predominately in the computational field to validate methods and data in cooperation with CEA using the European Code System ECCO/ERANOS. PSI performed an evaluation of various advanced reactor systems for the Swiss Department of Energy. The aim was to assess the development potential of various nuclear systems taking all the relevant aspects into account.

United Kingdom. The UK continues to participate in the European Collaboration, mainly funded by BNFL. Activities focus on the further development of the EFR design, on the CAPRA program led by CEA and on broader fuel cycle studies. The experiments associated with the closure of PFR have now been completed by the destructive examination of materials from the secondary circuit and sodium storage tanks. The experiments showed no cause for concern with respct to failure during longer term operation. On the first stage of decommissioning, the PFR core has been removed and replaced by dummy core. Decommissioning activity is currently concentrating on the provision of a sodium disposal plant and the supporting transfer system. Successful reprocessing of fuel, axial and radial breeders and residues from PFR has been demonstrated. In 1996 further 24 radial breeder and 4 core sub-assemblies were reprocessed; in total approximately 127 kg Pu and 3654 kg U were separated.

The European Commission continued its fast reactor research activities along the same lines as previously, but with the main emphasis on partitioning and transmutation (P&T) of long-lived radionuclides and reactor safety. The P&T work was carried out by research institutions in the Member States and by the EC Joint Research Centre (JRC) as cost-sharing actions. The JRC has also been performing its own programme through institutional and competitive research activities. The JRC institutes involved in these studies are the Institute of Systems, Informatics and Safety (ISIS) in Ispra, the Institute for Transuranium Elements (ITU) in Karlsruhe and the Institute for Advanced Materials in Petten. The implementation of P&T involves research in three areas: (i) partitioning of long-lived radionuclides from the high level waste, (ii) development of fuel and targets containing these long-lived elements in view of their (iii) transmutation in various burners (fission reactors and accelerator-driven transmutation devices). The European Commission has partly supported experimental work on partitioning both in the framework of the shared-cost program and at the JRC/ITU, fuel and target development at the JRC/ITU and an overall strategy study on the potentialities of P&T for nuclear waste management as a shared-cost action. After the high pressure load of the European Fast Reactor (EFR) Project, the interest for LMFBR severe accidents is somewhat reduced. Actually, some savings were made in the funding of computer code developments for whole core accident calculations thanks to the co-ordination activities of the Whole-Core Accident Calculation (WAC) Group which managed to converge all calculation developments in the EU towards one single code, namely the European version of SAS/4A, which has been adoped also by Japan. As an alternative solution, especially for the modelling of preirradiation and fuel pin mechanics phenomena, the computer codes TRAFIC (UK) and GERMINAL (F) have been made available to some of the WAC Group activities in view of specific calculations in parallel with SAS/4A. As a result of a common interest for developing severe accident analysis codes, a second common follow-up benchmark exercise about Russian BN-800 in its nearly-zero void reactivity version has been proposed jointly by EU and IAEA/IWGFR with the aim of also including India and Japan besides the Russian Federation. The main goal of this new comparative calculation of BN-800 encompasses these items: (i) to establish a basis so as to evaluate the characteristics of the BN-800 reactor with a nearly zero sodium void reactivity core design under hypothetical severe accident conditions, as far as an energetic unprotected loss of flow (ULOF) is concerned, and (ii) 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.

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FAST REACTOR RESEARCH ACTIVITIES IN BRAZIL

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Abstract

Fast reactor activities in Brazil have the objective of establishing a consistent knowledge basis which can serve as a support for a future transition to the activities more directly related to design, construction and operation of an experimental fast reactor, although its materialization is still far from being decided. Due to the present economic difficulties and uncertainties, the program is modest and all efforts have been directed towards its consolidation, based on the understanding that this class of reactors will play an important role in the future and Brazil needs to be minimally prepared. The text describes the present status of those activities, emphasizing the main progresses made in 1996.

1- Introduction

In 1996, the total electricity consumption in Brazil was of the order of 260 TWh (1625 kWh per capita), which represents around a 6% increase with respect to the previous year. About 95% of the electricity come from hydric resources, most of them located in the southeastern region of the country. This region, which is highly industrialized, consumes almost 60% of the total electricity produced in Brazil. It is important to point out that about half of the total hydroelectric potential is located in the Amazon Basin and its exploitation will certainly be associated with high financial and environmental costs.

Brazilian uranium resources are presently estimated in 300,000 metric tons and the estimate on the thorium resources is of the order of one million metric tons. The country has presently one reactor (PWR/626 MWe) in operation and a second one (PWR/1300 MWe) has recently received authorization for completion, which is planned for 1999. A third reactor (PWR/1300 MWe), originally planned to be constructed in the same site as the other two is still awaiting for a government decision.

The Brazilian government has kept a low profile on the complete consolidation of an electricity generation program based on nuclear energy. The reason for this resides not only upon the fact that only 25% of the hydroelectric potential is presently being used but also upon the costs involved, which is a central issue of the present discussions related to the privatization of the electric sector. The utilization of nuclear energy for "social applications" is being emphasized, in order to improve its public acceptance. In addition, the nuclear research institutions are being oriented to make their planning with an "eye in the market" , which can put longer-term programs in jeopardy.

Based on the understanding that fast reactors will play an important role in the future and Brazil needs to be minimally prepared, a research program has been initiated with the main objective of establishing a consistent knowledge basis which can serve as a support for an eventual transition to this class of reactors in the future. Although the materialization of an experimental fast reactor is still far from being decided, we felt it was convenient to prepare a reference design which could serve as a focus for several R&D activities. These activities were chosen to fit the existing experiences and were to be conducted within the limitations both in personnel and in financial resources. For this reason, efforts have been concentrated mostly on items related to the primary circuit of an experimental reactor, whose thermal power was chosen to be 60 MW.

2- Reference Design- Main characteristics

The main characteristics for the reference design /1/ were chosen in 1992, in such a way as to benefit from the most recent available literature and also from the already existing experiences. For example, the choice of metallic fuel (U-Zr or U-Pu-Zr) was based not only on the observation that these alloys were receiving great attention due to their high burnup capabilities but also on the fact that the Nuclear Research Institute (IPEN, Sao Paulo) had previous experience with metallic fuel for research reactors and became motivated in engaging in a challenging research towards the development of U-Zr alloys and metallic uranium recovery. The same line of reasoning was used for involving other research groups with HT-9 as a reference for cladding material, electromagnetic pumps, passive safety and so on.

For the reference design, fuel pin dimensions and other data were taken mostly from ALMR (former PRISM) and EBR-II and were used for calculations which led to a general core configuration. The reference design, whose main characteristics are indicated in Table 1, is presently being used for testing our calculational tools, for checking different methodologies for core calculations, etc.

Table 1: Main Design Characteristics

Thermal power (MW)	60
Electric power (expected) (MW)	20
Primary operating pressure (MPa)	0.11
Core inlet temperature (° C)	370
Core outlet temperature (° C)	470
Maximum burnup (MWd/T)	70000
Fuel type	U-10%Zr
Coolant	Sodium
Primary coolant circuit arrangement	pool type
Cladding material	HT-9

The concept of the fuel subassemblies for the REARA-60 reference design was based on the international experience, with the dimensions taken mostly from ALMR. The number of fuel pins in the subassembly is 61, separated by helicoidal wire wraps and arranged in a triangular matrix, forming a hexagonal set. The fuel is a U-10%Zr metallic alloy with active length equal to 62.0 cm and having two (top and bottom) 30.0 cm nickel reflectors.

In the core there are 6 control subassemblies (4 primary+2 secondary). The control material is boron carbide (B_4C) enriched in ^{10}B, with the pellets being clad in stainless steel cylindrical tubes. In the present reference design the control subassembly inserting and withdrawing mechanisms have not been defined.

Consideration has been given to the utilization of some types of special subassemblies distributed in the reactor core, in order to provide means of controlling neutron leakage, protecting against loss of flow accidents, controlling the excess of reactivity and so on. Among them we have the Gas Expansion Modules (GEM), which have been tested in the Fast Flux Test Facility (FFTF) and are to be located (3 subassemblies) in the active core periphery in order to protect against loss of sodium flow.

The core arrangement which has been used for calculations is indicated in Figure 1. The target average burnup is 70 MWd/kg, which will require about 1000 days of full power operation. Using cross section data from the Japanese library JFS-2 (JAERI-Fast Reactor Group Constant Set - Version II, 70 energy groups) and the diffusion theory codes EXPANDA (generates weighting spectrum in 70 groups and uses it to collapse cross sections do 6 groups) and CITATION (RZ ge-

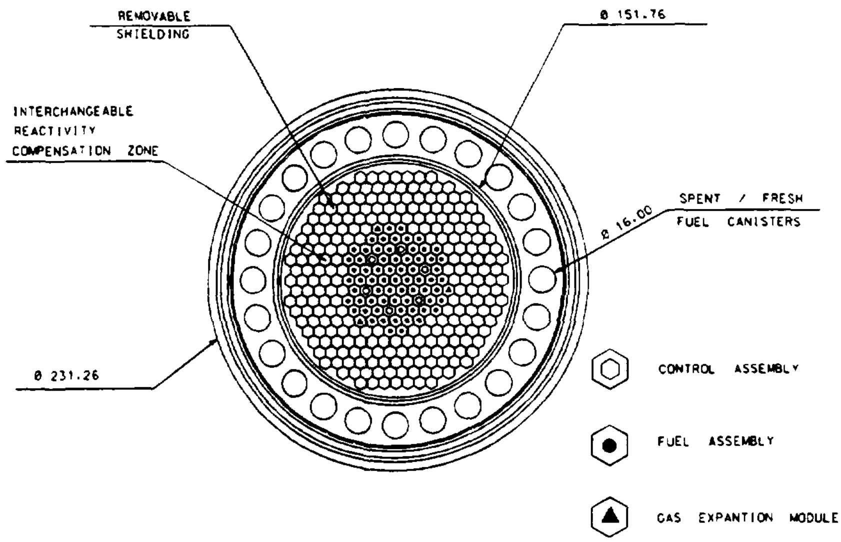

Figure 1: Core arrangement at BOL

ometry, 6 energy groups), evaluation of some reactor parameters have been performed at the Advanced Studies Institute (IEAv, Sao Jose dos Campos) and are indicated in Table 2.

Table 2: Main core parameters

Maximum flux (10^{15} n/cm^2.s)	1.97
Average flux (10^{15} n/cm^2.s)	1.07
Doppler coefficient (-TΔk/ΔT)	7.07×10^{-4}
Sodium void reactivity (-Δk)	3.27×10^{-2}

Also based on data selected from the literature on EBR-II and ALMR, a set of reference values for thermohydraulics calculations have been fixed and used for the reference design of the Intermediate Heat Exchangers (IHX), which are straight shell/tube (primary coolant in shell) and counterflow type. REARA-60 has two identical IHXs, each one associated with two pumps, whose characteristics are not defined yet.

The IHX main characteristics, shown in Table 3, were determined with the computer program TCIPRO /2/, which is written in C-language and runs in PCs.

Table 3: IHX main characteristics

Primary sodium flow rate (kg/s)	226.0
Secondary sodium flow rate (kg/s)	152,0
Number of tubes (#)	1397
Central pipe OD (m)	0.311
Total heat transfer area (m^2)	233.0
Total height (m)	3.859

3- Research and Development Activities

3.1- Reactor physics

Independent core calculations performed at the Nuclear Engineering Institute (IEN, Rio de Janeiro) pointed out to discrepancies -with respect to calculations done at IEAv- which were mainly attributed to the differences in nuclear data sets.

Under a formal cooperation agreement, IEAv and IEN are presently working together on a series of sensitivity analyses with the objective of developing a better understanding of those differences, in order to choose the best route for more refined calculations, if necessary.

3.2- Heat transfer/removal

A small natural circulation circuit using water as the circulating fluid has been recently built in the Department of Mechanical Engineering of the Aeronautics Technological Institute (ITA, Sao Jose dos Campos), as part of a formal agreement between IEAv and ITA. Using an 8 kWt heat source, the circuit will be used for studying basic phenomena associated with natural convection, for experimental validation of theoretical models and so on. For being installed in an engineering school, the circuit will also be useful for academic purposes.

Also in the area of heat removal, an exchange with researchers from the Indira Gandhi Centre for Atomic Research (IGCAR, India) has been recently initiated in order to identify topics of common interest for an eventual cooperation. This exchange has been facilitated by the visit to Brazil of Dr. Placid Rodriguez, Director of IGCAR.

3.3- Structural materials

In the area of structural materials, experimental research is underway with the objective of understanding the basic processes involved in the production and characterization of HT-9, which has been taken as a reference for cladding material. A first sample has shown a slight deviation with respect to the composition used as a reference and a second sample, produced in improved conditions, is presently being analyzed.

3.4- Uranium recovery by electrorefining

Under a formal cooperation agreement with IPEN, uranium recovery by electrorefining techniques is being investigated, also to understand the basic processes involved.

In a first experiment, which has been useful for identifying some critical items which are part of the processes, a basket containing almost 100 grams of metallic uranium was introduced in a five liter cylindrical electrolytic cell. Temperature has been raised to 500 °C and the basket was immersed in a mixture composed by the electrolyte (59mol%LiCl+41mol%KCl) and the anode (metallic cadmium). The uranium dissolution process has been maintained for 72 hours, under an argon atmosphere, and resulted in 94,45 grams of dissolved uranium. For the U-electrodeposition the cathode was maintained rotating at 50 rpm, but the experiment was forced to terminate after about two hours due to unexpected operational problems. For this reason, the mass of uranium electrodeposited (in dendritic form) was only a few grams.

In spite of the problems, for our purposes this experiment has been useful for an initial understanding the role of parameters involved in the process, in order to optimize the conditions in which the next experiments are to be performed.

3.5- Electromagnetic pumps (EMP)

The acquisition of experiences in modeling and designing of electromagnetic pumps (EMP) is one of the objectives of present activities. Experimental research has led to a first EMP which was successfully tested -using mercury as working fluid- at IEAv. This pump used an Iron C-type magnet with a continuous-current coil to obtain the required magnetic induction. A second pump, using rare-earth permanent magnets (Sm-Co) in place of the C- magnets in order to reduce the geometric dimensions is also finalized and is being prepared for evaluation of its characteristic curves. R&D work on alternating current EMPs is planned to be initiated in 1997.

3.6- Sodium technology

In 1981, a contract with ANSALDO/NIRA (Italy) was signed to design and construct three sodium loops and auxiliary systems. Due to insufficient funding the loops were never constructed. In 1996 funding has been finally provided for assembling the first loop (named SS-10), which is to be used as a test bed for experiments related to sodium purification and transferring. The loop will be constructed in IEN and completion is expected by mid-97.

3.7- Heat pipes

An investigation on the substitution of the heat exchanger internal tubes by heat pipes has been initiated based on theoretical modeling, with the objective of evaluating if there are gains which could be achieved. The first discussions have resulted in the basic concept showed in Figure 2, which

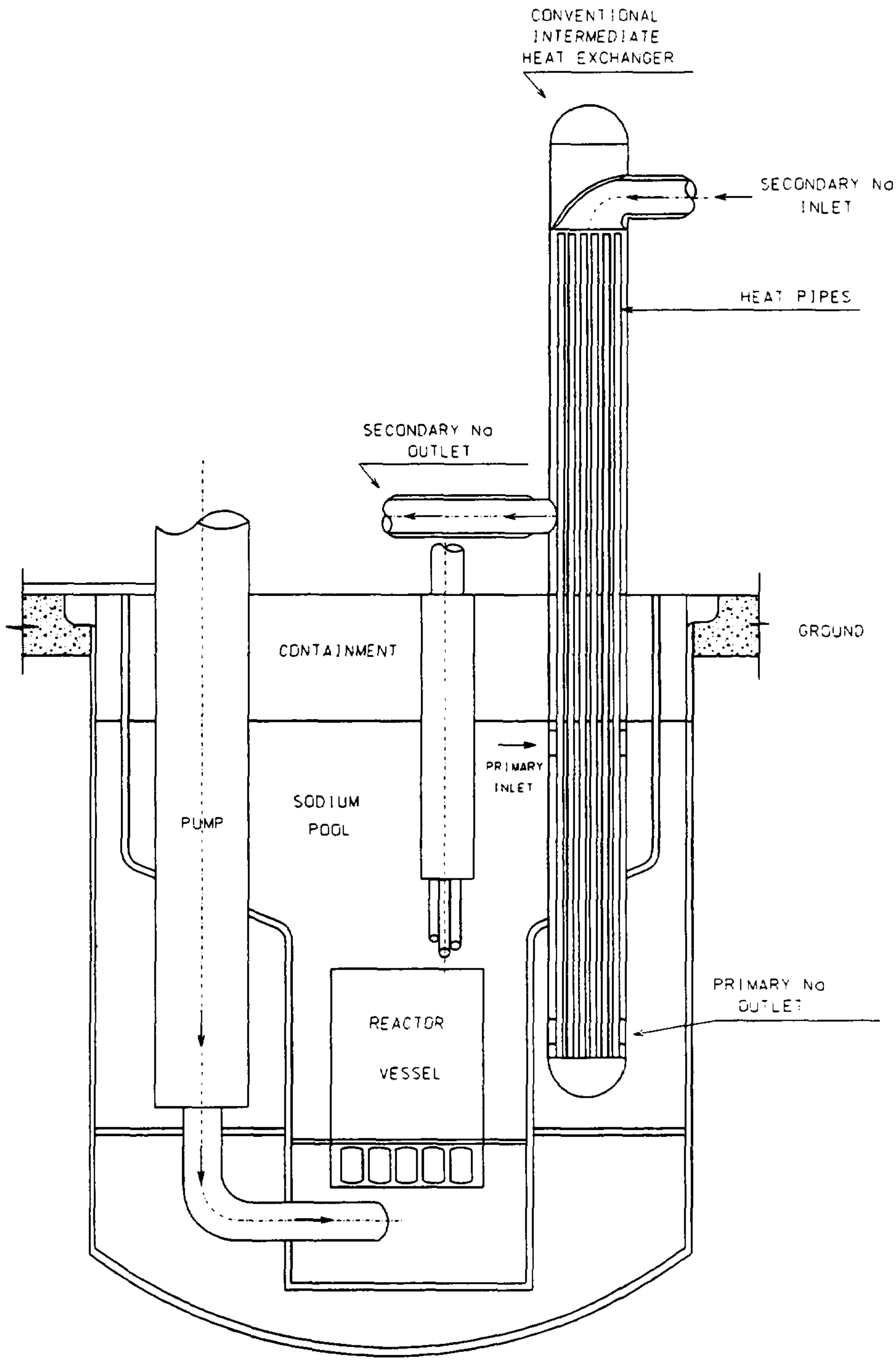

Figure 2: Schemactic view showing IHW with heat pipes replacing internal tubes

is to be used for the initial calculations. This concept indicates, among other things, that safety levels can be increased because the chances of accidental contact between primary and secondary sodium are drastically minimized. As a result, the elimination of the intermediate heat exchanger from the primary circuit could, at least in principle, be possible. This and other possibilities will have to be supported by calculations and further discussions, which are planned to be continued during 1997.

Along with these research activities, there are others involving studies on shielding analysis and actinide burning calculations.

4- Final Remarks

As previously mentioned, REARA-60 is being used as a reference for R&D activities in a few areas. Considering that there is no decision as to its materialization, yet for some time in the future the reference design will be our "experimental installation". The present priority is the consolidation of this research program and all efforts are being made for showing that a long-term program will not only result in the materialization of the reactor, but can also generate, along its way, relevant expertise which can also be useful for other sectors or programs.

Discussions on the "optimization" of our activities have also occupied part of our time. How can we effectively contribute to the international efforts in the fast reactor field, given the fact that brain power and investments for small installations and experiments are the only assets easily available at the present? Is it enough to be involved with research activities which are compatible with our investment capability, for reporting them in the IWGFR annual meetings? Or should we take part in tasks which are either part of an international fast reactor project or of a broader international cooperation?

In our viewpoint, the consolidation of our program depends not only upon convincing Brazilian decision-makers about the importance fast reactors may have in the future, but also upon showing that the results we produce can be useful to the fast reactor field, on a broader scale.

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THE STATUS OF FAST REACTOR TECHNOLOGY DEVELOPMENT IN CHINA

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Abstract

The paper will outline the main activities on fast reactor technology in China.

In the year 1996, with the increasing of about 15 GWe installed electricity capacity , the total national electricity generation capacity has reached 225 GWe in the Country. Two nuclear power plants, Qinshan Phase 1 and Daya Bay have their rather good operation. The load factor of Qinshan Phase 1 was 84.7%. 76.1% and 64.1% for Daya Bay Unit 1 and Unit 2 respectively. During the Ninth 5-year (from 1996 to 2000) four NPPs Consisting of eight units of installed 6620MWe will be constructed. Under the framework of the High Technology Programme the Chinese Experimental Fast Reactor (CEFR) with the power 65MWth matched with 25MWe turbine-generator is still under preliminary design stage, which is sodium cooled pool type, $(Pu,U)O_2$ as fuel , in -core primary spent fuel storage, two mechanical pumps and four intermediate heat exchangers for primary circuit, two loops for secondary circuit, two independent immersed heat exchangers and air coolers with high stacks for passive residual heat removal system. Some design changes are presented in the paper.

Concerning the R&D for the CEFR, besides the facilitics already prepared, for demonstration of thermohydraulic characteristics of natural convection, a water simulation reactor pool facility in about one third scale is planned, in order to prepare the reactor physics experiments for its start-up, the zero power fast neutron facility with 50kg U-235 has been restored, for endurence testing of core subassemblies and getting some sodium loop operation experiences, Italian ESPRESSO and CEDI are under reconstruction in our lab.

As for the engineering preparation of the project CEFR, the Feasibility Study Report was appoved by Authorities on November last year. The site preparation and the design of incorporated to grid are just started.

Finally, the activies of the international cooperation are presented in the paper.

1. INTRODUCTION

As a developing Country, China has a rather fast development in her national economy during past more than ten years. In last year, 1996, the annual increasing rate of the gross national products was 9.7%.[1] With the incsease of about 15 GWe[1] new installed electricity capacity, the total national electricity generation capacity has reached 225 GWe, ranking fourth place in the world. But the possession of electricity capacity per capita was less than 0.2kW, much lower than the average level over the world. The electricity generation industry should be further developed to meet the needs of the national economy development and a continuous improvement of living standard.

In this country two nuclear power plants, Qinshan Phase 1 300MWe PWR and Daya Bay twin 900 MWe have their rather good operation record during last year. The load factor of Qinshan Phase 1 was 84.7%, for Daya Bay Unit 1 and Unit 2 76.1% and 64.1% respectively.

During the Ninth 5-year (from 1996 to 2000)four NPPs consisting of eight units, of installed capacity of 6620 MWe[2] will be constructed, which are presented in Table 1.

Table 1. NPPs CONSTRUCTED AND PLANNED

NPP	Power (Mwe)	Reactor	First concrete	Connected to grid	Commercial Operation
Qinshan	Unit 1 600	PWR	1996-06-10	2002-06	
Phase 2	Unit 2 600	PWR	1996-04-01		
Lingao	Unit 1 984	PWR	1997-05-15	2003-04-15	2003-07-15
	Unit 2 984	PWR	1997-01-15	2003-12-15	2004-05-15
Qinshan	Unit 1 720	PHWR			
Phase 3	Unit 2 720	PHWR			
Jiangsu	Unit 1 1000	PWR			
	Unit 2 1000	PWR			

It is envisaged that new nuclear power plants of about 12 GWe will be built by the year 2010, By that time the total capacity of installed NPPs will be 20.86 GWe, estimated 3.5% of the total national installed capacity of electric power. The nuclear power installed capacity is further envisaged to be 40 ~ 50GWe approximately 6% of the total installed capacity in 2020[2].

Under the framework of the High-Technology Program the Chinese Experimantal Fast Reactor(CEFR) with the power 65 MWth matched with 25 MWe turbine-generator is still under preliminary design stage. The main design characteristics of the reactor are as following: pool type for primary circuit arrangement , $(Pu,U)O_2$ as fuel, but first loading will be UO_2, austenitic stainless steel as core structure material, in-core primary spent fuel storage, two mechanical pumps and four intermediate heat exchangers for primary circuit, two loops for secondary circuit, two independent immersed heat exchangers and air cooler with high stack for passive residual heat removal system.

As for the engineering preparation of the project CEFR the Feasibility Study Report was approved by the Authorities on November last year. Site preparation including buying necessary territory from local Government and design of connected to the grid are just started.

2. CEFR DESIGN

After the collection and preparation of necessary computer codes and the completion of main technical selections , the conceptual design of the CEFR was started in 1990 and completed in 1993 including the confirmation and optimisation to some important desgn selections and characteristics. Having spent almost whole 1994 for its preparation, the CEFR preliminary design has been started in the early of 1995, and will be completed in the middle of 1997. Then, a review and authorisation to the CEFR preliminary design by the Authorities should be followed.

2.1. Changes of Design Selections

It has been settled that some basic design technical selections were changed in the preliminary design if compared with its conceptual design. The main changes of CEFR design technical selections are support configuration of the reactor vessel, containment form and buildings layout and sodium flow distribution measures for core subassemblies.

2.1.1. Reactor Vessel Support

During the CEFR conceptual design and design study phase it's emphasized to search into a top hanged or top supported for reactor vessel support, and which was decided to adopt for CEFR, considering three main reasons. First, top supported is simpler structually than bottom supported,

and could avoid to use bellows for components, which are a little fragile.
second, components i.e. primary pumps, intermediate heat exchangers, decay heat exchangers etc. could be supported from the reactor roof, namely hanged from the top, a suitable radial restrain and axial expansion tolerance could be given. So top supported is a little easier than bottom supported. and third, more experiences for top supported in other countries.

Considering the possible earthquake intensity, CEFR site is localed in the region of 7 degree, near 8 degree in standard 12 degree classification, the horizontal acceleration for safe operation S1 will be 0.107 g on the base rock and for safe shutdown S2 will be 0.214 g. The analysed results to the both axial and peripheric stresses of the reactor vessel (main sodium tank) under these two circumstances that the maximum of peripheric stress at bottom supported is only half of that of top supported. and also we understand the bottom supported has been well realized in BN-600. These are the main reasons to make this change.

2.1.2. Containment Form and Buildings Layout

In conceptual design a cylindrical concrete building was selected, fuel handling building, steam generator building, main control room and diverse service building and radioactive wastes management building were separately arranged with respect to the reactor building.

Due to the same reason of possible strong earthquake in this region, it is easier to treat, in architecture and safety point of view, a complete concrete base, not so large, involving all buildings or structures of safety class. So in the preliminary design, the related design selection has been changed to 36 × 36 m square reactor building surrounded by a square ring building with abovementioned diverse functions to form 63 × 63 m square building.

2.1.3. Sodium Flow Distribution Measures for Core Subassemblies

For sodium flow distribution of core subassemblies a diverse fixed-number orifices configuration on the core support plenum was designed in the conceptual design, in which all central subassemblies which need forced flow by high presure sodium were same subassembly foot.

A disadvantage of this flow distribution measure is much difficult to decide in design the gap between subassembly foot and support tube with fixed-number orifices. It is hardly to control the right flow rate when the gap is larger, and there is some risks of blockage or stick between foot and tube when it is smaller.

A new flow distribution measures have been selected in preliminary design, in which the fixed-number orifices are only on the subassembly foot. As its penalty different subassembly foot is needed for 4 zones of core; It's impossible to change subassemblies between diffenent zones, a proper mechanical approach to be used to surely exclude wrong loading; more troubles are induced in the fuel handling management and more spare subassemblies are needed.

2.2. Main Results from CEFR Preliminary Design

Besides some changes of main technical selections, some design parameters and technical choices have been updated in the preliminary design. Its brief is given in following paragraphs.

2.2.1. Reactor Core

As shown in Fig 1, the CEFR reactor core is composed of 81 fuel subassemblies with $(Pu,U)O_2$ fuel, which are divided into 4 zones with same fuel composition and different subassembly foot by zone to zone in order to supply sodium flow rate according to the power distribution. There are 8 control and safety subassemblies in which 3 safety and 3 compensation subassemblies containing B_4C enriched with B-10 to 92% and two regulation subassemblies with natural boron carbide. The fuel region is surrounded with 336 stainless steel subassemblies which has 3 types due to their power and necessary cooling ability. And then, embraced by 230 graphite-boron shielding subassemblies and 56 primary storage position for spent fuel subassemblies. The center to center distance of neighbouring subassembles is 61.5mm.

In each fuel subassembly there are 61 fuel pins of 6.0 mm diameter in triangular arrangement, using wire wrapped positionning to keep 7.0 mm lattice pitch of fuel pin. Each fuel subassembly composed of a handling head matched with fuel handling devices, hexagonal tube in which the fuel pins are included and a cylindrical foot playing the role of positionning and introducing of sodium coolant. The structure drawing of the fuel subassembly is given in Fig 2.

The reactor is equiped with two independent shut down systems, first of which is composed of 3 compensation and 2 regulation subassemblies with a rapid drop down time less than 1.5 seconds, the second shut down system includcs 3 safety subassemblies with a rapid drop down time less than 0.7 seconds.

It is decided to use UO_2 as first loading of the core, the enrichment of the uranium-235 should be 60.5% based on conceptual design.

2.2.2. Reactor Block

The CEFR reactor block as shown in Fig 3 is composed of a reactor top deck in which there are mainly double rotating plugs, two primary pumps, four intermediate heat exchangers and two decay heat exchangers, and also composed of main and guard vessels and internal structures on which the reactor core and its grid plenum, the abovementioned components belong to the primary circuit and shielding components are supported. A small gap of 87.5 mm between main vessel and guard one is designed to ensure that the reactor core will be still immersed under the primary sodium and the natural convection of primary sodium will be still kept when a leakage on the main vessel and the lowest primary sodium level kept by the guard vessel have happened. For this concept the disadvantage is very difficult to realize inspections with an inspector movable on the surface of the main vessel, only leak detectors and thermo-couples are equiped.

The double rotating plugs are traditional in which control rod mechanisms, core-above-structure are mounted on small plug. A straight moving fuel handling machine is eccentrically mounted on the small plug too. With the double rotating plugs and the fuel handling machine, any movement of subassembly could be realized between any two core subassembly positions or between a core subassembly position and a transfer pot which is connected with two slant transfer machine and a transfer elevator by which the subassemblies could one by one enter into or go out the reactor vessel.

2.2.3. Heat Transfer System

The primary circuit in the reactor main vessel includes two prinary pumps and four intermediate heat exchangers. In cold pool a pump with its neighbouring two intermediate heat

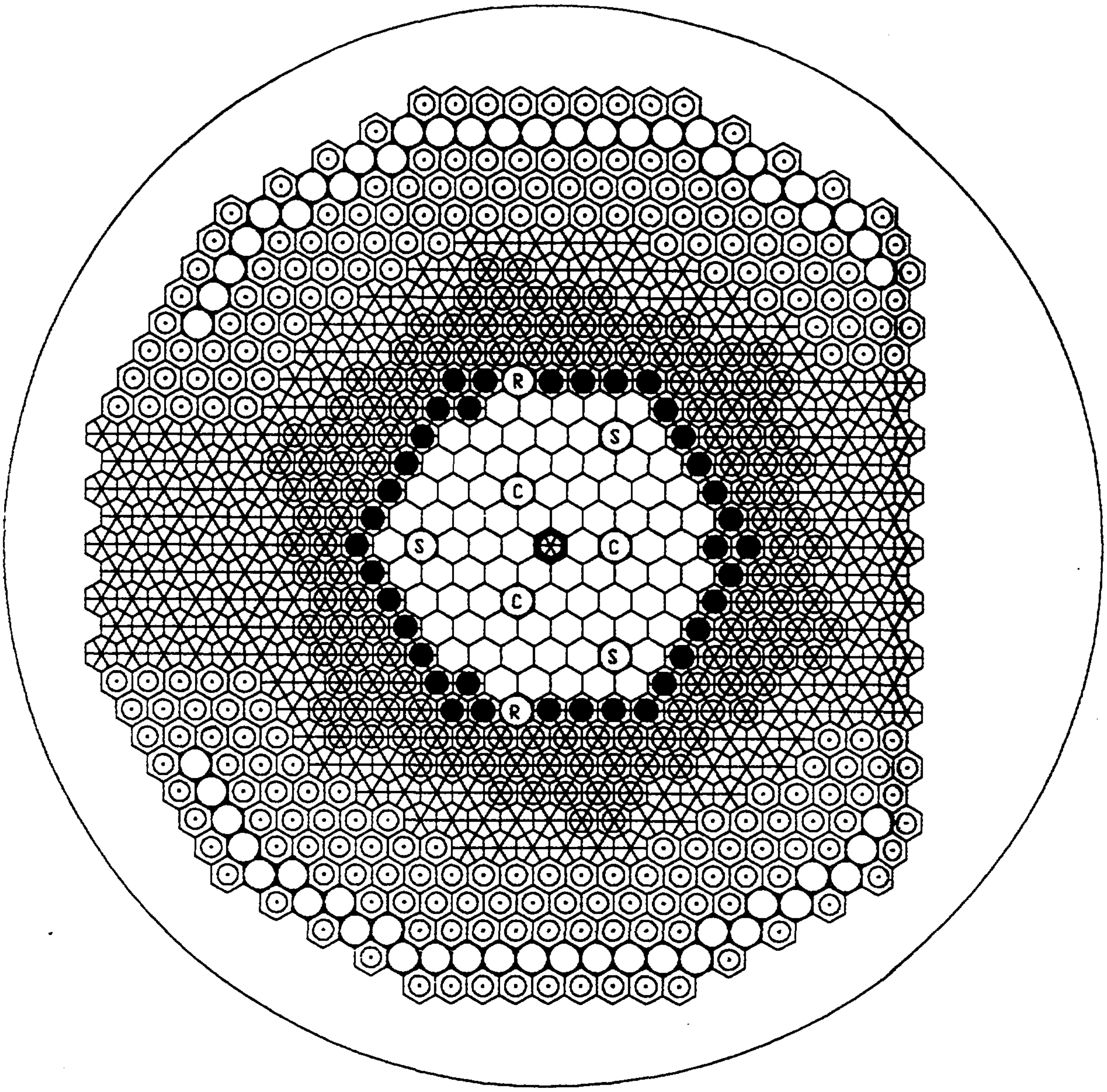

FIG. 1. CEFR core.

	Fuel subassembly	81
	Stainless steel rod	1
	Stainless steel reflecter subassembly	37
	Stainless steel refleetor rod	132
	Stainless steel refleetor rod	167
	Sheilding subessembly	230
	Storage position for spent fuel subessonbly	56
S	Safety subassombly	3
R	Regulation subassombly	2
C	Compensation subassembly	3

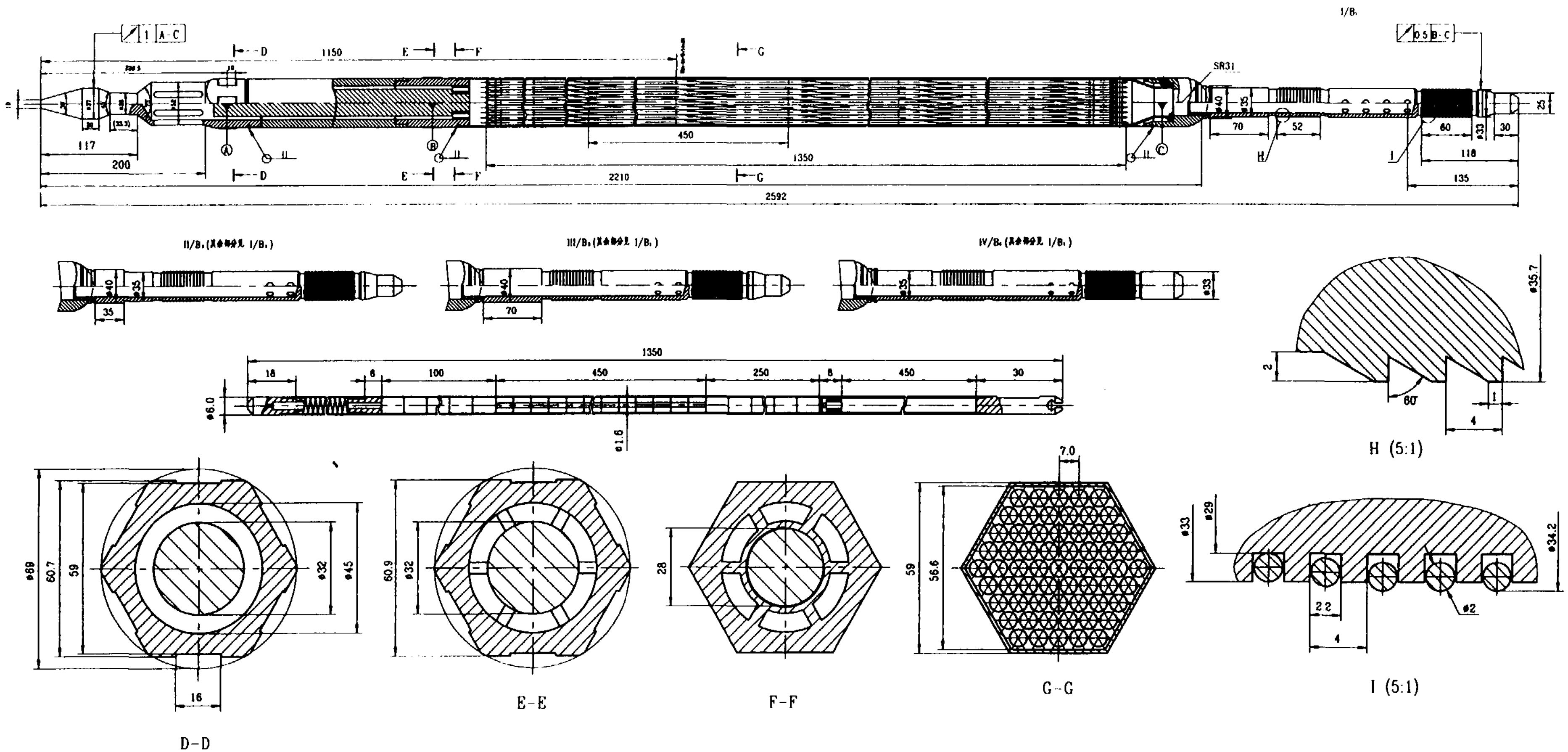

FIG. 2. CEFR fuel subassembly.

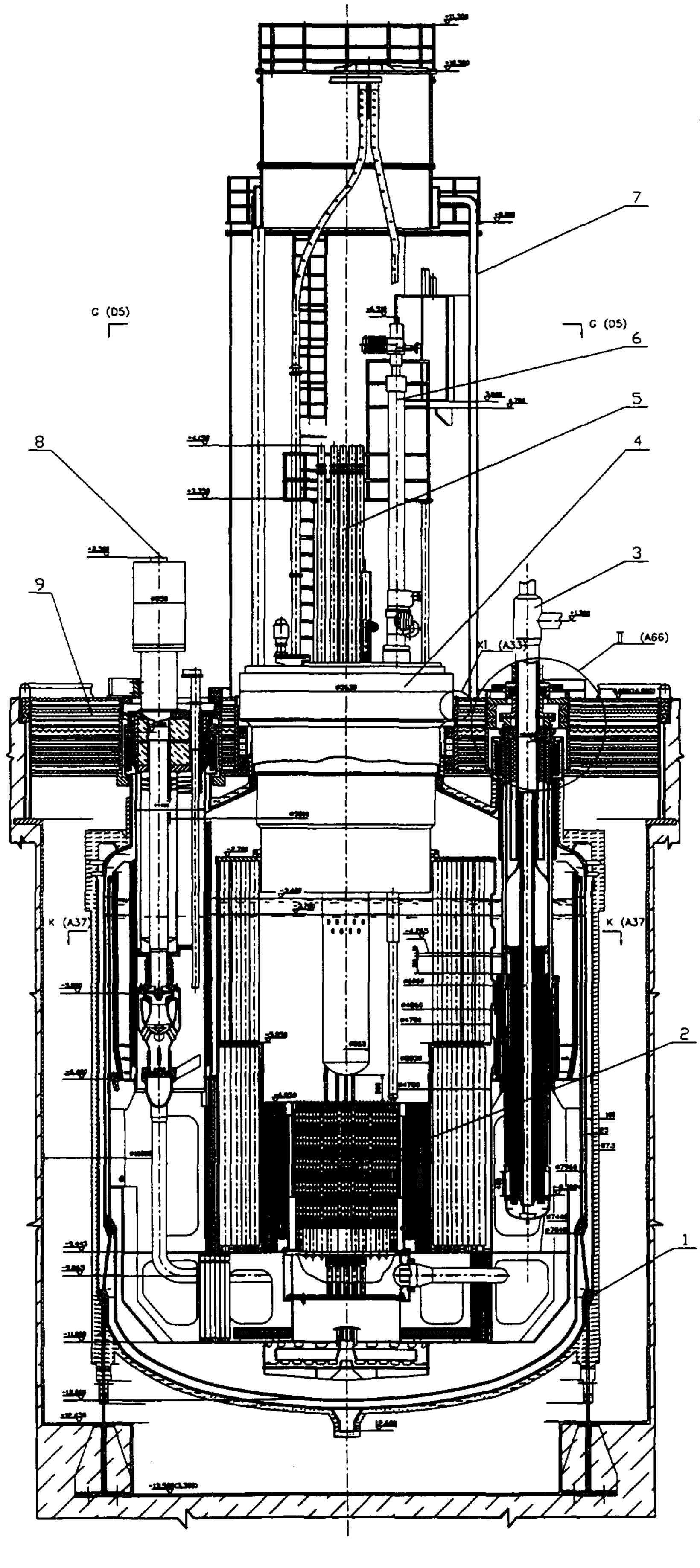

FIG. 3. CEFR reactor block.

exchangers as a group is separated by the plate with another group, but in hot pool they are connected. In the normal operation, the 360 ℃ sodium from cold pool is pumped into the grid plenum through high pressure pipe, the sodium temperature will be 530 ℃ when it leaves from the reactor fuel region and then mixed with sodium of hot pool, it enters into the intermediate heat exchangers with the temperature about 516 ℃ and leaves them with the temperature 353 ℃.

The secondary circuit is composed of two loops, each one includes one secondary pump , one steam evaporator, one steam-water separator, one superheater, one expansion tank, valves and draining tank and emergency draining tanks. The steam generators will produce 96.2t/h dry steam with the temperature 480 ℃ and the pressure 14 MPa for a 25MWe turbine generator. The heat in the condenser will dissipate to the air by a cooling tower.

2.2.4. Residual Heat Removal System (RHRS)

Two independent passive residual heat removal systems as one of specific safety installations are equiped in the reactor, each one includes one residual heat exchanger immersed in primary sodium or named independent Na-Na heat exchanger and one sodium-air cooler with one high stack. The normal power of each system, namely at its service condition, is 0.525 MWth. but , at its stand-by condition the power lost will be 0.052 MWe. The design parameters are given in Table 2

Table 2 RHRS DESIGN PARAMETERS(ONE SYSTEM)

Parameter	Unit	Condition, Service	Condition, stand-by
Power	MW	0.525	0.052
Primary Na flow	kg/s	3.7	
Secondary Na flow	kg/s	2.2	
Air flow	kg/s	2.05	
Primary Na Inlet Temp.	℃	514	514
Primary Na Outlet Temop.	℃	400	472
Secondary Na Inlet Temp.	℃	310	472
Secondary Na Outlet Temp.	℃	495	512
Air Inlet Temp.	℃	-30 ~+50	-30 ~+50
Air Outlet Temp.	℃	≤300	≤493

2.2.5. Containment

According to the nuclear safety rules and guidance published by Chinese National Nuclear Safety Administration, the containment of the reactor should be as the final barrier for radioactive materials and against external incident. The reactor containment concept adopted in CEFR is primary containment and secondary containment combined.

The primary containment is composed of reactor concrete pit with steel layer and a reactor top dome with its steel structure. They are separated by the reactor top deck, airtighted, and with independent emergency ventilation system with decontamination function.

The secondary containment is of 36 × 36 m concrete building with flat bottm and quasi-half sphere shell cap, the wall thickness will be about 1.2 m, Design parameters are as following:

Design Pressure internal or external: 0.07MPa

Leak rate under 0.01MPa: 5%ΔV/V in 24h.

2.3. Main Design Parameters

Parameter	Unit	Conceptual Design	Preliminary design
Thermal Power	MW	65.5	65
Electric Power, net	MW	20	20
Reactor Core			
Height	cm	50	45.0
Diameter Equivalent	cm	58.5	60.1
Fuel		$(Pu,U)O_2$	$(Pu,U)O_2$
Pu,total	kg	121.6	111.0
Pu-239	kg	93.2	67.7
U-235(enrichment)	kg	97.6(30%)	95.8(36%)
Linear Power max.	W/cm	430	430
Neutron Flux	$n/cm^2 \cdot s$	2.97×10^{15}	3.7×10^{15}
Burn-up, target max.	MWd/t	100000	100000
Burn-up, first load max.	MWd/t	50000	60000
Inlet Temp of the Core	℃	360	360
Outlet Temp of the Core	℃	530	530
Diameter of Main Vessel	m	8.0	8.010
Primary Circuit			
Number of Loops		2	2
Quantity of Sodium	t	~ 300	~ 300
Flow Rate, total	t/h	1400	1465(1710m^3/h)
Number of IHX per loop		2	2
Secondary Circuit			
Number of loops		2	2
Quantity of Sodium	t	48.2	48.2
Flow Rate	t/h	986.4	977
Tertiary Circuit			
Steam Temperature	℃	480	480
Steam Pressure	MPa	10	14
Flow Rate	t/h	93.5	96.2
Plant Life	a	30	30

3. RESEARCH AND DEVELOPMENT

The funding offered to the R&D of CEFR are not sufficient during past years. Only limited prograss has been gained last year, which is related to sodium technology, sodium test loops and fast neutron zero power facility.

3.1. Sodium Aerosol and Its Removal

Aerosols generated from sodium fires in oxygen riched atmosphere are the source of considerable hazards associated with radioactivity released if the sodium fire is from primary sodium systems, with poor visibility through aerosol clouds and aggressive corrosion burden on structural materials. Therefore, an effective method reducing the aerosol release should be designed through proper ventilation and filtration systems of the CEFR. The experiments are conducted for studying the removal of sodium combustion aerosols by water-spray.

The sketch of the experimental equipments is shown in Fig 4.

The whole process of a single experiments is described as following:(1) put some sodium (about 93 g) in a stainless steel cup on the bottom of the container. (2) heat the container with the heater, when the sodium is about to be ignited, blow the air into the container, so that the sodium burns thoroughly. (3) take aerosol samples from the outlet of the water-spray column and analyze them.

Some results have been obtained under these experimental conditions;

(1)Sodium combustion rate is about 22 ~ 27 $kg/m^2 \cdot h$

(2)The output of aerosol is about 7.5 ~ 10.0%

(3)Any unnecessary bend should be eliminated in order to avoid more aerosol deposited on the bend pipe.

(4)Water-spray is proved to be a good method of removing sodium combustion aerosol, its maximum efficiency is about 70%

(5)Other method, such as filters, of moving sodium aerosol should be adopted after water spraying to remove the remaining aerosol.

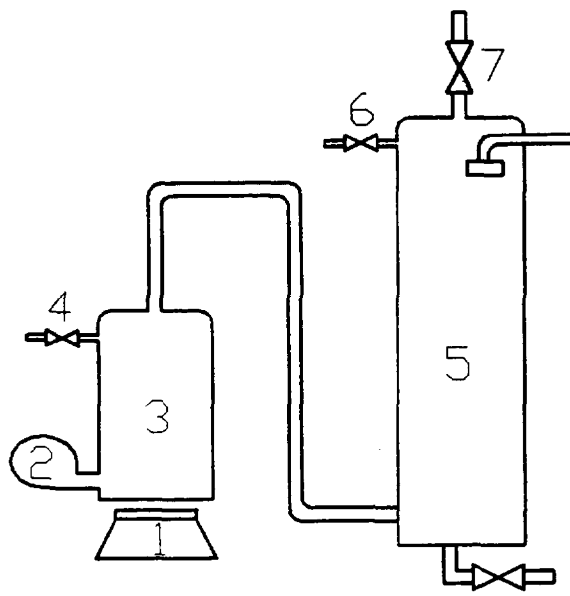

FIG. 4. Sketch of aerosol test facility.

1. Heater
2. Blower 1.5 - 2.0 m^3/h
3. Sodium container, Φ234 mm x 600 mm
4. & 6 : Sampling pipe
5. Water-spray column, Φ 169 mm x 1520mm
7. Outlet
8. Flow meter
9. Water tank
10. Nitrogen bottle

3.2. Sodium test loops

Two sodium test loops, ESPRESSO and CEDI, 110m^3/h and 320m^3/h respectively, ceded from ENEA, Italy as a gift of the technical cooperations are under reconstruction. It is planned to use ESPRESSO loop for the endurence test and thermoshock test of the CEFR core subassemblies' models, and CEDI loop will be used for proving of some sodium components if necessary. In this case much modifications will be done. Another important role of both loops is to provide the necessary experiences on sodium handling and loop operation in such big scale, considering the sodium loops constructed before in China are all in small scale of less than 40m^3/h flow rate.

As the intended schedule. ESPRESSO will be filled with sodium in the end of this year, and operated in the early of 1998. CEDI will be one year later for only pure sodium loop operation. A group of CEFR operators will be trained at first on both loops.

3.3. Fast Neutron Zero Power Facility

A fast neutron zero power facility with only 50kg U-235 has been built up in 1970, then moved to the South-West center of Reactor Engineering in Sichuan Province. Basic zero power physics experiments have been done at this facility including critical parameter measurements, fission rates, neutron flux distribution, neutron spectrum, material reactivity etc. in 1988, it was removed to CIAE again, and now it has been rebuilt and will be used for proving of the neutronics experiment methods which will be served to CEFR first start-up and to primary test for the neutronic and other radiation detectors. It is considered also it will be valuable to the evaluation of some specimen nuclear cross section using its hard spectrum.

4. CEFR ENGINEERING PREPARATION

Feasilility Study Report has been approved by the Authorities in November last year. It means that CEFR is becoming an annual project. In parallel to the preliminary design the preparation of following 5 reports are started: Prelimingry Safety Analysis Report, Environment Impact Analysis Report, Professional Health Report, Radiation Safety Report, and Fire Extinguishing Report which should be provided to coorespondent Authorities at different stages.

The application and negotiation to and with local government for buying the additional site territory which is about 10 ha is just started. and design of connected to the grid has been started, and the negotiation with the electrical grid Administration is under going .

5. INTENATIONAL COOPERATION

During past years, China has had her some cooperations with some countries on the fast reactor technology fields.

As technical presents, the Italian Energy, New Technology and Environment Agency has ceded some research facilities and components including ESPRESSO and CEDI, sodium purification facilities, core subssembly dimension meter etc. Now they are still under reconstruction, and its clear that the experiences of big sodium loop construction and operation in future are much valuable to our teams who will take part in the CEFR construction and operation.

Under the framework of the reactor R&D cooperation between China National Nuclear Cooporation and Commissariat de `Energie Atomic de France, we have good cooperation on fast reactor technology including computer codes cooperation, ceding sodiun components and instrumentations as gifts to China Institutc of Atomic Energy, information exchange on sodium technology and on neutronic physics which are all going fluently.

The cooperations with Russian FBR Institutions have been conducted during past years including training to CIAE' engineers, experts' lectures, CEFR design consultancy and zero power facility experiments etc. Some design experience for fast reactors is gained from this cooperation.

IAEA, as the International Agency stimulating peaceful use of atomic energy, has given much helps to China in the fields of FBR technology development including the informations and financial support to the participation of technical committee meetings.

Chine has a rather weak bases for the development of fast reactor technology, it is intended to have more cooperations with other countries to share the experiences.

6. CONCLUSION

The fast breeder reactors have been developed for at least five decades in the word up to now. Some countries have stopped their commercialized fast breeder reactor development due to the electricity demands are approximately saturated, the Uranium market is still sufficient to meet their nuclear power, and especially, the conventional energy resources have not been exhausted in next some decades. For these countries, the estimation of the energy needs in the early of 70s is tolally diffirent with today's reality.

As a developping country, China has a ambitious demands to energy resources in next several decades. At that age will China have the same situation like today's situation of other developped countries? the answer will be negative. China will not have enough suitable and, economical conventional resources. except nuclear. It is obviously that nuclear power in large scale needs fast breeder reactors, which just is the case of China

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FAST REACTOR DEVELOPMENT PROGRAMME IN FRANCE

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Abstract

Firstly the general situation regarding production of electricity in France is briefly described. Then in the field of Fast Reactors, the main events of 1996 are presented.

At the end of February 1996, the PHENIX reactor was ready for operation. After review meetings, the Safety Authority has requested safety improvements and technical demonstrations, before it examines the possibility of authorizing a new start-up of PHENIX. The year 1996 was devoted to this work.

In 1996, SUPERPHENIX was characterized by excellent operation throughout the year. The reactor was restarted at the end of 1995 after a number of minor incidents. The reactor power was increased by successive steps : 30 % Pn up to February 6, followed by 50 % Pn up to May then 60 % up to October and 90 % Pn during the last months. A programmed shutdown period occurred during May, June and mid-July 1996. The reactor has been shutdown at the end of 1996 for the decenial control of the steam generators.

The status of the CAPRA project, aimed at demonstrating the feasibility of a fast reactor to burn plutonium at as high a rate as possible and the status of the European Fast Reactor are presented as well as their evolution.

Finally the R and D in support of the operation of PHENIX and SUPERPHENIX, in support of the "knowledge-acquisition" programme, and CAPRA and EFR programmes is presented, as well the present status of the stage 2 dismantling of the RAPSODIE experimental fast reactor.

1. GENERAL SITUATION

In 1996, the total amount of electricity produced in France was 488.9 TWh, out of which 378.2 TWh (77 %) were produced by nuclear power plants, 41.8 TWh (8.5 %) by conventional thermal plants, and 68.9 TWh (14 %) by hydraulic plants.

72.6 TWh were exported, 3.5 TWh were imported, 5.9 TWh were used for pumping, while line losses came to 31 TWh. Consequently, the net electrical power consumption in France was 383 TWh, corresponding to an increase of 3.8 % in comparison to the 1995 consumption.

At the end of 1996, "Electricité de France" had 55 PWR units (Thirty four 900 MWe units, twenty 1300 MWe units, and one 1450 MWe unit, CHOOZ B1) in operation. Total capacity for PWR was 59795 MWe (58 340 MWe end of 1995).

The availability factor for these units was maintained at 82.7 % (81 % in 1995) : 82.2 % for the 1300 MWe units and 83.1 % for the 900 MWe units.

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* With contributions from :
P. COULON for the PHENIX paragraph,
D. CLEMENT (NERSA), B. MESNAGE for the SUPERPHENIX paragraph,
S. PILLON (CEA) for the CAPRA paragraph,
G. HUBERT (EDF), JC. LEFEVRE (FRAMATOME-NOVATOME) for the EFR paragraph.

1996 was marked by a slight increase of unexpected shutdown periods (3.2 % in 1996 instead of 3 % in 1995) and a new reduction in programmed shutdown periods (14,1 % instead of 16 % in 1995).

In 1996, a good safety level was maintained : 1.6 event per reactor/year. A level 1 classification was given to 84 events. A level 2 classification was given to 2 events (INES scale).

In the field of Fast Reactors, the main events of 1996 were the following.

At the end of February 1996, the PHENIX reactor was ready for operation. After review meetings, in February 1996, the French Safety Authority did not rule on the possibility of extending the lifetime of PHENIX, and did not want to authorize performance of the 50 th cycle. The Safety Authority has requested safety improvements and technical demonstrations, before it examines the possibility of authorizing a new start-up of PHENIX. The year 1996 was devoted to this work (§ 3).

Concerning the CREYS-MALVILLE plant (SUPERPHENIX) (§ 4) the reactor was restarted at the end of 1995 after a number of minor incidents. The reactor power was increased by successive steps : 30 % Pn up to February 6, followed by 50 % Pn up to May, then 60 % up to October and 90 % Pn during the last months. A programmed shutdown period occurred during May, June and mid-July 1996.

The reactor has been shutdown at the end of 1996 for the 10 year visit of the steam generators, during which the radial blanket should partially be removed, and experimental assemblies should be loaded, as a part of the Knowledge Acquisition Programme (KAP), in French "PAC" (§ 5).

A special panel, the "CASTAING Commission" was appointed by the French government to evaluate the relevance of the "PAC". The Commission delivered its report in June 1996, recommending the pursuit and extension of this programme.

The CAPRA project, initiated in February 1993, aimed at demonstrating the feasibility of a fast reactor to burn plutonium at as high a rate as possible. The first two-year phase of the CAPRA project studies (1993-1994) was completed. Complementary studies, now in the framework of the CAPRA programme, were performed in 1996 and will be presented in § 6.

The European Fast Reactor was launched in 1988 and has reached an important stage with the completion of the Concept Validation Phase in 1993. The status of the EFR programme will be presented in § 7 as well as its evolution.

The four main objectives can be summarized as follows :

- Good operation of PHENIX (§ 3) and SUPERPHENIX (§ 4).
- The "knowledge acquisition" programme (§ 5).
- The CAPRA programme (§ 6).
- The EFR programme (§ 7).

The R and D in support of these objectives is presented in § 8.

This R and D must also be placed within the context of the international collaboration Specific agreements exist between Europe and Japan, between France and Russia (CEA-MINATOM agreement in which Germany and the United Kingdom are "Associated Partners"), between France and China, and between CEA and General Electric.

For information the present status of the stage 2 dismantling of the RAPSODIE experimental fast reactor is presented in the next paragraph (§ 2).

2. STAGE 2 DISMANTLING OF THE RAPSODIE EXPERIMENTAL FAST REACTOR

The RAPSODIE experimental loop type sodium cooled fast reactor went critical for the first time in 1967. This reactor located at CADARACHE was operated by the CEA. The nominal power was increased from 24 MWth to 40 MWth in 1970 after modifications of the core and the systems.

Up to the discovery in October 1978 of a very small sodium leak estimated at 10 g/year across the primary vessel, RAPSODIE did not experience any safety problems and provided considerable fuel data (maximum BU 26 %). Later in 1982 a larger nitrogen leak appeared across the double wall of the reactor (fig. 2.1). The area of this leak was estimated at 0,5 cm2. Because the sodium double containment was not assured, it was decided to close the reactor. Nevertheless is was possible to carry out a series of end-of-life tests just before closing down the reactor. The reactor was finally closed on April 15th, 1983.

Pre-decommissioning operation took place from 1984 to 1986 with the removal of fuel and blanket assemblies and the drainage of the primary sodium. This sodium was purified from ^{137}Cs (around 1 % were left).

A partial decommissioning (stage 2 in IAEA classification) was decided in 1996. The operation started in 1987 and was completed in 1994.

The dismantling operations were carried out by UDIN (Unité de Démantèlement des Installations Nucléaires) "Nuclear Facilities Dismantling Unit" (CEA unit).

The steel and nickel reflector assemblies (130 000 Curies in 1987) were removed and washed to remove the residual sodium, then were put into containers to be transferred to an interim storage zone. These operations were completed in September 1988.

In the same way, the devices installed in the core were removed, i.e. irradiation and in-pile measuring devices and control rod mechanism. These highly radioactive elements, located in the rotating plugs were replaced with biological shielding plugs, washed and then cut up and packed into 45 litre containers. These operations were completed at the beginning of 1990.

After cutting the pipes, the primary vessel was isolated. Afterwards the reactor block was sealed (fig. 2.1). This work was achieved in 1991. At the present time the primary vessel contains approximatively 170 kg of primary sodium on the form of oxide deposits and "puddles" of sodium trapped. First and second barriers are maintained in nitrogen atmosphere at an overpressure of a few millibars.

The systems isolated from the reactor vessel were washed and decontaminated with a participation of the European Commission.

The sodium purification system, made up of narrow pipes, could not be totally drained. It was completely cut up into small sections which were washed in a special cell in the hot building connected to RAPSODIE buildings.

All these operations took place over 1988, 1989 and 1990.

Afterwards these systems (primary loops, Na-Na heat exchangers, pumps, Na-air heat exchangers, auxiliary systems, ...) were dismantled. The stainless-steel debris were melted (70 tons).

The 37 tons of primary sodium were destroyed in the DESORA facility (40 kg/h during continuous operation). This operation took place from November 93 to march 1994. 170 m^3 of 10 N soda were produced and sent to "La Hague" where they were used for effluent processing.

Today the reactor block is sealed and the cells of the containment building are empty (fig. 2.2).

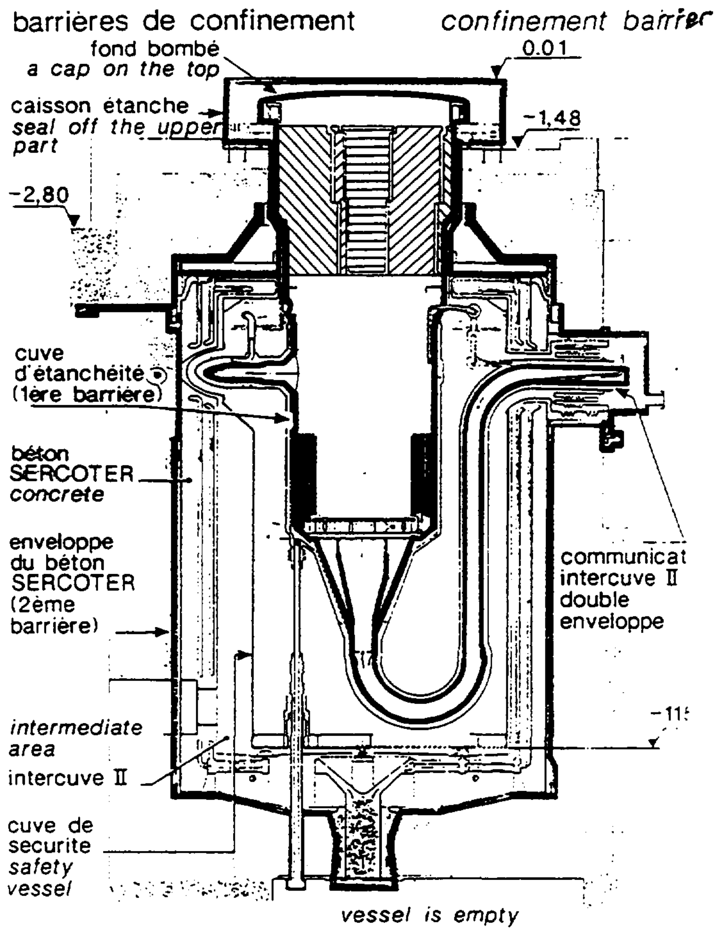


Fig. 2.1. - Reactor block barriers

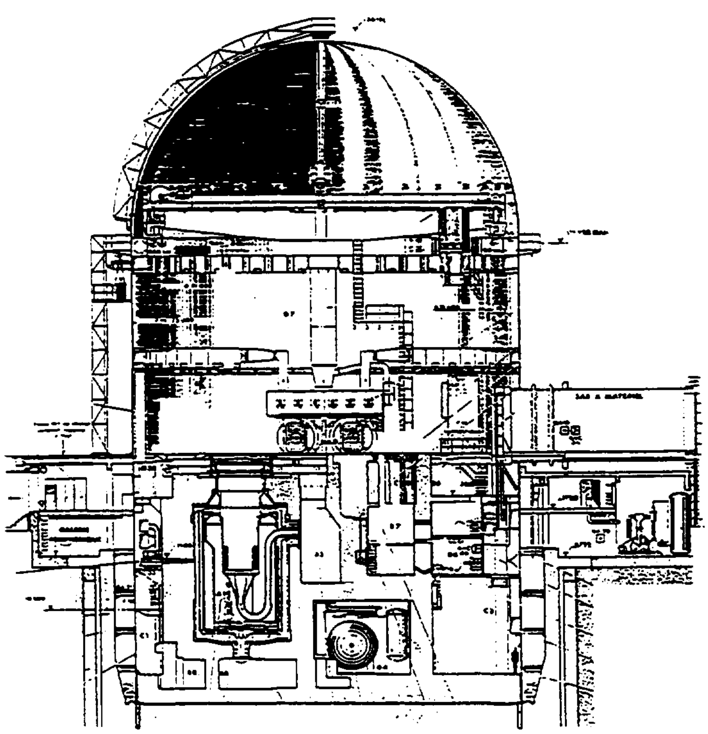


Fig. 2.2. - RAPSODIE containment after dismantling

The operation to be carried out are :

- To destroy 20 m3 of secondary sodium contained in two tanks.
- To clean and destroy two cold traps.
- To destroy the two cesium traps.
- To clean and destroy the DESORA facility.
- To clean and destroy washing pits.

and to achieve an important documentation work.

The complete dismantling (stage 3 in IAEA classification) has not yet been decided. However the containment and monitoring means implemented should allowed a safe waiting period of several decalles.

3. SITUATION OF PHENIX

For the reasons explained in the report presented during the 28 th IWGFR meeting [Ref. 1], major decisions were taken in 1994 in order to extend the life time of the reactor another 10 years. A work programme was decided which involves long shutdown periods. Such a period occurred after the 49 th cycle. This cycle was completed on April 1995 without any incident.

A the end of 1995, the secondary loops SL2 and SL3 were available. Finally, the plant was ready for operation at the end of February 1996.

After technical review meetings, in February 1996, the Safety Authority did not rule on the possibility of extending the life time of PHENIX, and did not want to authorize performance of the 50 th cycle.

The Safety Authority has requested safety improvements and technical demonstrations, before it examines the possibility of authorizing a new start-up of PHENIX, in particular :

- The installation of a Safety Scram System (Complementary Shutdown System).

- A demonstration of the possibility of reinforcing the plant against seismic loads (historically maximum seism) taking into account new design rules.

- An improvement of the emergency cooling system in case of earthquake.

- Consideration of the foreseeable soundness of the core support line.

The first three points were already included in the PHENIX life extension programme and their status is presented below (respectively batches 7, 8 and 9, 10).

The studies on core support (4 th point above) includes :

- The review of manufacturing quality of the core support.

- Defects assessment in the conical shell (figure 3.1).

- Detailed description of thermohydraulical of cold collector (with TRIO-VF, see § 8.4.3) and core support thermomechanics.

- LBB arguments.

• Evaluation of damages other than thermomechanical damages (i.e. : irradiation damages, ageing of materials, possible corrosion, ...).

The life extension programme was divided in batches. The status of these batches is presented below :

Batch n°1 : The order of three extra IHXs ; they will be delivered in november 1997.

Batch n°2 : The replacement of the 321 - stainless - steel pipes in the secondary circuits.

At the time of construction two types of steel were used for the main secondary sodium pipes :

• Ordinary stainless steel (steel grade 304) for all the lines in the reactor building and a large part of the lines in the steam generator building.

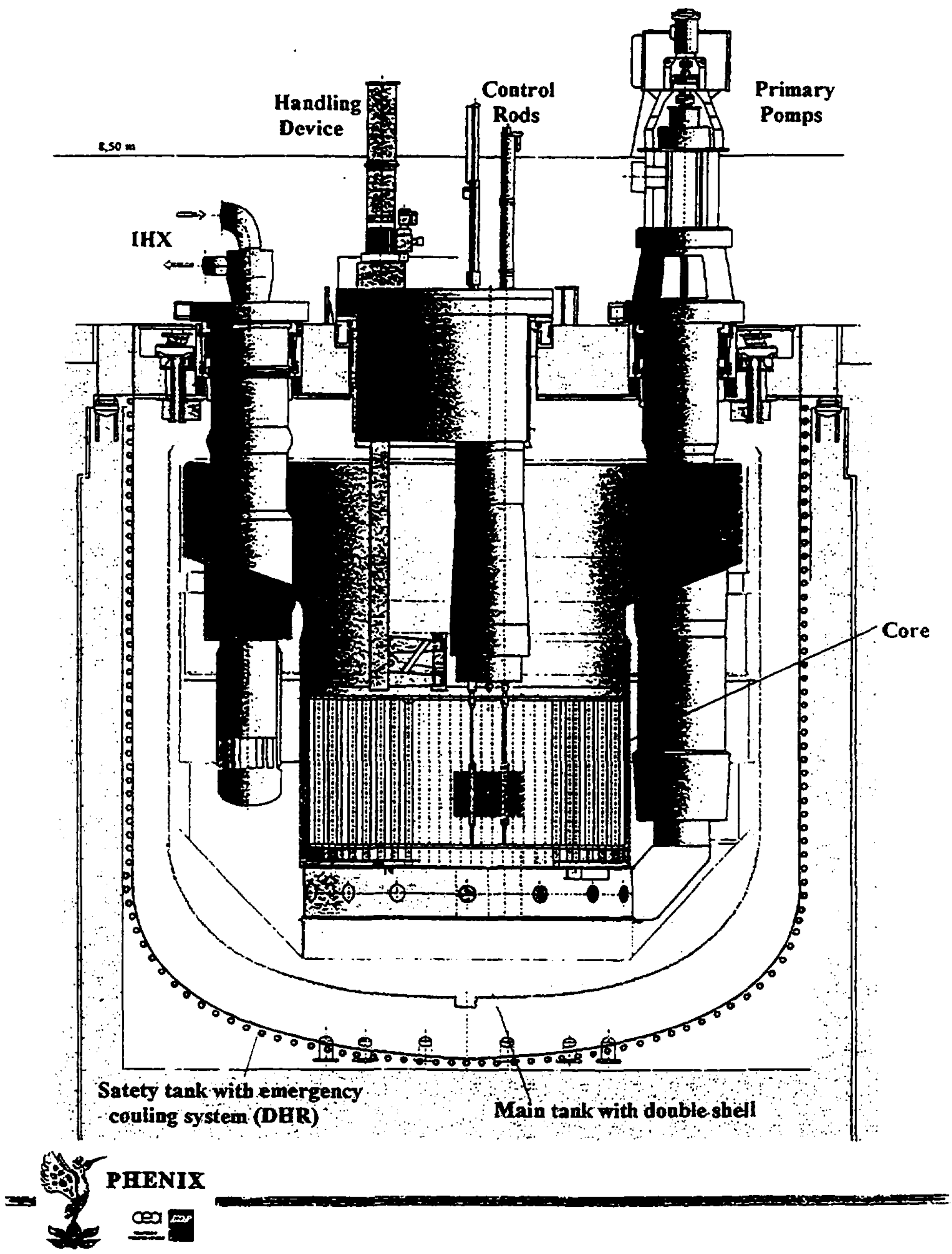
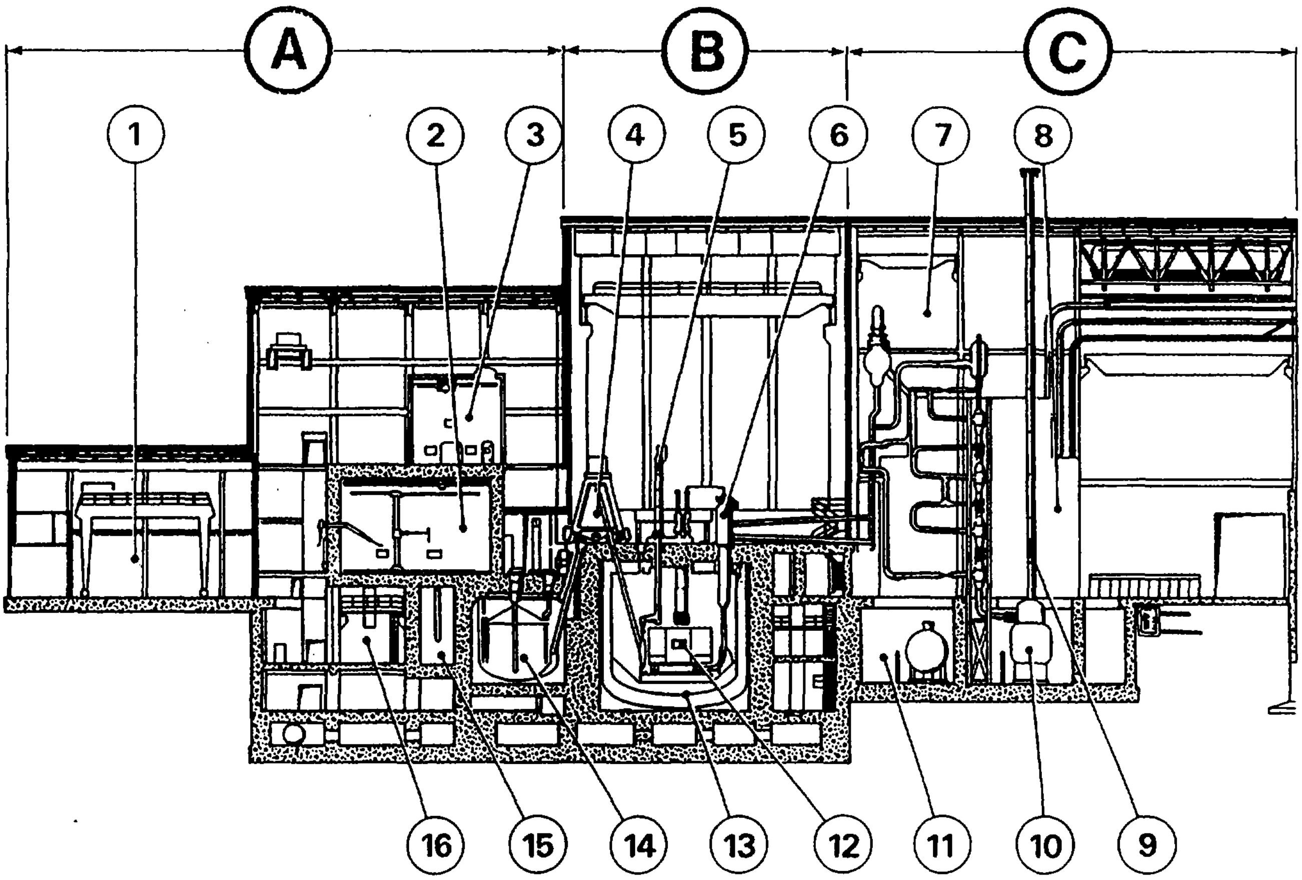


FIG. 3.1. - PHENIX - Primary circuit

LONGITUDINAL CROSS SECTION OF PHENIX



(A) Handling area

(1) Stores
(2) Spent fuel pit
(3) Upper pit
(4) Handling lock
(5) Transfer arm
(6) Primary sodium pump

(B) Reactor building

(7) Secondary sodium pump
(8) Steam generator
(9) Unloading device
(10) H/Na separator
(11) Secondary sodium storage
(12) Core

(C) Generator building

(13) Reactor block
(14) External storage drum
(15) Spent fuel washing area
(16) Lead casket discharge bay

Fig. 3.2. - Longitudinal cross section of PHENIX

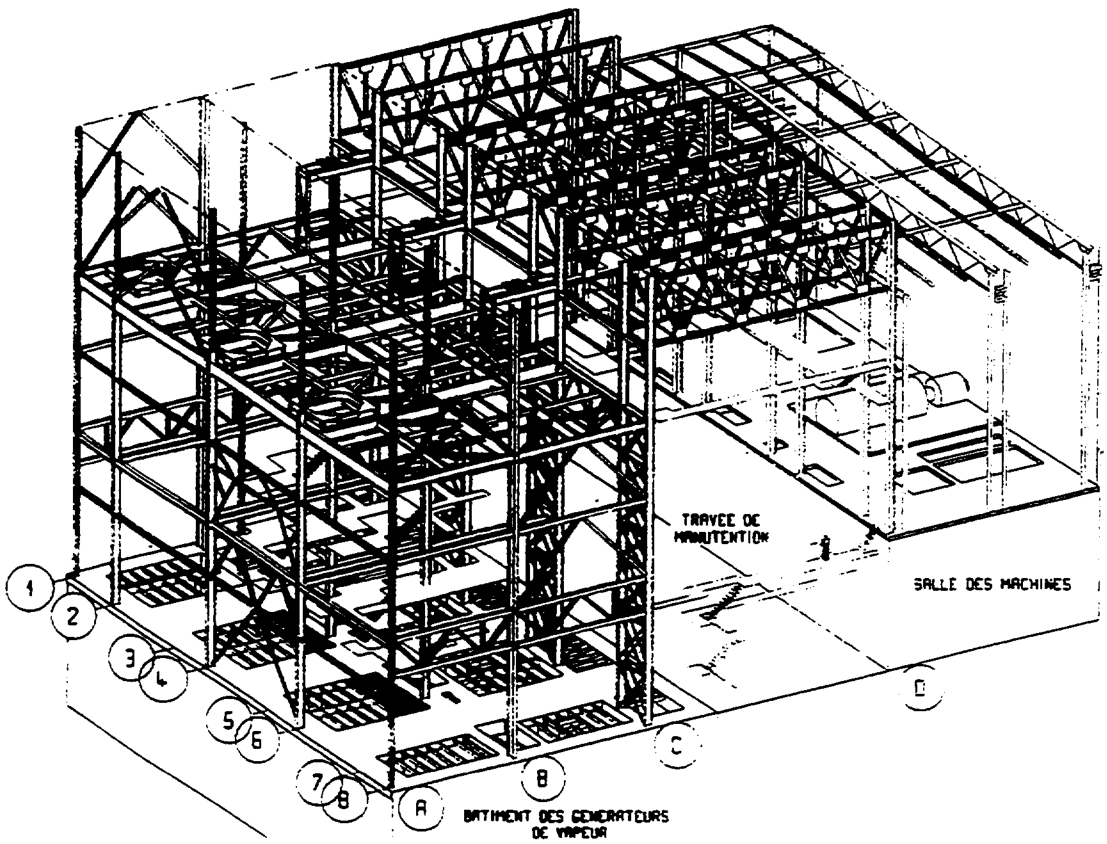
- Stainless steel with added titanium (steel grade 321) for a proportion of the lines in the steam generator building, which was intended to enhance the mechanical performance of the steel at high temperature.

Intensive programmes of non destructive tests since 1989 have shown that the 321 stainless steel has not performed satisfactorily over the time, with fissures in welded joints kept under load and high temperature.

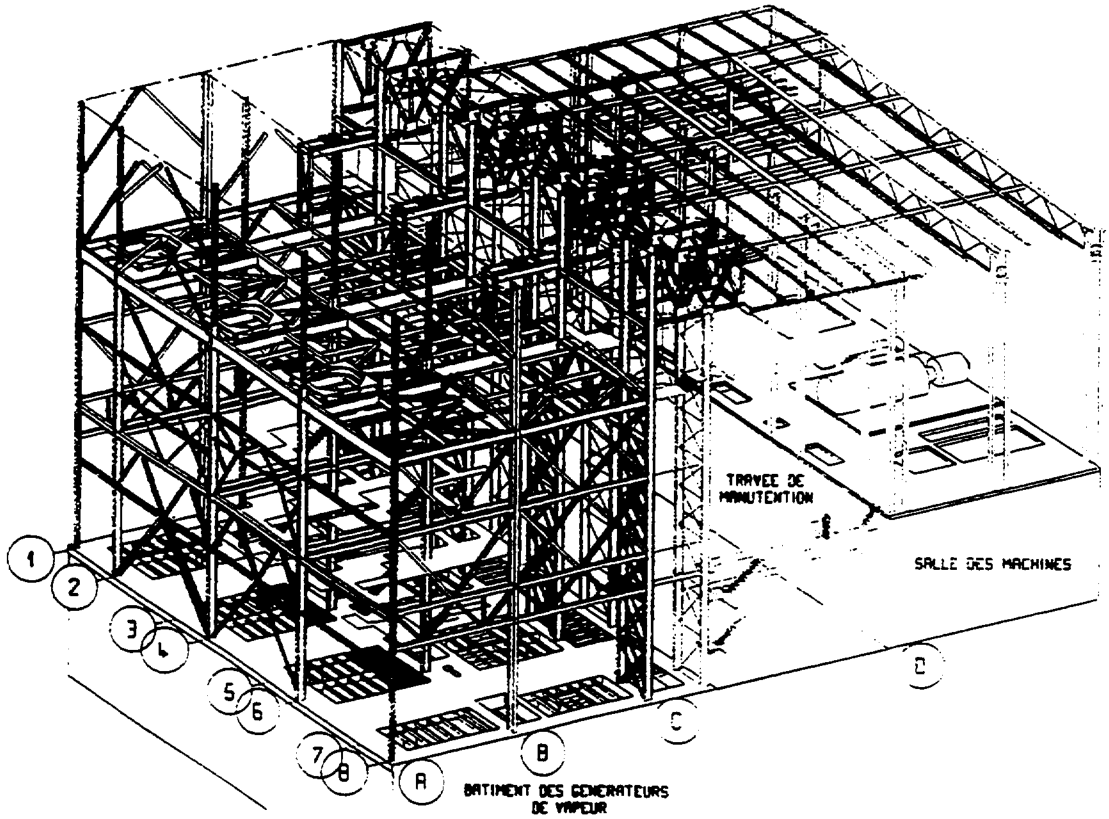
The sections of pipework made in this steel are being replaced by 316 SPH steel. The secondary circuits numbers 2 and 3 have already been completed.

Comprehensive checks are being made at the same time on the welds in stainless steel 304 pipes (bends, tees and connections).

Batch n°3 : Sodium fire protection in the steam generator building.



Before



After completion of the work

Fig. 3.3. - Separation of steam generator building and turbo generator building and turbo generator building

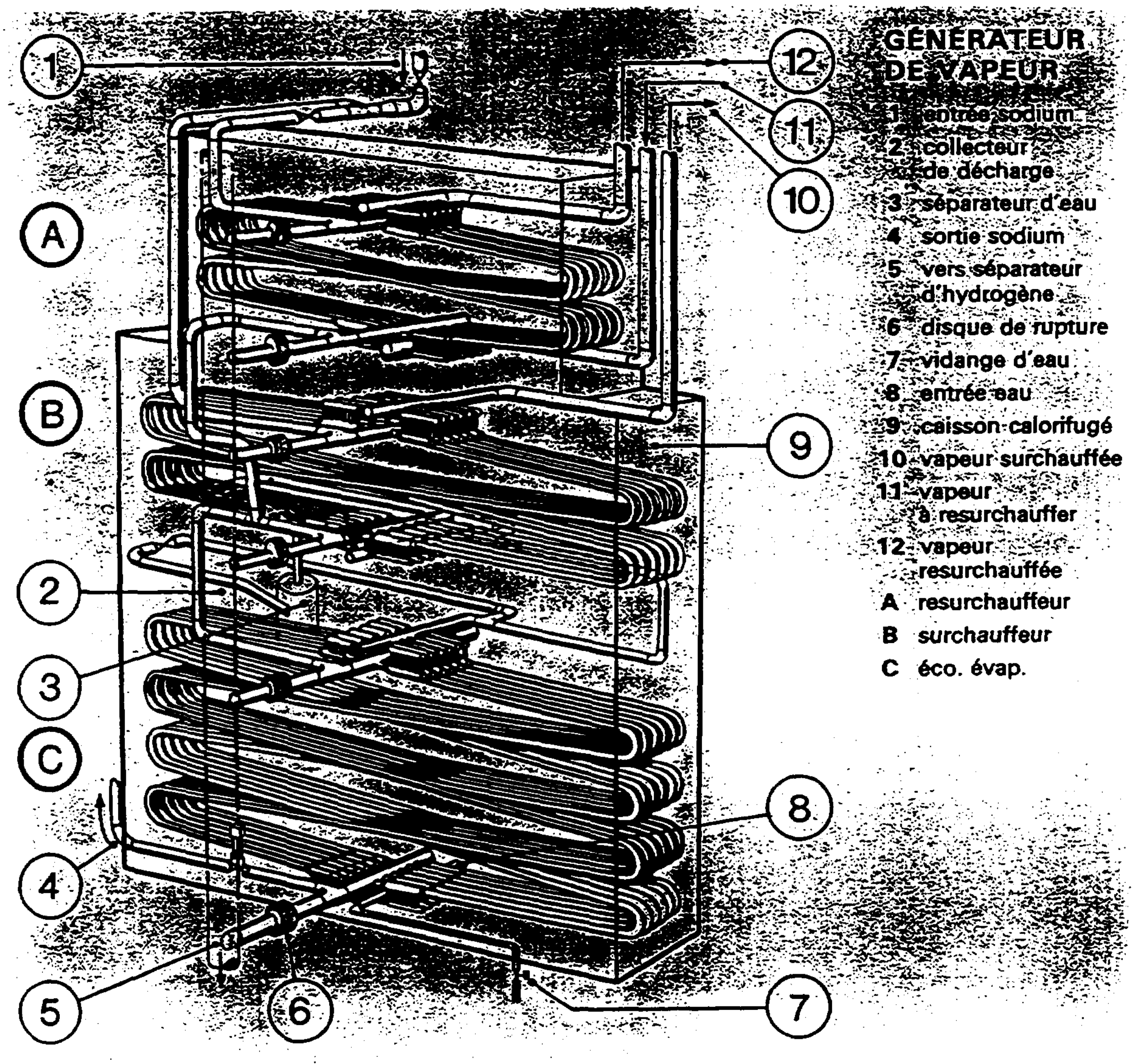


Fig. 3.4. - PHENIX steam generator

The steam generators are located in the same building and this could be a common mode of failure in case of a sodium fire in one of the three circuits.

In order to improve the protection against sodium fire the following works are in progress and should be archived in 1998 : partial isolation of each loop from the others, isolation of the sodium from the steam circuits, thermal protection of some structures (particularly the supports of the sodium pipes, the main parts of the structural steel network, the steam generator vessels), the construction of sodium recuperation tanks in case of leaks, the construction of a protection gallery for the command equipment, modification of the ventilation circuits, etc... .

Some of these works will be completed before the 50 th operating cycle.

Also the detection systems were improved (video camera, ...).

<u>Batch n°4</u> : Renovation of the auxiliary sodium circuits.

The intention is to eliminate any auxiliary sodium circuits not really needed. This simplification will be achieved in 1997.

Batch n°5 : Evaluation of the risks of cracking and thermal fatigue. This study was achieved in 1996.

Batch n°6 : Tests related to helium injection circuit in the space between the vessels (fig. 3.1) in order to improve heat transfer in case of loss of heat sink. This possibility will be used according the nominal power to be decided.

Batch n°7 : Safety Scram System (Complementary Shutdown System).

This system was installed and tested in the reactor respectively at the end of 96 and beginning of 97. This system and its control mechanism are directly derived from those of SUPERPHENIX.

Batch n°8 : Reinforcement of the against earthquake.

Batch n°8.1 : Reinforcement of the steam generator building. This work will be completed in 1998.

Batch n°8.2 : Reinforcement of the annexes building. This work will be finished before starting the 50 th cycle.

Batch n°8.3 : Reinforcement for the turbo-generator building.

The turbo generator and steam generator buildings, which have part of their structures in common (fig. 3.2), will be modified so as to separate them entirely and give them independent structures. To do this it has been decided to split the steam generator hall in two by the construction of an offset portal attached to the steam generator hall (fig. 3.3). Removal of the water tanks from the steam generator building into the turbo generator building is presently being studied and will be made in 1997.

Batch n°9 : Reinforcement of handling building against earthquake. To be achieved in 1998.

Batch n°10 : Emergency cooling system.

It is postulated that the present closed circuits which should evacuate the DHR in the event of an accident involving loss of normal DHR circuits might no longer be operational after an earthquake.

This is due to the risk of loss of the external water feed which provides for heat exchange. The circuits would then function in open mode bringing external water directly into the final backup circuits.

The work lot 10.1 involves replacing the old open circuits by new pipework guaranteed against earthquake and which would be accessible after en earthquake and could be fed independently by water from the river Rhône.

After the 50th cycle, new closed circuits equipped with air/water exchangers should guarantee the evacuation of DHR, even after an earthquake.

Batch n°11 : Primary pump examination (and repair if necessary).

Batch n°12 : Improvement of hydrogen detection in the steam generators.

Batch n°13 : Set-up in the hot plenum of an instrumentation mast (thermocouple).

Batch n°14 : Reinforcement and protection of the water-steam pipes.

Batch n°15 : Steam generator collectors (fig. 3.4).

Defects were detected during non destructive testing of the steam generators circuits. Because repair would have been difficult it was decided to replace them by collectors in 321 steel.

Restarting of the PHENIX plant.

Before starting the 50 th cycle, the works referred as batches : 2, 3, 4, 8.2, 8.3, 10.1 and 15 should be achieved together with the core support line studies.

Technical review meetings are planned in June 1997 before the Safety Authority decision regarding the performance of the 50 th cycle.
This cycle will be carried out with two secondary circuits at 2/3 of nominal power (loop SL2, SL3).

4. OPERATION OF SUPERPHENIX

D. CLEMENT (NERSA), B. MESNAGE (CEA).

4.1. General aspect of 1996

In 1966, Superphenix was characterized by excellent operation throughout the year, as illustrated by the diagram of electricity production in Figure 4.1. The operational programme was conducted in accordance with the constraints of the core as well as with the authorizations delivered by the Safety Authority. These, in turn, were linked to the tests and data acquisition concerning the behaviour of the equipment at various operating steady states: 30 %, then 50 and 60 %, and up to 90 % of the nominal power (NP) during the last few months of the year.

Over the totality of the year, the plant was coupled to the grid for 246 days (nearly 70 % of the time) and supplied 3.4 billion kwh. The availability of the reactor, apart from scheduled outage, was 95 %, fuel burnup reaching 318.8 equivalent nominal power days by the end of the year.

No notable technical incident troubled the operation of the plant in 1996. Five unscheduled shutdowns, four of which resulted in the automatic activation of the protection systems with motorized control rod insertion (rapid shutdown), marked the operating periods of the reactor. These shutdowns, that were followed by a rapid restart, lasted only a total of about twenty days.

The steady operation of the facility enabled the operator to acquire further knowledge of the behaviour of the plant, to complete the requalification of the work performed since 1993 to fight sodium fires, and to achieve the acquisitions as scheduled in the KAP (Knowledge Acquisition Programme) (§ 5) assigned to Superphenix. Several experiments in the framework of the KAP were performed at the end of 1996.

The main scheduled outage took place during the second quarter of 1996 and was devoted to the replacement of 21 control rods and to annual maintenance operations. Performed as planned, it was followed by a series of neutron tests, then by a period of waiting for a restart authorization from the Safety Authority.

The operation of the plant was stopped the 24th December for a second scheduled outage. This shutdown period will cover the first six months of 1997 in order to carry out decenial controls such as steam-generator hydrotest, controls related to safety and rearrangement of the core pattern.

Two series of tests, focused on the entire handling chain of both new and irradiated assemblies up to the APEC, also enabled confirming the good operation of the equipment during the year and validating the organization to be provided with a view to the handling campaign scheduled in 1997, for example with the loading of the first three experimental KAP fuel subassemblies (§ 4.4).

Another outstanding event in 1996 was the issue in June of a report by the commission chaired by Professor Castaing. In his findings, the commission confirmed the scientific usefulness of Superphenix and its suitability for conducting the research planned in the knowledge acquisition programme. It underlined in particular that the knowledge required with respect to the reduction of long-lived radioactive wastes cannot be "obtained more simply or at lower cost in other existing facilities." This opinion was shared by the Government, which, in a communiqué, stressed the scientific interest of more general scope presented by Superphenix as a vast source of fast neutrons, notably as regards the research required in the framework of the law of December 30, 1991, relative to long-lived radioactive waste management.

Finally, in spite of the decrease in the number of visitors following the reinforcement of the safety measures taken by the Government (the anti-terrorist Vigipirate plan), Creys-Malville maintained its efforts with respect to information and dialogue throughout the year, and, in particular, continued to distribute its weekly news bulletin concerning the operation of the facility. In the autumn, the plant received a Japanese engineer on detachment for a year in the framework of a cooperation agreement signed in October between the NERSA and the PNC (Japan).

4.2. Plant operation

Following the coupling of the turbo-generator sets on December 30 and 31, 1995, the operation of the plant at 30 % of the nominal power lasted throughout the month of January. This operation permitted, on the one hand, profitable monitoring of the chemistry of the primary sodium via the behaviour of the cold traps and the plugging meters, for which a cleaning procedure was successfully tested, and, on the other hand, the detection of the impeded displacement of certain sodium circuit piping, which resulted in various actions whose appropriateness was later confirmed during monitoring of piping displacement at higher power.

A failure of a conductivity meter on the cooling water circuit of an alternator led to the shutdown of the corresponding turbo-generator set on January 7, thus validating the procedure involving switching the load from one generator set to another.

The power rise to 50 % nominal power was initiated on February 2, as soon as the authorization from the Safety Authority was delivered. When attained a few days later, this steady state was maintained up until May 3, at which time, the core having operated for about 240 equivalent nominal power days, it became necessary to replace the control rods that had reached the end of their guaranteed service life.

This steady state, during which the numerous tests and data acquisition required to further knowledge of the behaviour of the plant were performed, in particular permitted the verification of the good operation of each turbo-generator set at its nominal power. In addition, one or other of the generators were shut down several times to allow various actions to be performed on the equipment of the electricity production facilities, confirming the operator's full control of piloting operations with respect to switching from one set to another, without perturbing the reactor.

Interrupted on April 23 for a few days by a fast shutdown that occurred as a result of faulty water flow control following the activation of a turbine-driven feedwater pump, this period of steady state operation ended, running solely on the B train turbo-generator. The A train transformer was shut down as the analysis made periodically of the cooling oil of the transformer had revealed that a preventive internal inspection was required.

This 50 % nominal power steady state was also marked by the appearance of a slight argon leak on the repair sleeve mounted in 1995 [Ref. 2]on the intermediate heat exchanger N°2 of the E secondary circuit. The leak had no effect on its operation, which stabilized during the year to around 2.5 NTP l/h, thus confirming the reliability of the repair.

After the annual maintenance outage and the neutron test campaign at 180, 250, and 345° C, whose purpose was to check the conformity of the new rods and determine their characteristics with respect to reactivity, the plant was restarted around mid-July, as soon as restart authorization was obtained. This restart was performed with both turbo-generator sets, as the results of the expert assessment of the transformer that had previously been shutdown were satisfactory.

Following a rise to power perturbed by a rapid shutdown triggered by an electrical disturbance of the protection systems of a steam generator, due to lightning striking in the vicinity of the control building during a violent storm, a 60 % nominal power steady state was reached on July 28 and continued up until October 15, the date on which the authorization to increase power to 90 % NP was obtained from the Safety Authority.

Various tests and data acquisition operations planned during the operation programme were continued during the 60 % NP steady state. Power was temporarily reduced several times: to 50 % NP in order to conduct the last adjustments to the equipment of the electricity production facilities and to avoid automatic shutdown of the plant in the event of a spurious trip of a turbine. On another occasion, it was reduced to 40 % NP to enable repairing a steam inlet piping support, with the corresponding turbine shut down.

This steady state was also interrupted for a few days by a rapid shutdown that occurred on August 6, following a trip of a secondary pump due to the failure of an electric module of the pump speed regulation system.

The 90 % NP steady state was soon attained, as the parameters involved change little between 60 and 90 % NP, and started on October 23. It was volontarily interrupted a first time on November 6, following the detection of unacceptable vibration of a bearing of the D primary pump shaft (see Figure 2). The replacement of the bearing housing and the seal mount, well controlled by the operator, was performed in less than 10 days. Expert assessment of the bearing revealed a rupture in three parts of one of the 37 bearing rollers; this rupture could be attributed to a fault of metallurgical origin.

The 90 % NP steady state was again interrupted on November 25 for one week to allow completion of the tests planned in the framework of the knoledge acquisition programme, which included inserting neutron flux detectors into two intermediate heat exchangers (§ 4.4). This shutdown, which was planned specifically for the KAP, occurred, however, a few days in advance of the scheduled date, following a fast shutdown due to an inappropriate handling action by an operator during the changing of a filter on the oil seal circuit of an alternator.

The first December, during the rise to power a rapid shutdown was triggered by in increase of hydrogen in the sodium of the secondary loops. The cause was the start-up of cold traps where the sodium was frozen. Due to thermal gradient and differential dilatations in the iron mesh, hydrides were released in the sodium.

The 90 % NP steady state finally ended with a 13-day irradiation period which permitted irradiating the flux detectors to their saturation point, as well as completing the plant restart programme. On the one hand, the final requalification of the modifications made to the facility to improve protection against sodium fires was performed, with respect to which the good behaviour of that the new, so-called "sandwich", Na leak detection systems should be noted. On the other hand, two specific tests were also successfully performed: first of all, with the reactor at power, calibrated tests of hydrogen injection into the secondary circuits to study the response of the leak detection systems of the steam generators; then, using the shutdown on December 24, a SG rapid decompression isolation test starting at 80 % NP to study steam generator behaviour during the transient.

The shutdown of the reactor was followed by the very fast recovery of the neutron flux detectors which were sent to the CEA at Cadarache for activity counting, then by the characterization at 180° C of the reactive state of the core before the start of planned maintenance operations and the associated handling campaign.

4.3. Assembly maintenance and handling

The annual maintenance outage that took place from May 3 to June 16 was devoted to replacing the 21 control rods of the main shutdown system by new rods in parallel with the rearrangement of the assemblies in storage on the core periphery, a handling operation that was performed at the beginning of June, and, with respect to the facility as a whole, to the numerous periodic maintenance actions and tests that are mandatory in the framework of the general operating rules, in particular:

- The overhaul of the motor-generator sets,
- The checking and replacement of the electric batteries,
- The replacement of self-locking devices on two secondary circuits, in parallel with the maintenance of the electric pump motors,
- The inspection of the couplings of the drive motors of the primary pumps,
- The visual inspection of the annular space of the plugs, which confirmed the absence of sodium aerosol deposits,
- The draining of the four decay heat removal circuits for maintenance and inspection of the sodium-air exchanger,
- The internal inspection of the main transformer of the A train after draining of its coolant oil (80 tonnes), which revealed only a minor fault on the magnetic shunts that did not in any way jeopardize its operation.

Decided on December 24, the scheduled 1997 outage will be devoted to the rearrangement of the core pattern and to the first campaign of assembly removal planned in the framework of the operating programme of the plant and the KAP, as well as to numerous maintenance and checking operations. The main operations will concern mandatory inspections of the steam generators and the replacement of the cold traps on the four secondary circuits, as well as of the two integrated purification cartridges of the primary sodium.

In anticipation of this handling campaign, a vast programme of team training and streamlining of the entire handling system of the assemblies from the reactor block to the APEC (spent fuel storage tank) was carried out throughout the year.

In addition to the handling operations performed in the framework of the annual maintenance outage, but which required only moving assemblies within the primary vessel, the new rods being already present at the core periphery, two overall test campaigns were performed in March and October and will be completed at the beginning of 1997 by the checking of the biological shielding using a 666 Tbq (18 000 curie) radioactive source. The new assembly system saw the acceptance, the checking, and the storing of the three experimental KAP subassemblies, as well as the fuel and steel assemblies. An assembly mockup with a ferritic steel hexagonal tube was also immersed in the APEC pool to check its behaviour under water.

These various actions thus enabled completing the operational startup of the equipment of the handling system and acquiring the data required for the future organization of irradiated assembly unloading.

4.4. The Knowledge Acquisition Programme (KAP)

In 1996, progress was made in the three fields covered by the PAC: the demonstration of prototype FBR operation, research into plutonium consumption, and that concerning the destruction of long-lived wastes.

The satisfactory performance of the general requalification programme of the plant throughout the year permitted collecting a maximum of data on the operation of the reactor, the monitoring systems, and the equipment of the water-steam circuits while the power level (90 % NP) reached at the end of the year permitted the performance of three specific tests.

One of these tests consisted in inserting control rods at 50 and 80 % NP to qualify the prototype system for processing the core outlet temperature measurements (ALPES system, see also § 8.3.1.1), while the others were measurements intended to reduce the uncertainty margins of the calculation tools :

- Measurement of the heating of the primary circuit due to the decay heat of the core after a rapid shutdown of the reactor,
- Measurement, at the level of the two intermediate exchangers, of the neutron flux released from the core using activation detectors inserted into the thimbles (Figure 3).

Two events concern the implementation of the PAC:

- The production from the Cogema-Framatome fabrication lines and delivery in November of the three experimental assemblies (ref. 2), whose irradiation started in 1997: two CAPRA subassemblies which are high plutonium consumers and one NACRE subassembly containing 2 kg of neptunium for incineration in homogeneous mode,
- The delivery of 72 steel subassemblies (ref. 2) intended to replace the first ring of fertile subassemblies with a view to the first stage of Superphenix's conversion to a burner reactor, scheduled in 1997.

5. THE KNOWLEDGE ACQUISITION PROGRAMME

A detailed information of this programme was given during the two last IWGFR meetings (ref. 1, ref. 2).

In 1996 progress was made in the three fields covered by the knowledge Acquisition Programme : the demonstration of prototype FBR operation, research into plutonium consumption, and that concerning the destruction of long lived wastes.

The progress of these tasks was already presented in § 4.4.

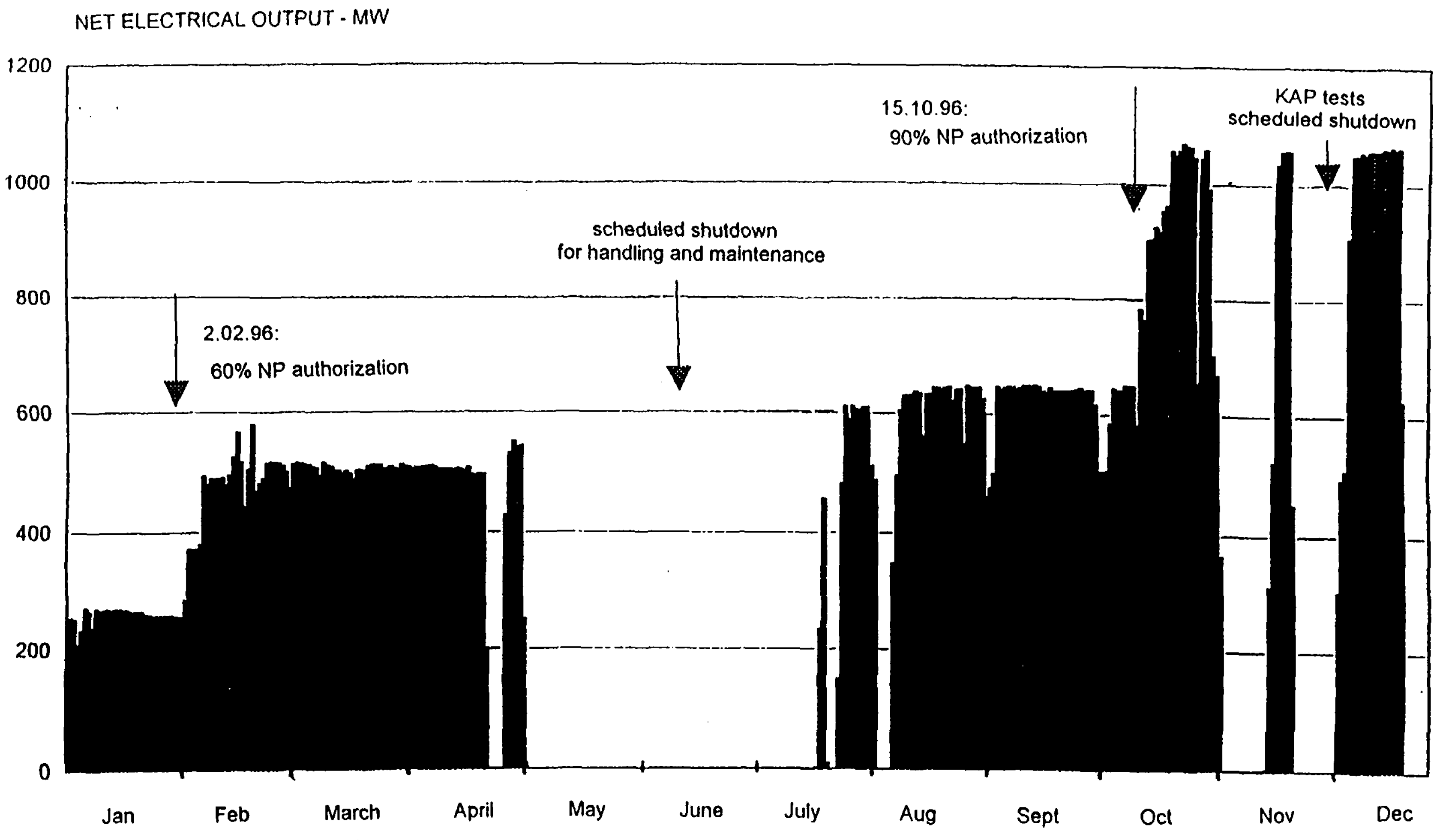

NET ELECTRICAL OUTPUT - MW
1200
1000
800
600
400
200
0
15.10.96:
90% NP authorization
KAP tests
scheduled shutdown
scheduled shutdown
for handling and maintenance
2.02.96:
60% NP authorization
Jan
Feb
March
April
May
June
July
Aug
Sept
Oct
Nov
Dec

Fig. 4.1.

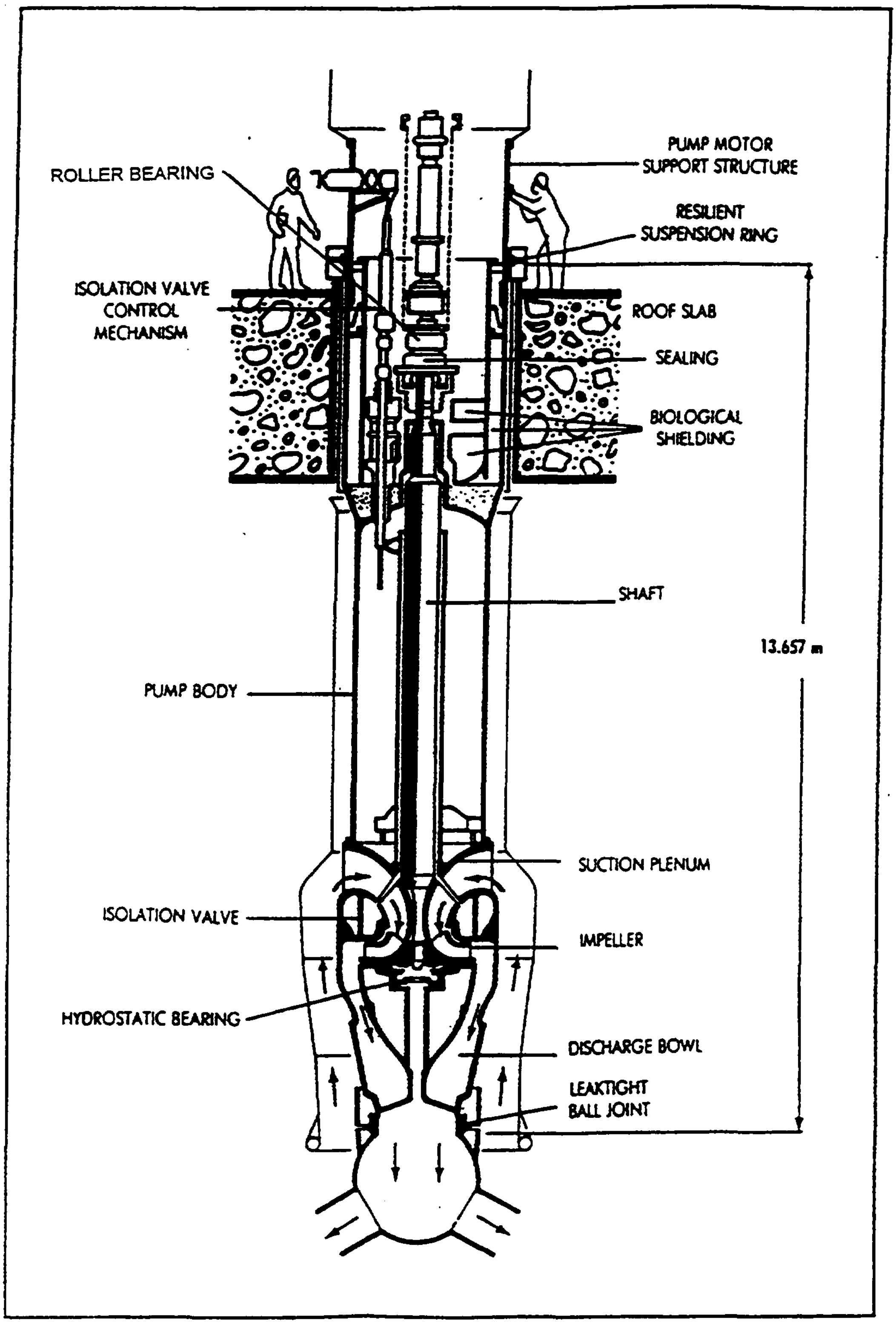
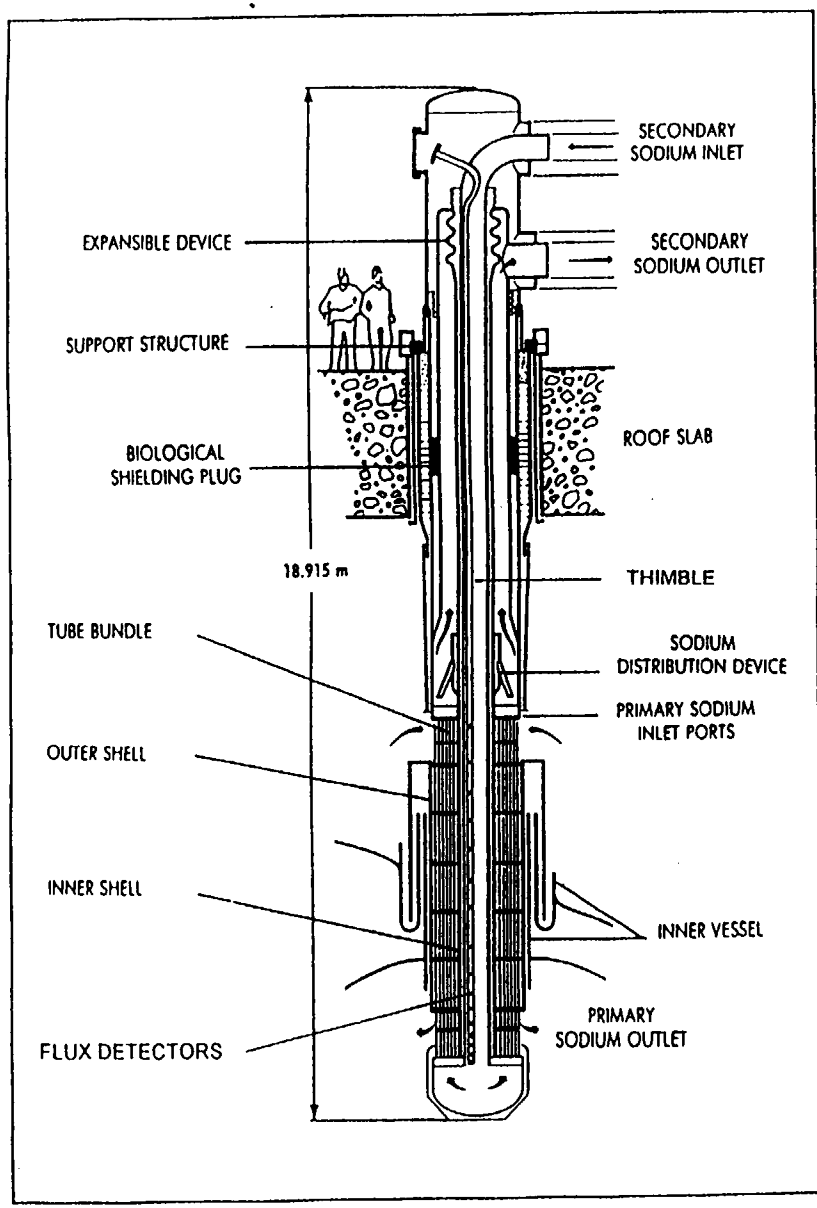


Fig. 4.2. SUPERPHENIX primary pump



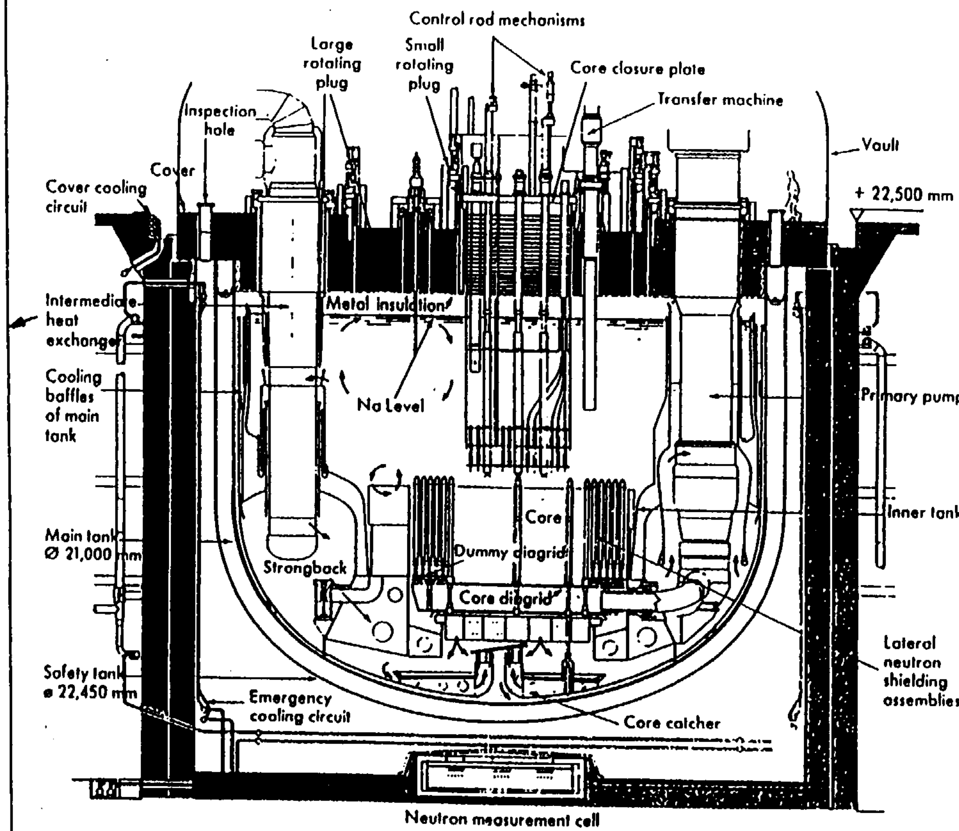


Fig. 4.3. Flux measurement in SUPERPHENIX intermediate exchanger

6. THE CAPRA PROGRAMME
S. PILLON (CEA)

6.1. Introduction

This paragraph summarises the work which was performed in 1996 in the frame of the European CAPRA project. Then, it gathers all the contributions of each partners : BNFL, NNC, AEA-T, FZK, NOVATOME and CEA.

First, it would be interesting to mention that 1996 was a transition period from the CAPRA project to the CAPRA programme. This transition is indicative of the end of a demonstration phase and the beginning of a validation one. It suggests also using the innovative trends in a more generic field for Fast Reactors like safety features or physics methods.

The demonstration phase had consisted in proving the feasibility of an oxide core able to burn plutonium and minor actinides. A design has been proposed and optimized : it is the "4/94m". In the same time, a preliminary feasibility study of a more innovative core based on a plutonium without uranium fuel has also been investigated. It was demonstrated that no serious difficulty or impossibility was attempled, especially in the frame of the safety features, provided that this innovative core should be dedicated to burning of poor plutonium quality grades.

The validation phase will consist in optimising the oxide core thanks to a large experimental feedback on fuel element behaviour under irradiation (thermics, thermomecanics, thermodynamics...), physics and neutronics (new adjusted formulaire, code optimization,...) and safety field. As regards the innovative core, it will consist in proposing a core design at the same level of feasability than the reference one.

6.2. Oxide core studies

6.2.1 High burnup core variant optimisation

The first objective to optimize the oxide core was to increase the fuel residence time which is half of the EFR's one. This a real possibility taking into account the reduced value of the materials peak damage dose. So the "high burnup core" variant has been studied and optimized as regards thermohydraulic behaviour and scatter-batch refuelling scheme. It is characterized by an increase of the fuelled pins number (387 instead of 336) to the detriment of unfuelled pins (82 instead of 133) and by the introduction of a moderator ($^{11}B_4C$) in the unfuelled pins. A 6-batch cycle was modelled and has shown the possibility of suppressing the intermediate mid-cycle shutdown for reactivity control via diluent management. Finally the increase of the cycle length is significant (+45%) compared to the reference core "4/94m" (1322 EFPD instead of 855 EFPD for an achieved burnup of 25 h.a.%). However the sodium void worth is rather high and must be reduced in further studies.

6.2.2 Mitigation of core-melt accident consequences

Analyses, which have been performed to assess the recriticality and core-melt energetics potentials for the CAPRA reference oxide and the Pu w/o U core have shown that compared to a conventional core the risk of recriticality is increased because of the much higher Pu-enrichment.

However, the special design of burner cores can be used advantageously to reduce the recriticality risk by exploiting inherent features. So, the available diluent system can be optimised to enhance the fuel discharge from the core under core-melt conditions. By an early fuel relocation, the escalation into a neutronically active whole-core-melt with its associated energetics might thus be prevented.

Transient analysis of core melt accidents show that for conventional cores with e.g. 20% Pu-enrichment, a loss of 20% of fuel already strongly mitigates the energetics potentials and a loss of 40% makes a power burst very unlikely. In the CAPRA core, more fuel must be discharged with the above percentages increasing to 40% and 70% respectively. This numbers reflect conditions of neutronically active whole core pools where extensive compactive motions could take place. In a more realistic assessment, taking into account the heterogeneous structure of the CAPRA core, the diluent system has an important double function. As long as it is largely intact it reduces the mobility of the molten inventory and prevents any significant sloshing motions. After failure, it can serve as an efficient fuel removal path.

The largest potential for fuel discharge under accident conditions offers the diluent S/A system of the core. The CAPRA core contains 52 diluent S/As where 22 of them are partly filled with $^{11}B_4C$ as moderator to improve void and Doppler values. Usually six fuel S/As are grouped around one diluent. The diluents can become activated for fuel removal after local fuel pool formation and melt-through of the S/A walls in the early transition phase. The empty steel pins can be eroded after fuel enters the bundle and the large hydraulic diameter of the bundle becomes available for the fuel flow. These diluents might be optimises for a maximum potentiel of fuel release.

For assessing the potential for fuel removal through the diluent system the SIMMER III code has been applied as the prime calculational tool. Different types of diluents have been investigated starting with the ordinary diluent assembly and gradually changing the steel and sodium fractions within the hexcans. The fuel removal path through the diluent system was investigated for various boundary conditions of the surrounding fuel/steel pools. The results show that under accident conditions with molten, mobile fuel, enough inventory could be discharged from the core within the proper timescale to prevent the formation of a neutronically active whole core pool with a high energetics potential. Results of the exploratory investigations are promising in that an additional medium line of defense might be introduced into CAPRA cores to cope with the recriticality risk. According to the present status of analysis prompt critical conditions cannot be completely excluded, but scenarios with high reactivity ramp rates and energetics are very unlikely.

Experimental information on the transient behaviour of CAPRA pins, diluents and pools with inert materials is urgently needed. The proof that inherent features and special measures/devices will work under the various circumstances of a core-melt accidents needs extensive code calculations. Experiments have to show the basic operability under specific accident conditions.

6.3. Pu without U core

6.3.1 Fuel studies

It has already been assumed that this core should be devoted to poor plutonium isotopics grade in order to recover Doppler reactivity feedback thanks to the Doppler coefficient of Pu240 and to a lesser extent Pu242. The choice of the fuel has not yet been done : it must be compatible with sodium and with PUREX reprocessing (the burned plutonium can hardly exceed 40 to 50% of its initial content so that once-through options are not of concern), have a good thermal conductivity to keep important margins to the melting for the safety features.

Due to the PUREX constraint, nitride appears to be the best candidate in spite of the N^{15} enrichment necessity to avoid C^{14} production. So different nitride compounds are investigated : PuN alone, (Pu, X)N solid solution (with X= Zr, Y or Ce), PuN/XN cercer compounds (X= Ce, Al, Zr, Y, Ti) or PuN/Metal cermet (Metal = steel or V).

A kinetic model, based on the FACSIMILE code, has been developed to study the effects of composition change during reactor scenarios. Vaporisation flux and vapour pressure data have been used in a preliminary study of transport of gaseous species from the external surface of the fuel and from interconnected pores. The results from this initial kinetic study and the review, together with the comparatively higher thermal conductivity indicate that nitride fuels are stable under normal operating conditions. However the behaviour of a nitride fuel pin during severe temperature transients requires further detailed investigation.

6.3.2. Physics calculations

Safety parameters (Doppler and sodium void worth) have been calculated for the PuN-fulled version (annular pin) of the CAPRA core with very poor Pu grade called "Mec plutonium quality" which corresponds to the equilibrium composition reached by plutonium, at constant power generation, when it is recycled twice as MOX in PWR up to 55 GWd/tHM, once as oxide reference fuel in CAPRA reactor and once in the CAPRA PuN core. An interesting overview of the core features for different Pu compositions is now achieved. There is a range of low grade Pu that would lead to significant Doppler and sodium void worth values as high as in the CAPRA oxide core.

6.3.3. Safety studies : CDA and recriticality risk

Exploratory investigations have been performed for uranium-free core. Several compaction and separation scenarios have been analyzed and compared to the behaviour of EFR and reference CAPRA cores.

The neutronic analyses of these disrupted cores show that the reactivity changes from separation processes dominate those from compaction. Both compaction and separation lead to the highest reactivity increases in the U-free core. For this latter core, a criticality problem is identified and can be more pronounced at high temperature if PuN decomposition is assumed. In the case of complete decomposition and Pu-pure configurations, the masses leading to criticality are reduced by a factor of two.

To give an indication of energetics caused by recriticalities, a comparison has been performed between an homogeneous fuel/steel/inert pool and a layered pool of fuel and separated steel/inert. These two configurations represent compaction and separation scenarios resp. The transient analyses have been performed with SIMMER-II code. The results show that all three cores roughly yield the same range of energetics under recriticality conditions. So for the U-free core, the criticality problem seems to be more of a concern than the energetics one.

However, the critical problem in U-free core might be mitigated if it should be proven that the special pin design with the large central hole leads to an incoherent fuel dispersal (squirting fuel) and that fuel discharge over long distance from the core in the early stages of the accident.

More detailed investigations are necessary, also performing mechanistic analyses starting from the initiation of the accident and following the whole melt progression up to final neutronic shutdown in the core. The SIMMER-III code will be used when the implementation of an advanced neutronics module is finished.

6.4. R&D programme

The irradiation programme continues :

- CAPRIX 1 is always waiting for the re-start of PHENIX

- TRABANT 1 pins 1 and 3 are now out of HFR reactor because of pin failures. For the first one filled with $(U, Pu_{0.45})O_2$, no preliminary explanations must be seriously proposed between a device or a fuel problem. For the second one, the failure which concerns only one of the half-pin filled with (Pu, Ce)O_{2-x} is due to the bad thermal conductivity of the fuel and consequently its melting. The second half-pin filled with the same fuel but with a higher O/Pu+Ce ratio has had a good behaviour. These two pins will be transported to Cadarache for examinations in 1997.

- CAPRA 1A and 1B have been fabricated by COGEMA. These two S/As are now at SUPERPHENIX, waiting for its re-start.

- Definitions of TRABANT 2 and PUNPOM experiments for HFR reactor are performed and an important irradiation programme has been defined in BOR60 (Russia) and will be launched in 1997.

The first phase of the CIRANO programme, which consisted in removing the blankets, is now ended and the second phase which consists in modifying the isotopics vector and the Pu enrichment is launched. A newly developed calculation scheme (ERALIB/ECCO/ERANOS) will be provided in september 1997.

Thermohydraulic experiments have been realized on the CAPRA1 mock-up.

7. THE EUROPEAN FAST REACTOR PROGRAMME

G. HUBERT (EDF), JC. LEFEVRE (FRAMATOME-NOVATOME)

7.1. Introduction

The European Fast Reactor (EFR) is the product of the development programme in Europe and embodies the experience from the development and prototype reactors in France, Britain and Germany. EFR has reached the point where prospects can be seen for competitive electricity generation, with a level of safety equivalent to that of up-to-date NPPs, typically represented by the European Utilities Requirement being prepared for future LWRs.

The ultimate objective for fast reactors has always been to maximise the utilisation of the natural uranium resource and in common with the main development programmes world wide, EFR has pursued the sodium coolant technology. The safety approach recognises the differing requirements of a sodium cooled fast reactor core compared to the established water and gas cooled thermal reactors which has resulted in a different balance between prevention and mitigation with consequences for the shutdown, decay heat removal and containment systems.

The success of Superph,nix is now most important for the progress to commercial exploitation. The feedback of Superph,nix operating experience to the commercial designs of the future is formalised in the 'Knowledge Acquisition Programme' (KAP). The interaction with the safety authorities is part of this feedback of experience, particularly for issues raised as a result of the unanticipated problems encountered with operation of such a prototype commercial sized plant.

Against this background a renewed strategy for the European Fast Reactor has been agreed by all the involved parties from the Utilities, Design and R&D organisations. The strategy recognises the significance of the potential missions for the fast reactor and the perspective in the medium term of an adequate uranium supply for the established thermal reactor systems without the need for a decision on the construction of the commercial series of fast reactor during the next decade. This time which is available must be used to fully establish confidence in the technology with attention on the key issues affecting operability, inspectability and repairability all of which reflect on the availability and economics and indirectly on the safe operation of the reactor system.

Subsequent to the preparation of the renewed EFR strategy the role of Superph,nix in relation to the back end of the fuel cycle has been reinforced by the findings of the French Government

Commission to evaluate the capacity of the reactor as a research tool (the so-called 'Castaing Commission') which gives its support to the strategy.

The on-going programme therefore takes full benefit of the Superph,nix Knowledge Acquisition Programme, addresses the issues that are raised concerning sodium coolant and inspectability, seeks to exploit the flexibility of fast reactors in the fuel cycle and explore innovative concepts for the improvement of the EFR Reference Design.

Finally, in order to respond to issues raised during licensing it is necessary to have a well balanced assessment of the reactor concept using sodium coolant compared to other coolants and to review the basis for taking the earlier decisions in the light of the current changed priorities concerning plutonium production, minor actinide transmutation and inspectability.

7.2. Strategic Objectives

In the past fast reactor development has been mainly oriented towards the breeding mission, which according to the renewed strategy remains the long term objective, but now the role in providing solutions for the backend of the fuel cycle needs to be investigated.

In fact the strategy recognises that for the breeding mission there is no longer a need for very high breeding gain for the introduction of fast reactors due to the plutonium production from installed LWRs, only modest breeding would suffice to sustain a series of fast reactors. The new missions for plutonium and minor actinide burning can use similar technology and so maintain compatibility with the development for the breeder mission. It is, however, required that for all missions the reactor be an economic power generator.

In parallel to the continuing improvement of the EFR Reference Design consideration should be given to innovative design solutions which are not inhibited by the extent of validation needed or the timescale for their introduction in the reference design.

Finally according to the currently anticipated timescale to commercial exploitation an important objective of the programme is to promote international collaboration to maximise the mutual benefit of the national programmes. So a programme of broad interest is pursued which includes elements aimed specifically at collaboration with shared tasks which by themselves can bring significant benefit but can also lead to an improved understanding, possibly to common specifications and eventually to joint projects.

7.3. Key Dates for the European Programme

The concept design phase has been completed by the end of 1993, i.e., the so called "Phase 2: EFR Concept Validation" which achieved the twin goals set by the European Utilities:

- economic performance with the generating cost of a commercial series of EFR competitive with its contemporary LWR, and

- licensable in all participating countries with a requirement for a safety level which meets the ambitious targets of future nuclear plants.

The on-going EFR programme was put in place after EFRUG decided to delay any decision to enter the preconstruction phase which had always been intended after a period (several years) of successful operation of Superph,nix. This renewed programme is conditioned by the following prevailing circumstances:

- although there exists a large size operating plant, Superph,nix, and a well validated reference design intended as the first of a series of commercial reactors, EFR, there are a number of important issues to be addressed before commercialisation,

- the time when fast reactors must be commercially introduced as an energy source which is independent of natural uranium supply cannot be precisely predicted but is several decades into the future,

- as well as the fundamental interest in fast reactor as the most efficient utilisation of uranium there is interest in the potential role related to the back-end of the fuel cycle, and

- there is a developing international collaboration which is assuring mutual benefit from the individual national experience and is continuously improving the ability for participation in joint activities for more efficient progress to commercialisation.

In the short term the design and corresponding R&D programmes are well defined and in accordance with the renewed EFRUG strategy. The key milestones are:

<u>1998 Innovative concepts review</u>

- Review the basic concept options for fast reactors against current conditions and priorities.

- Select the innovative options on the basis that either they can be introduced in the EFR Reference Design or are worthy of being pursued further on their particular merits as an alternative

<u>2000 EFR Reference Design</u>

The progressive introduction of the outcome of the studies to be completed for a comprehensive update of the EFR Reference Design:

- a design capable of accommodating a broad range of fuel qualities,

- a design which draws on the lessons from the experience of construction and operation of fast reactors worldwide.

- the completion of a number of generic and joint international collaborative tasks (also to identify prospects for widening the collaboration), and

- an update of the EFR documentation on safety propositions, R&D requirements for full validation of the design, and an update of the economic assessment.

For the medium term the important elements are the continued operation of Superph,nix and establishing, through the CAPRA programme, the wider interest in exploiting the flexibility of fast reactors to adapt to the changing circumstances arising from the back-end of the fuel cycle, with key dates as follows:

1998 Complete irradiation of CAPRA type S/As (Pu content ÷ 30%), isotope irradiation and actinide burning tests in Superph,nix.
Ordering of the third core for Superphenix.

1999 Loading of the second core in Superphenix.
Introduction of new irradiation tests, actinide burning (Am, Np) tests, and new CAPRA type S/As with high Pu content pins.

2004 Loading of the third core in Superph,nix.

Continue irradiation of advanced fuel materials, actinide burning tests, and loading of prototypic CAPRA S/As in Superph,nix.

2006 Term of the 1991 law on nuclear waste management in France, establishing the interest in this role for fast reactors.

7.4. Programme content and current activities

The design programme and corresponding R&D programme are closely integrated by the common objectives outlined above. The design programme is broken down into six activity areas :

- EFR Reference Design
- Feedback from Operating Plant Experience
- Generic studies
- Specifications, Codes and Standards
- Innovative concept studies
- International collaboration

The practice of the EFR project continues to be pursued, i.e., to have a secure Reference Design offering the possibility for a decision to construct with introduction of improvements only when they are suitably validated. Flexibility is retained to accommodate a range of breeder and burner cores with the minimum of adaptation of the reactor design.

The feedback from the operating fast reactors is achieved through a systematic review of the design of each system and component in the light of the European reactor experience and to the extent possible all fast reactor experience worldwide.

So the Reference Design is the evolution of the EFR Consistent Design from the Phase 2 studies. It is a design where the economic goal is achieved through a compact layout by minimising the number of main components (6 Intermediate Heat Exchangers IHXs and 3 Primary Sodium Pumps) and a small rotating shield (see Table 1). The 6 IHXs are connected to 6 large straight tube Steam Generators by 6 separate secondary circuits having rejected the lower cost option of pairing into 3 circuits in the interests of a potentially higher availability. The safety goals are achieved with the established fast reactor practice of diverse shutdown systems but supplemented with risk minimisation measures known as the 'third shutdown level', and a high reliability Decay Heat Removal system which takes advantage of the good characteristics for natural circulation.

A first round of the systematic review of the reactor operating experience has been completed for the absorber rod drive mechanisms, main sodium pumps and intermediate heat exchangers (IHXs) which has revealed sensitive points for EFR. Examples are the importance of succeeding with the EFR features which prevent the sodium aerosol problems on the absorber mechanisms, the need to confirm adequate operational flexibility whilst avoiding the check valve at the primary pump outlet and to ensure full benefit is taken of the piston seal experience for the IHX / pool boundary replacing the gas bell originally selected for EFR.

Integration of a plutonium burning core can lead to two EFR variants (breeder and burner) so as to avoid compromise for either mission as an economic power generator but it it desirable that the differences are not extensive. This desire that the core should be accommodated with minimum adaptation for the mission was demonstrated in the first CAPRA feasibility studies. A burner core with the power of EFR and the same core envelope and absorber rod arrangement was proposed but with a different pin diameter and fuel subassembly pitch to improve the plutonium burning efficiency (see Table 2). So the diagrid and rotating plug overall dimensions remain the same for breeder and burner cores, but have to be adapted to the subassembly pitch, giving an otherwise identical reactor based on the same technology, i.e., oxide fuel and similar safety characteristics and reactor behaviour.

TABLE 1. EFR LEADING STATION PERFORMANCE PARAMETERS

Reactor heat output (MWt)	3600
Alternator output (MWe)	1580
Nett electrical output (MWe)	1470
Nett efficiency	0.41
Core inlet temperature (øC)	395
Core outlet temperature (øC)	545
Feedwater temperature (øC)	240
Steam temperature (øC)	490
Steam pressure (at SGU) (bar)	185
Primary sodium flowrate (kg s-1)	20172
Secondary sodium flowrate (kg s-1)	15330

TABLE 2. EFR CORE PARAMETERS FOR 'BREEDER AND BURNER' VARIANTS

	EFR Core(*)	CAPRA Core(**)
Number of fissile S/As	387	366
Number of diluent S/As	1 (centre)	52
Number of control and shutdown S/As	24	24
Number of diverse shutdown S/As	9	9
S/A pitch (mm)	188	181,4
Number of pins per S/A	331	469
Number of fissile pins per S/A	331	336
Pin length (mm)	2645	2305
Fissile column length (mm)	1000	1000
Clad external diameter (mm)	8.2	6.35
Fuel residence time (efpd)	5 x 340	3 x 285
Breeding Gain range	-0.2 / +0.15	-0,65

(*) EFR core CD 91 from Phase 2 studies
(**) CAPRA reference core 04/94 from feasibility study

The generic studies, which address fundamental fast reactor concerns are of broad interest generally to sodium cooled fast reactors and are not specific to a particular design so have elements which form part of the international collaboration. The studies are in continuity with Phase 2 and take into account the recommendations from the Ad Hoc Safety Club, the group of European independent safety specialists who assessed the safety of the EFR design and the evolution of the safety requirements for future plants. The generic tasks include:

- Core safety: Whole Core Accident (WCA) prevention and computing tools
- Containment enhancement

- Sodium fire prevention and mitigation
- DHR function optimisation
- ISI & Repair capability improvement.

The objective is to propose improvements and/or increase the confidence in the EFR Reference Design through an improved technological base. The proposal for the rectangular containment is an outcome of the generic studies. These have demonstrated that the consequences of the WCA outside the main vessel and reactor roof are limited to the leakage of a quantity of primary sodium with the associated contamination to the above roof area. Also, the only source of pressurisation of the containment building that is significant is a sodium fire, and for a primary sodium fire, for which the containment building must be effective, the pressure can be limited well within the capability of a rectangular building. The study showed the importance of the segregation of the above roof area from the rest of the containment building which is achieved naturally by the 'polar table'. This consists of a closure on top of the above roof area, with limited leakage path to the crane hall, which can be readily enhanced by some provisions as a fire barrier.

Prominence is given to the continued development of the specifications, design rules and computing tools in the programme as one of the six activity areas. Again because it is of broad interest generally to sodium cooled fast reactors, it is particularly fruitful for international collaboration and can be a bridge to improved understanding and eventually possibly to joint projects.

For the specifications there is a parallel activity to that on LWR where a European Utilities Requirement (EUR) document is being prepared for future reactors covering design, operation and safety. The task amounts to a comparison of the EFRUG specification for EFR with the EUR for LWR with the interest to have common specifications as the most straight forward way to demonstrate equivalence of the reactor systems and avoid differences which can be an implied criticism of one system or the other. However, because the characteristics of the reactor systems differ there are difficulties to be overcome in order to achieve common specifications, the containment mentioned earlier being an example where the release criteria can and should be the same but the requirement of the safety functions to meet the criteria are not necessarily the same, with fast reactors having a different balance of contribution between prevention of faults leading to radioactive release achieved by the shutdown and heat removal systems and mitigation provided by the containment.

Development and application of the design and construction rules is a continuous process to reflect the experience being gained and improved understanding. The omissions in the existing rules concerning data and methods are addressed and simplified methods of analysis without the penalising conservatism are researched.

There is flexibility in the programme so that in parallel to the development of the Reference Design a broad range of innovative and alternative concepts will be considered. The innovative designs based on the sodium cooled pool reactor concept have the potential to be adopted as a package or for particular features to be incorporated in the Reference Design in the medium term. A number of reactor structural arrangements have been selected for their potential to improve inspectability and give confidence that significant benefits will eventually be realised. These designs are based upon advanced manufacturing process rationale, improved weld configurations and reduced numbers of welds which also bring benefits to the inherent integrity of the structures.

For the alternative concepts the interest in the first place is to be informed, to be sure that the correct decisions have been taken for EFR, provide the convincing case for the EFR design as appropriate, to be fully aware of the interests of the collaborating partners where they promote the alternative concepts and to the extent possible feedback ideas into the EFR design.

The R&D programme and the interactions with design activities include:

- Support to the operating reactors,
- CAPRA and a number of development activities related to core safety,
- ISI and repair methods and technique development,
- International collaboration

The success of the operating reactors is the highest priority in deciding the allocation of R&D resource. Much of this, although directed specifically at Ph,nix and Superph,nix, brings benefit for the future plants with examples being the support needed for the licensing case with respect to sodium fires and the structural integrity case for continued operation and life extension for Ph,nix and previously for PFR. Even though PFR has reached the end of operation the next step of decommissioning is also important for the acceptance of nuclear power and can have a useful feedback to future reactor design.

The development of ISI techniques and repair methods is important for the operating reactors but the programme also has a longer term perspective for the future plants where designs can be adapted to the promising techniques.

International collaboration is an activity promoted by each organisation. The R&D side has traditionally taken the responsibility for establishing the agreements but increasingly the utilities are taking initiatives and their sponsorship of the designers includes collaborative work. So particular aspects of the generic studies undertaken in the design programme are promoted as joint tasks within the international collaboration to improve mutual understanding in all aspects of each other's designs (requirements, regulations and practices and design solutions) and seek a common technical approach to the key issues for fast reactor (e.g., safety specifications, severe accident treatment and the consequence on design and safety justification and the role of fast reactor in the fuel cycle).

In each of the generic areas listed above there are collaborative activities between either or both the design and R&D organisations. This includes specific joint tasks between European and Japanese design companies in each area, and generally on the safety functions between European and Russian companies. In the development and validation of codes there are a number of shared activities between the R&D organisations.

7.5. Conclusions

There is a renewed strategy for fast reactor development in Europe responding to the changing conditions for nuclear power generally and the introduction of fast reactors, recognising their already advanced stage of development in Europe but always with the ultimate goal for the most efficient utilisation of natural uranium.

The key issues for sodium cooled reactors are addressed in the near and medium term design and development programme. This programme takes advantage of international collaboration and has as an important objective the expansion of the collaboration.

The steps needed beyond the near and medium term programme and the need and timing of a further demonstration plant will depend upon the outcome of the programme concerning:

- the evolution of the Reference Design with the importance being placed on achieving improved inspectability and maintainability,

- the missions for fast reactor and the need for a demonstration of these missions, and

- the development of international collaboration

8. R&D

8.1 Fuel element and core material

- Irradiation progress

In 1996 because of the extended shutdown of PHENIX no further progress could be achieved in dose/burn-up values.

The leading subassembly with reference cladding (CW 15-15 Ti) and an EM10 wrapper, reached # 155 dpa at the end of the 49 th cycle of PHENIX (April 1995).

This subassembly was examined (non-destructive examinations). The main results are :

- excellent behaviour of EM10 wrapper (deformation of external across flats (wrapper tube) less than 0,5 %, non measurable elongation and deformation),
- significant deformation of CW 15-15 Ti cladding and consequently mechanical interaction between fuel pins.

- The TRABANT 1 irradiation in the HFR reactor.

The irradiation experiment TRABANT (Transmutation and Burning of Actinides) was planned and executed in a trilateral collaboration CEA, FZK and Institute for Transuranium Element (ITU) within the framework of the CAPRA project. This experiment contained three pins in capsules.

Fuel pin 1 : one pin with a short fissile column, annular $UPuO_2$ (45 % Pu fuel). This pin was fabricated by CEA-CADARACHE.

Fuel pin 2 : one annular oxide pin with high Pu content (40 %) and 5 % Np content ($U_{0,55}$ - $Pu_{0,40}$ - $Np_{0,05})O_2$. This pin was fabricated by ITU.

Fuel pin 3 : one Pu without U solid pin in two parts :
The upper part filled with $(Pu_{0,43} - Ce_{0,57})O_{2-X}$
The lower part filled with $(Pu_{0,43} - Ce_{0,57})O_2$
This pin was also fabricated by ITU.

Irradiation of pin 1 started in July 1995 at HFR Petten and reached around 6 at % burn-up- Non-destructive examinations were carried-out in November 95 and did not reveal any anomaly. The irradiation of this pin 1 and fresh pin 2 and pin 3 started in December 1995. Intermediate non-destructive examinations were planned in July 1996.

Neutronographies showed :

- a rupture at top of fissile column of pin 1 with possibility of fuel melting
- no anomaly of the pin 2
- a fuel melting of the upper part of the pin 3.

The bad behaviour of the pin 3 was not completely unexpected. Destructive examinations are necessary in order to obtain an explanation of the rupture of pin 1.

It is planned to present the above results at GLOBAL-97.

- Fuel modelling :

- The version GERMINAL 1.3 was released in 1996.

GERMINAL 1.3 was validated on around 30 tests including the CABRI tests : PF1, PFX, E9, JOG1, JOG2, AG ϕ (ref. 2). This qualification base has been extended to recent irradiations :

IFOP1 : annular $UPuO_2$ (45 % Pu) fuel at 1.15 at % Pu.

SUPERFACT 1 : annular SUPERPHENIX $UPuO_2$ (24 % Pu ; with 2 % Np) fuel at 6,6 at % Pu.

The coupling of GERMINAL 1.3 with a detailed thermomechanical clad module is in progress.

- The ICARG code development has been started. The goal of this code is to calculate the consequences of a clad rupture (oxide fuel - sodium chemical reaction). The qualification base consists of 7 clad ruptures in PHENIX and SILOE.
- Studies on Oxide Dispersion Strengthened (OPS) ferritic steel have been started regarding weldability and new grades are developed based on EM10 ferritic stainless-steel.

- Absorber element

The results of the post-irradiation examination of the ANTIMAG experiment irradiated in PHENIX (250 10^{20} capture/cm^3) showed the efficiency of the shroud around B_4C pellets : B_4C fragments restrained, limitation of clad carburation and showed also saturation of dedensification at high burnup.

B_4C pellets were fabricated by carbothermic process and characterized.

8.2. Core physics

8.2.1. Nuclear data and the "neutronic formulaire" ERANOS

This ERALIB1 adjusted library has been derived from a 1968 group application library based on JEF2.2 and a large integral data base (355 integral parameters from 71 different systems) containing the ad hoc required data to validate the cross sections for the major nuclear processes. The consistency of the integral and microscopic information is demonstrated by using the rules of information theory and a simple recipe to identify the nonconsistent integral data. The energy scheme used for the statistical consistent adjustment procedure has been designed to optimize the decoupling of cross section effects.

The performance of ERALIB1 for fast reactor applications is considered to be satisfactory. An example of improvement with ERALIB1 is shown on fig. 8.1. where relative differences between experimental and calculated critical masses are shown according to spectral hardness index.

This work is the first step in the process towards a unique data set which will be valid for all applications (core neutronics, shielding, fuel cycle) and for all types of fission reactors and consequently the integral data base needs to be enlarged. Complementary information on ERALIB1 is given in ref. 3.

- The development of the neutronic formulaire for fast neutron reactors ERANOS has been satisfactory. In 1996, for the first time 3D - transport calculations are currently performed for PHENIX and SUPERPHENIX.

Former configurations SUPERPHENIX have been analysed and differences between experiments and calculations have been explained combining this 3-D transport calculations with detailed thermohydraulics calculations (for example SUPERPHENIX power distribution).

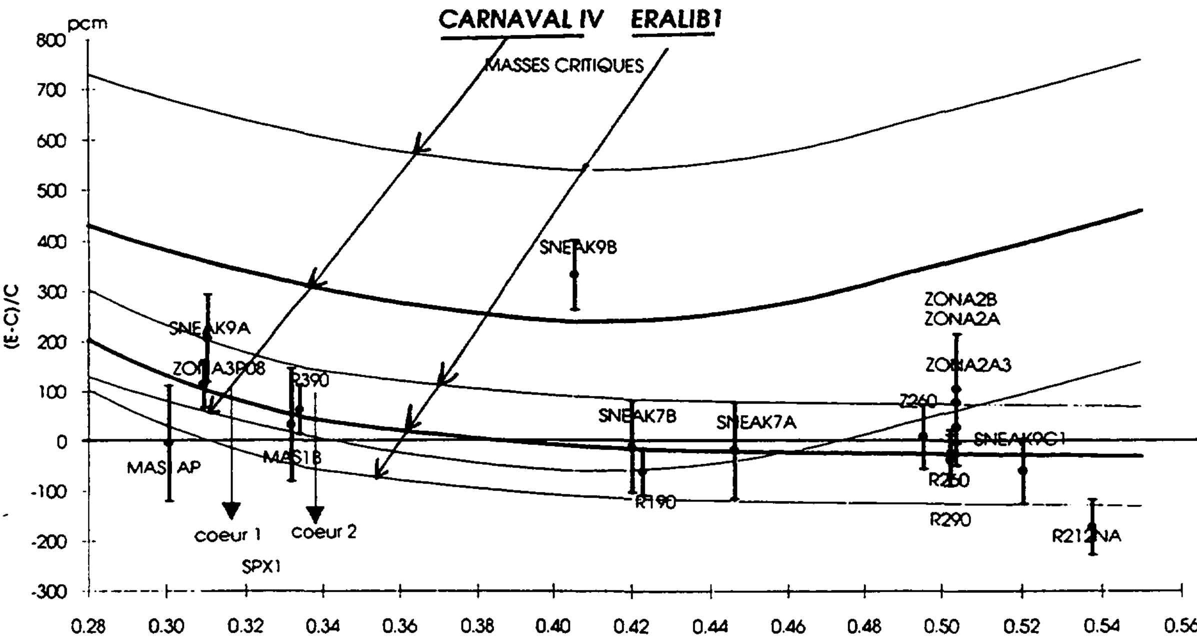


Fig. 8.1. - Critical Masses - Distribution of (E - C) / C values before and after adjustment, according to special hardness index r

8.2.2. The CIRANO programme

The CIRANO programme is a part of the various programmes (neutronics, fuel, ...) carried out to demonstrate the feasibility of using a fast reactor to burn as much plutonium as possible.

In order to reach this purpose, the U^{238} content must be as low as possible. Consequently, with respect to breeder cores, this requires :

- Removing the fertile axial and radial blankets,
- Increasing the plutonium enrichment,
- Substituting inert material for the U^{238}.

The CIRANO programme is conducted at the MASURCA critical facility at CADARACHE.

Only a few experiments were performed without fertile blankets and stainless steel reflectors and it seems that the discrepancies between the experiments and calculations increase as the thickness of the blanket decreases. Therefore, the first set of studies consisted in gradually removing blankets from ZONA 2A (classical UO2 + PuO2 Core) as in shown is the following schedule :

<u>Phase 1</u> :

April 1994	ZONA2-A	reference core (with axial and radial blankets) (ref. 2),
September 1994	ZONA2-A3	radial blanket removed and replaced by sodium-steel elements,
February 1995	ZONA2-B	core without blankets.

Phase 2 :

July 1995	ZONA2-B.SI	core without blanket and with internal storage (SI) : without B4C axial shielding on half the core and with an extension of the UO2/Na radial reflector on the same half of the core,
October 1995	ZONA2-B.SI1	Same configuration as ZONA-2B-SI, but fuel subassemblies set up in the internal storage area (12 tubes).
December 1995	ZONA-2B.SI2	Similar to ZONA 2-B SI1 but with a protection of B4C SA between the core and SI location.

All the above experiments were performed with the following Pu enrichment (% atoms) and isotope vector (% atoms).

	E % (atoms)	Isotopic Vector (% atoms)						
	Pu	Pu238	Pu239	Pu240	Pu241	Pu242	Am241	r
ZONA2 PIT	27.5		76.4	18.5	1.7	0.7	2.7	0.523

Phase 3 :

From March 1996 to late 96, the ZONA-2B core configuration (ref. 2) was studied but substitutions of various fuel types were made in a small control zone (ϕ = 30 cm ; more precisely equivalent of 6 tubes. The total core consists of 56.25 tubes) :

- ZONA2-POA in May 1996. The central zone had a E % (atoms)Pu = 27.5 % and Pu 240 = 8 %.
- ZONA2-P2K in June 1996. The central zone had a E % (atoms) Pu = 30 % and Pu 240 = 31 %.

The usual following measurements were performed

- critical masses
- fission rate axial and radial traverses
- spectral indexes
- γ heating (ionization chambers and TLD).

This central zone is too small to establish an asymptotic spectrum. Thus the most useful parameters are the reactivity worths of the substitutions.

From July to September 1996 subcritical configurations with an external source were studied (ISAAC programme).

Phase 4 :

This phase will occur in 1997. It will consist of a full core simulation of a CAPRA core (Pu enrichment 44 %). The MAZE core will be built with cleaner plutonium from ZEBRA platelets with a very heterogeneous cell design. The following table summarises the characteristics (enrichment, Pu isotopics, and spectral hardness for SUPERPHENIX, CAPRA and MAZE cores), and it indicates that the "equivalent" MAZE core (where the Pu isotopic vector is artificially made to be that of the CAPRA core) is representative of the CAPRA core.

Characteristics of the MAZE Core

	E % (atoms)	Isotopic Vector (% atoms)						
	Pu	Pu238	Pu239	Pu240	Pu241	Pu242	Am241	r
SUPERPHENIX C1	13.9	0.4	69.2	23.0	4.8	1.6	1.0	0.324
CAPRA C1	41.3	5.7	39.2	26.7	13.0	14.3	1.3	0.397
ZONA2 PIT	27.5		76.4	18.5	1.7	0.7	2.7	0.523
MAZE	44.2		83.7	13.9	1.7	0.3	0.5	0.480
MAZE Equi	44.2	5.7	39.2	26.7	13.0	14.3	1.3	0.393

$$\text{Note : } r = \frac{\overline{\nu \sum_f}}{\xi \sum_s}$$

Thus, except for the effects of higher Pu isotope cross sections which are obtained from Phase 3, MAZE will give physics information on the CAPRA "dilution" concept cell heterogeneity. Specific measurements of sodium void worth, diluant and rod worths and their consequences on reaction rate traverses are also planned in 1997.

In order to prepare this phase the central zone of ZONA2 B (ϕ = 30 cm) was loaded with SNEAK platelets in order to assess the influence of platelets versus rodlets (in the central zone the E_{Pu} = 26 %, Pu - 240 fraction = 8 %) This configuration ZONA2 K was tested in October 1996.

8.3. Safety

8.3.1. Core surveillance

8.3.1.1. Core temperature surveillance and monitoring

The ALPES system should improve the core thermal surveillance by numeric processing of the thermocouple temperature measurements at subassembly outlets. This system, still in the experimental stage, has been set-up on-line at SUPERPHENIX since July 1995.

In order to determine the performance of the ALPES system with respect to the detection of an inadvertent control-rod withdrawal incident, control-rod insertion tests were performed at SUPERPHENIX in December 1996, at power level of 50 and 80 % nominal power.

Neutronic calculations have shown that the consequences of such a test are symmetrical to those of a control-rod withdrawal.

The tests were successful and showed the possibility of detecting the incident in good conditions. Different algorithms were tested, the simplest being :

$$S_{RIB} = \frac{S_1 - S_2}{S_2}$$

S1 increase of temperature around the concerned rod
S2 increase of temperature around the rod diametrically opposite
Numeric filtering allows to keep fast variation of the signal for comparison with an absolute threshold. The threshold has been determined in order to avoid spurious detection (less than 1 for 10 years of operation).

Such a signal is shown on figure 8.2.
In this test the total reactivity insertion was - 14 pcm and rod insertion was 85 mm.

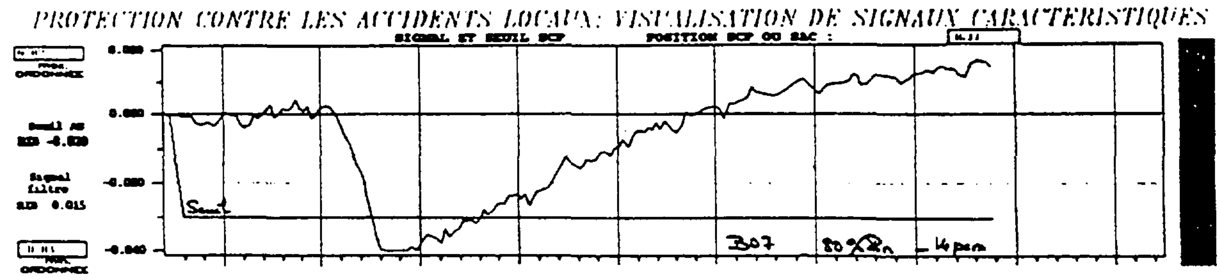


Fig. 8.2. - S_{RIB} signal

8.3.2.2. Clad failure detection

In PHENIX and SUPERPHENIX, primary sodium is taken directly from the vessel and transported by an assembly of pipes and pumps to a module located on the reactor slab. The presence of any delayed neutron emitters is then revealed by neutron counters.

In SUPERPHENIX, this surveillance is carried out by 8 modules (1/4 of the core per module) while in PHENIX a single module receives the sodium taken at right angles from each of the 6 intermediate heat exchangers.

The main difference between the two reactors concerns the transit time between the assembly and the detection system. This time is shorter in SPX than in PHENIX :

PHENIX : approx. 30 s, of which 26 s are due the sampling system

SPX : from 13 to 16.5 s depending on the region of the core (periphery or center), of which 6 s are due to the sampling system.

The sampling device (piping, electromagnetic pumps, moving parts, etc.) is very complex so, for this reason, a new system was tested in SPX.

The experiments are aimed at finding a system to measure delayed neutron emitters directly in the vessel. This offers the following advantages : no sampling device is required, the system is simplified, costs are reduced, safety is enhanced through a reduction in response times and the system fully complies with the integrated concept.

In order to do this, a new type of detector (high-temperature, high-sensitivity fission chamber) was developed in France and tested in PHENIX.

The I-DND system was thus installed in SPX from the very beginning of the operation of this reactor, on an experimental basis to validate the detection principle and measure its performances in comparison with the sampling system (during the CARAMEL tests).

These tests proved the validity of this detection principle and showed that its efficiency was similar to that of the conventional system and that transit times were shorter.

In order to determine I-DND thresholds and to acquire a better experience, the experimental SATIN 2000 system has been set-up on line at SUPERPHENIX in 1996.

8.3.2. CABRI and SCARABEE programmes

As reported last year [Ref. 2], the CABRI-FAST international programme conducted by CEA-EDF and PNC (Japan) has been completed with the LT1 test, performed in December 1995.

During 1996 it was agreed by CEA-EDF and PNC to conduct the RAFT programme (Reactivity Accident Fuel Tests).

The objectives of this programme are :

<u>Local fault (Control-Rod Withdrawal Accident scenario)</u>

- To study the possibilities of fuel ejection under the following conditions : the induction of a clad failure during a slow ramp rate when a low melt fraction (around 10 %) is obtained : RB1 test to be performed in CABRI.

- The coolativily of a bundle with pins having a low melt fraction in case of a rupture and fuel ejection from one pin : RB3 test to be performed in SCARABEE.

- <u>Transition phase studies</u> (beyond the initiation phase of a Hypothetical Core Disruptive Accident)

Melting of an irradiated 37 pin bundle (KNKII pin, 3,5 at %) following a total instantaneous blockage (Transition Phase TP1 test to be performed in SCARABEE).

Other tests are still at level of feasibility studies. They concern the problem of relocation of material in control rod subassemblies in case of molten pool in the core ; evaluation of heat transfers between fuel and steel in molten pool ; fuel motion in a three-pin cluster.

There was no test in 1996 because of the difficulties in the preparation of the RB1 test, essentially realisation of a notch filled with a solder on an irradiated pin (SCARABIX pin, ref. 2).

8.3.3. SCARABEE in-pile cell loop

Because of the discovery a few years ago of a very small sodium leak on the in-pile cell loop containing the test device it was decided to set-up a new in-pile cell.

The work started in January 1996 and should be achieved in 1997 in order to realize the TP1 test at beginning of 1998.

8.3.4. Hypothetical Core Disruptive Accidents code

8.3.4.1. Initiative Phase : SAS-4A code

SAS-4A is still being developed in collaboration with FZKA, PNC and CEA.

In 1996 the SAS-4A Ref 96 R1 has been developed. The model improvements concerns :

- modelling of in-pin fuel motion (EJECT Model)
- improvement of sodium boiling model
- plenum gas blow-out
- fuel stub motion simplified model
- improvement of thermomechanical fuel and clad modelling (DEFORM-4C)
- control rod expansion and diagrid modelling.

Validation of SAS-4A is still in progress on the basis of previous CABRI tests.

8.3.4.2. Transition Phase

Regarding the SIMMER III code, being developed by PNC (Japan), the Europeans (CEA, FZKA) are participating in the qualification of the code.

8.3.4.3. Expansion Phase

Important work has been made on this important phase of HCDA with the SIMMER II code and the "Advanced Fluid Dynamics Model" (AFDM code).

8.3.5. Source term and radiological safety analysis

The transport of the fuel aerosols 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. Papers on this code were presented during the IWGFR TCM [ref. 4] on "Evaluation of Radioactive Materials Release and Sodium Fires in Fast Reactors".

8.3.6. Sodium fires

Experimental work on sodium fire was completed in 1993.

Interpretation of the experimental observations is still in progress using the computer codes PULSAR and FEUMIX.

In order to acquire data to extrapolate the sodium experimental results for large leak rates (a few tons/s) the AIRBUS programme is still in progress. The goal of this programme is to measure the Sih coefficient (interface 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. The understanding of such experiments raises difficulties, not yet solved.

8.4. Technology

8.4.1. Complementary information about the March 31, 1994 accident at Cadarache.

The first information on this accident was given at the meeting of May 1995 (ref. 1). A more detailed information was given at the meeting of May 1996 (ref. 2).

It should be recalled that the accident took place in a basement next to the leaktight containment building of the reactor, at the end of dismantling operations. An explosion occurred during the destruction of 100 to 150 kg of sodium that remained at the bottom of a 55 m^3 tank. This destruction was performed by successive introductions of 50 l of ethylcarbitol. A member of the CEA team was killed and four other people were injured.

An internal Inquiry Commission was set up by the CEA on July 1996. The results of its works are summarized below ;

⇨ The accident area was cleared. The recovery of the various parts of the circuits and the metrology of the tank showed the agreement of the installation with the initial specifications ;

⇨ Metallurgical expertises are not yet completed because delays related to the administrative and judicial inquiries. The available results confirms those presented in 1996 :

- the pneumatic explosion of the tank,
- the tearing of the metal plate of the tank in the base metal,
- the rupture pressure was above design pressure

⇨ Investigations confirmed that the chemical reactions at the origin of the accident are numerous, complex and violent. It was not possible to identify completely :

- the initiating causes of the runaway chemical reactions,
- the catalytic reactions that may occur in order to give a completely satisfactory explanation of the accident.

Uncertainties stress the need to maintain the suspension of ethylcarbitol cleaning of sodium.

8.4.2. Sodium technology

8.4.2.1. Sodium leak detection

In order to improve the knowledge on sodium leak detection, the FUTUNA 2 test facility has been set-up during 1996. The tests to be performed will study sodium leaks in the range 0.05 cm3/mn to 30 cm^3/mn, this range being larger that the one of FUTUNA 1 (0,7 cm^3/mn to 10 cm^3/mn - ref. 2). The heat insulation will be either in contact with the pipe or not.

The "signature" of the loss of insulation will be studied in order to propose methods to identify spurious alarms.

The transport and detection of sodium aerosols in the argon space between a small leaked sodium tank and the heat insulation is studied in the MARENA facility.

8.4.2.2. Other sodium technology studies

- The study of the behaviour of impurities in sodium has been completed especially to identify deposits in the plugging meter (ACROPOLIS programme).

- The law of oxygen consumption at the sodium free surface was determined in the range of temperature 140°C - 250°C (EDONIS tests). The upper limit was imposed by experimental difficulties.

- The technical files regarding the set-up of one oxygen meter and one tritium meter in PHENIX were issued. The decision to implement such meters will be taken in 1997.

- The sulfophosphoric decontamination process was used on three PHENIX IHX. Contamination was measured with γ spectrometry at various stages of the operation and numerous chemical analysis were made.

8.4.3. Thermohydraulic studies

In 1996 most of the thermohydraulics studies were related to PHENIX and SUPERPHENIX applications. Furthermore there is a continuous effort in order to develop and qualify the computer tools.

8.4.3.1. The TRIO-VF computer code

The TRIO-VF computer code has been developed at CEA in order to calculate 3D fluid dynamics. The main characteristics are the following.

- Complex geometries with cartesian, cylindrical and curvilinear orthogonal coordinates.
- Possibility to describe internal obstacles and inclined walls.
- Modelisation with porous media is available.
- The steady or unsteady Navier-Stokes equations are solved using the finite volume discretization on a staggered mesh.
- The standard turbulence model K-Epsilon or Large Eddy Simulation can be used (§ 8.4.3.2).
- The time discretization can be first or second order.
- The advection terms are discretized by a third order schema (Quick Sharp scheme).

The TRIO-VF has been validated in many number of configurations : free-convection, forced-convection, thermal coupling between fluid and surfaces, flows with water-condensation, ...

The TRIO-VF code is being used in many applications in the nuclear field : hot and cold plena, IHX, steam-generators, subassemblies flows, ...

8.4.3.2. Numerical simulations of mixing phenomena between cold and hot fluid

Mixing phenomena between cold and hot fluids at the core exit, investigated in simulation experiments with air (AIRJECO) and sodium (NAJECO, were already presented [Réf. 1-2]. Simulations were obtained with Large Eddies Simulations (LES). The LES filtered Navier-Stokes equations were solved using the TRIO-VF code.

It was shown that the LES calculations reproduce properly average and fluctuation temperatures in case of mixing of hot and cold sodium jets having the same velocity (1m/s) as well as different jets velocity

8.4.3.3. Gas entrainment

Gas entrainment studies are in progress taking into account : identification of sources (free surface, pump shaft, spillway), behaviour and transport of bubbles (VIBUL computer code) possibility of gas pocket formation and removal.

In 1996 using the open litterature a model of the free surface deformation by a vortex has been developed. This model calculates the shape of the free surface, taking into account the local characteristics of the velocity field below the surface.

The test section VEDETTE has been defined to study vortices created by interference of two flows. For the range of "stable" vortices being studied, the above model overestimates the depth of the deformation by 10 to 30 %.

Works is in progress in the case of "instable" vortices, different geometries, ... and on improvement of measurement techniques and data processing.

8.4.3.4. Sodium aerosol physics

The TRIO-VF computer code was improved in order to calculate transport of sodium aerosols in gas volume. The generation phenomena and physics of sodium aerosols have been modelized as well as coupling between free convection, radiation transport and aerosols transport.

This modelling will be validated in the FRUCTIDOR mock-up (Fig. 8.3).

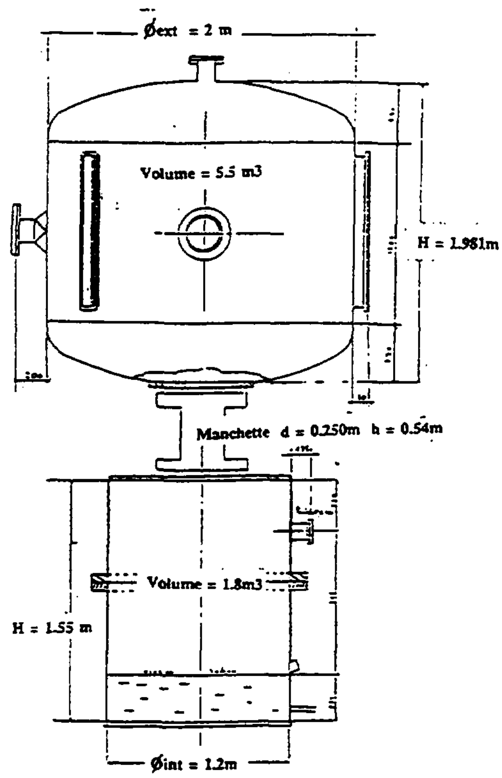


Fig. 8.3. - FRUCTIDOR mock-up

8.4.3.5. Thermohydraulics applications

The TRIO-VF computer code (§ 8.4.3.1) was used for reactor applications. For example the PHENIX core outlet thermohydraulics was studied with the L.E.S simulations (§ 8.4.3.2) in the following conditions :

- 234000 cells (10 mm mesh size in region below grid).
- about 50 turnover times (=Volume/Flowrate) needed to stabilize mean value and standard deviation of temperature.
- mean behaviour of flow similar to k-ε model prediction (cell by cell difference is below 11° on mean temperature at grid level).
- computing one turnover time required 7 hours on a 512 Mo-260 Mflops work station. Consequently the total computing time for 50 turnovers was 380 hours.

These calculations will be presented at NURETH-8.
Such calculations needs complementary validation on mock-ups : large scale water model of PHENIX hot pool operated at CEA (JESSICA mock-up).

Similar calculation were performed in the case of SUPERPHENIX to study the consequences of the removal of the first row of the radial blankets of the present core (ref.2). These calculations will be validated in water on a 1/10 scale model of SUPER-PHENIX at "Electricité de France" (SUPERDONALD mock-up) and in air on a 1/6 scale model at CEA (FREYJA mock-up).

8.4.4. Steam generators

ATLAS monotubular rig

The ATLAS monotubular rig, with a straight tube, was designed to obtained detailed data on heat transfer, pressure loss, dry-out correlations and flow instabilities in view of validation of the thermalhydraulics case of the EFR steam-generator units. The ATLAS rig was connected to the steam-water and sodium supply system of the SET facility at CEA-CADARACHE. The programme was successfully performed in 1995 with a series of 34 tests. Sensitivity studies on vapour pressure, flowrate and inlet temperature of sodium and water around nominal conditions (vapour pressure (185 bar)), in a large power range (20, 50, 80 and 100 % Nominal Power) were carried out. Around the 20 and 100 % Nominal Power Value, sensitivity studies were also performed. In addition, a series of experiments was performed at "low pressure" (60, 80 and 120 bars).

Analysis of the data was achieved in 1996 with the following main results.

- Heat transfer and pressure drop.

Correlation on heat transfer were obtained from a litterature survey. Calculations of the tests with these correlations were compared with the experimental data. One of these correlations gives a rather good agreement in the case of low pressure tests and a very good agreement in the case of nominal power tests.

The same approach was used for pressure loss giving a good agreement (10 %).

- Analysis of the experimental results showed the possibility to obtain large temperature differences (> 100°C) between liquid phase and vapour phase for low pressure experiments.

The litterature survey does not provide good local heat transfer correlations in the post-dryout zone.

- Analysis of dry-out and flow instabilities.

- No significant frequencies above 2.5 - 3 Hz.

- Dry-out length was characterised from flows parameters (heat transfer, mass velocity, pressure, steam quality).

In some experiments the water droplet velocity was estimated and compared to existing slip models.

8.4.5. In-service inspection and repair

8.4.5.1. Main vessel inspection

The MIR remote-contrôlled inspection device (Module d'Inspection des Reacteurs Rapides) was developed to monitor, throughout the lifetime of SUPER-PHENIX, the surface and internal defects on the main vessel of the reactor. Improvement of the MIR device is still in progress :

- Improvement for the inspection of the "triple point" (where the core support structure (support ring) rests on the main vessel) with respect to surface (emerging) defects. The procedure consists firstly in detecting the defect with an eddy current probe, then in dimensioning the defect with U.S transducers. An eddy current probe working at 180° has been fabricated. This probe has been tested at room temperature and will be tested at 180° C on a special mock-up. Regarding dimensioning of the defects, tests at room temperature are in progress using longitudinal waves at 60° and 2 MHz on realistic defects (cracks : h = 5, 10, 15 mn).

•	Studies on the volumetric control of the core support ring (fig. 8.4) resting on the main vessel at level of the "triple point" are in progress. Encouraging results were obtained using longitudinal waves in the axes of the support ring. These waves are generated with a composite transducer at 800 kHz. Detection with polarised horizontal waves at 45° and 800 kHz with successive reflections on the surface of the support ring is also tested. This type of waves is generated with an EMAT transducer (Electro-Magnetic Acoustic Transducer). The first results showed that this type of detection is not decisive.

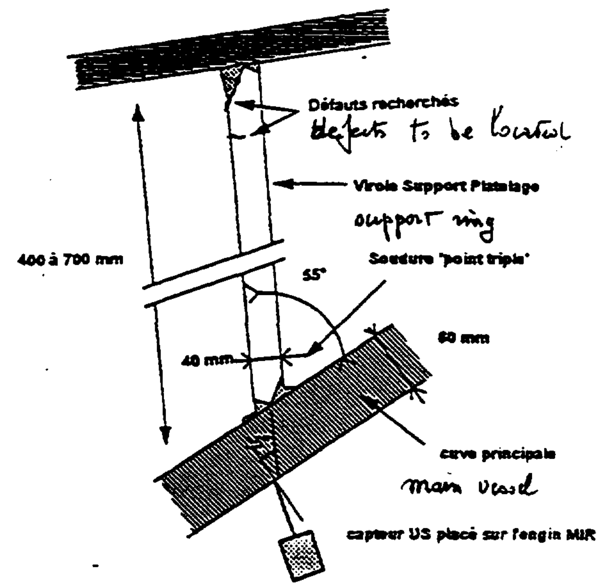


Fig. 8.4. Volumetric control of the core support ring

The use of multi-element transducers to check vessel welds is the subject of a more long-term improvement programme. With multi-element transducers, beam focus parameters can be modified according to the geometry or the thickness being inspected. The optimization of the transducer for each inspection configuration will permit improving the characterization and sizing of faults and reducing perturbation due to geometry. A transducer has been designed and built to check butt welds on the vessel. By adapting the delay laws of each element, it is possible to work in different configurations (45° or 70° angle of incidence, variable focus depth) and to correct perturbations of the beam with respect to differences of geometry (fit-up).

8.4.5.2. Structural monitoring in sodium

R&D work has been initiated with the aim to develop the means required for volumetric examination of the internal structures in sodium.

At present, as regards the non-destructive tests of the structures, there exists no device permitting the generation of ultrasound in sodium at high temperature. Two methods of non-destructive testing have been considered. The first is an electromagnetic method that uses electromagnetic acoustic transducers (EMATs). These transducers should allow the generation of ultrasound in the sodium via eddy current. The second method uses piezoelectric transducers made of a composite material. The aim is to develop transducers that are compatible both chemically and acoustically with sodium. Two studies are in progress to assess the technological feasibility of these transducers.

A method using multi-element transducers is also under development. The objective here is to perfect a method that will permit investigating the entire volume to be inspected when the translation possibilities of the transducer are restricted. The little space available for transducer movement will be compensated by angular scanning.

8.4.5.3. Ultrasonic telemetry

Ultrasonic telemetry in sodium is still in progress, mainly using signal processing (SYTRAC code).

Laser telemetry studies in air, including thermal stratification representative of the plena above the pool gave abnormal deviation of the beam : EPISTAR tests (limit conditions could explained this anomaly).

8.4.5.4. Fast ultrasound imaging system in sodium

Fast ultrasound imaging in sodium is an important part of in-service inspection. It was decided to design an ultrasound imaging system capable to use the information provided by the backscattering due to rough metallic surfaces. This system, based on an orthogonal imaging concept, comprises two linear arrays of 128 elements disposed orthogonally and operating at a centre frequency of 1.6 MHz.

Ultrasound images of metallic structures similar to those found in LMRs were produced using an experimental configuration immersed in water. In fact in this experiment the receiving array was simulated with a transducer corresponding to a three element array block in which only the central element is active. This transducer was moved at each of the 128 scanning positions (corresponding to the 128 array elements). The signal are processed to reproduce the reception beam. The emission was obtained with a large monoelement transmitting antenna generating a focusing line at a typical distance of 2 meters.

Fig. 8.5. shows example of results obtained with 700 mm targets placed at 2 meters from antennas.

Fig. 8.5.a. was obtained with a flat SS plate with a surface considered as almost perfectly smooth.

Fig 8.5.b. was obtained with a flat SS plate (1,5 cm thick) with an average roughness of 10 μm representative of FBR internal structure.

Fig 8.5.c. was obtained with a cylindrical surface (1.5 cm thick) with a radius equal to 1 m, also with average roughness of 10 μm.

Details on this system are described in a paper presented at IEEE conference on ultrasonics, Nov. 96, San Antonio, USA.

8.4.5.5. Above - sodium repair

- Cuttings with a YAG 1 kW pulse laser were performed in one cut on steel sheets up to 50 mm thick. High quality cuts were obtained but the recovery of debris is difficulties (100 g/m). In order to avoid this problem a multicut process is being tested. This multicut process raises difficulties because laser reflections after the first passes limit the cut to 10 mm. Up to now the best result was obtained when the last cut has a residual thickness of 1 mm after realizing a grove of 9 mm depth. In these conditions the mass of debris was 10 g/m.

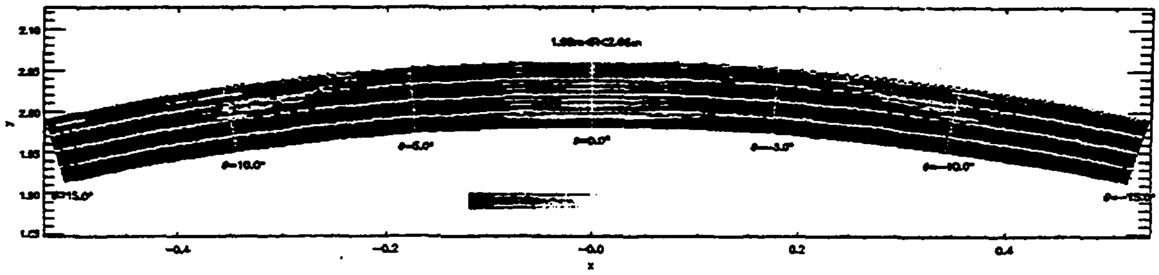


Fig 8.5.a. Echography of a smooth flat surface, for an elevation angle $\varphi = 0$

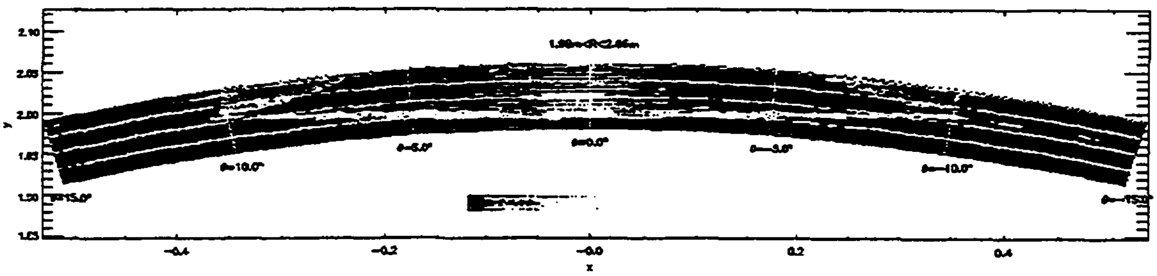


Fig. 8.5.b. Echography of a flat surface with roughness typical of LMR, for an elevation angle $\varphi = 0$

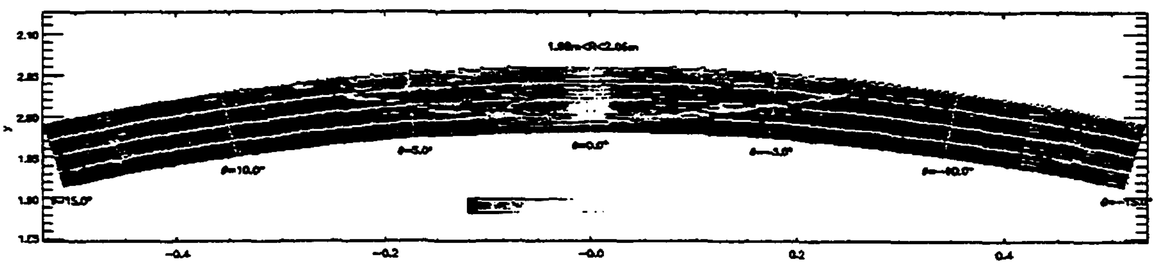


Fig. 8.5.c. Echography of a cylindrical surface with roughness typical of LMR for an elevation angle $\varphi = 0$

- In order to carry on the studies on the laser process to clean sodium-wetted surfaces a 40 MW peak pulse (10 ns) laser is being installed in the PENELOPE tank. The beam is transported through four optic fibers 15 m long (1 mm diameter).

- Tools are being set-up in the MIRSA facility [ref. 2] for tests to be performed in 1997.

8.5. Plant structural materials, mechanics, dynamics

8.5.1. Plant structural materials

8.5.1.1. Alloy 800

. SUPERPHENIX steam generators Alloy 800.

Due to its exceptionally high temperature corrosion and creep resistance, Alloy 800 is used for steam generator tubing in SUPERPHENIX.

A long series of corrosion studies in various conditions (high temperature water with and without hydrogen and with and without cloride pollution, high temperature caustic solution at various concentrations (0.01 %, 0.1 %, 1 %, 10 % NaOH solution), high steam temperature) were achieved on tubes with and without butt welds using slow strain tests and microautoclave capsules. These studies confirm the choice of Alloy 800 and give an upper limit for admissible soda concentration.

- Crack initiation and growth under creep and fatigue loading for Incoloy 800 alloy.

In order to complete the fatigue crack growth data obtained on a simulated heat affected zone (HAZ) a special planar weldment test sample was designed in order to study crack growth in a representative HAZ.

8.5.1.2. Superphenix Primary vessel stainless steel

A series of creep fatigue tests at 600°C and low stain rate on Z2 CND 17-12 steel (316 L SPX) has been achieved. Based on these results and results of previous tests a method was proposed to evaluate the behaviour of this material under these conditions. Similar work at 550°C is in progress.

8.5.1.3. Effect of thermal ageing

- Superphenix primary vessel OKR3U (19 Cr - 12 Ni - 2 Mo) weld metal

The effect of thermal ageing on the tensile properties on weld metal for use with type 316 steels is in progress.

Samples with exposure up to 90 000 hours at 550°C were obtained from PHENIX components. These samples consist of OKR3U metal, solution annealed or not.

Solution annealed and ageing OKR3U weld metals have a loss in elastic limit (around 150 Mpa) but an increase of ductility (around 15 %).

Creep data as well as impact strength data were obtained. They show the good behaviour of ageing weld OKR3U metal.

Microstructure examination (TEM) are in progress to explain these behaviours.

- Ageing material properties (elastic, creep, fatigue) of 304 SS and 321 SS were obtained from samples exposed up to 100000 hours at 350° and 550°C.

8.5.2. Mechanics

8.5.2.1. Design Rules

⇒ Rules for Design and Construction of LMR in France

In 1978, the CEA, EDF and NOVATOME decided to build and set-up a design and construction code comparable to that being drawn up at the same time for PWRs.

The "Regles de Conception et de Construction des Materiels Mecaniques des ilôts nucléaires RNR" (RCC-MR) was published in June 1985 and addenda were issued in 1987.

The second edition has been issued in 1993. The main modifications included in this second version will be described at the SMIRT 97.

The main results obtained in 1996 covered criterion documents on design rules (description and validation) :

- approval by an ad-hoc Committee of a criteria document on ratcheting.
- finalization of a criteria document for fatigue and creep fatigue rules and finalization of a document on acceptable limits for primary loads at high temperatures, both for approval by the ad-hoc Committee.

Other development were :

- finalization of a document (Appendix 3) on material data (including for example extended 316L (N) data base of more 2500 tensile tests and 700 creep tests) for approval by the ad-hoc committee.
- Progress on the implementation of the recent developments of European standardization in material specifications.
- Improvement of Appendix 16 on defect assessment is under review. Details will be given in a paper to be presented at ICÔNE 5, (5^{th} International Conference on Nuclear Engineering, May 26-30, 1997, NICE, France).

⇒ Ratcheting

Start-up and load variations in LMR cause sodium level variations in the hot plenum which load shells emerging in the free level area. Shell loadings are mainly due to the axial thermal gradient, since primary stress loading are generally low or insignificant, the same for wall thermal gradient.

Repeated cycling may cause different type of damages : ratcheting, creep-fatigue, buckling.

In 1996 the results of the VINIL tests and the BITUBE-INSA tests were analysed. The VINIL tests (ref. 1) consisted in producing ratcheting at the free level of sodium of large cylindrical shells with a ratio of radius versus thickness of around 300. The BITUBE-INSA tests consisted in two concentric tubes, tensily binded and submited to an axial tension. The external tube is heated. Consequently the internal tube is submited to an imposed displacement which superposes to the loading resulting from the axial tension.

From the analyses of these tests a modification of the ratcheting laws in the case of very low primary stress loadings was proposed. Furthemore a method based on elastic calculation was also proposed. This method is currently being validated on complex structrures and loadings.

⇒ A method to produce wall thermal gradient was tested. A pressurised tube is heated at 450°C then cooled by water. Thermal gradient 60°/mm were obtained in the 3 mm tube thickness. These pre-tests are necessary in order to control experiments on internally pressurised tube with thermal gradient through the wall.

⇒ There is still an effort in order to improve piping elastic calculations. A work on mechanical behaviour of a T junction or a nozzle is in progress with the development of a limit load analysis method and the setting up of the system of elastic problem equations. The resolution of this system in order to obtain an analytical solution raises some difficulties.

⇒ Thermal stripping.

The purpose of this work is to propose a rule less conservative than the one used at present : a maximum allowable temperature range ΔT around 40°C. A new interpretation of previous experimental results (AEA tests : SOMITE and SUPERSOMITE) was carried out. This analysis shows the possibility to use a less conservative method providing that the temperature variations at the surface of the structure be available. The validation of this method will need complementary experiments.

8.5.2.2. Mechanics of damage and failure

The purpose of the work is to develop and validate methods to analyse crack initiation and growth at high temperature for LMR components.

In 1996 three experimental programme are is progress (PROFIS, TUBFIS, TERFIS) as well as corresponding analyses :

- Creep fatigue crack growth on CT25 specimens in a 316 L (N) stainless steel at 650°C (PROFIS tests). Creep fatigue crack growth data were obtained at 650°C using compact tension specimens CT subjected to trapezoidal loading waveforms with holdtimes ranging from ½ h to 24 hours. All crack growth tests were performed using standard CT specimens with a width W = 50, a crack length ratio $a/W \geq 0.55$ and a thickness B = 25 mm.

Experimental crack growth rates, da/dN, are modelled using a linear summation of cyclic and creep contributions. A good agreement is obtained using a fracture mechanics creep parameter $C^*s(t)$, estimated from a reference stress approach, which continously takes into account the effects of primary and secondary creep during the tests. Detailed results will be presented at the SMIRT 14 conference (LYON, August 17-22, 1997).

- Crack growth data at high temperature (650°C) were obtained on tubes in bending conditions and cycling conditions with holdtimes of several hours (TUBFIS programme). The tests were performed on the base metal or on sample with a circumferential butt weld in the middle of the tube.

- TERFIS programme

This programme was presented last year (ref. 2). The purpose of TERFIS 5 tests carried out in 1996 was to quantify the influence of residual stresses on crack initiation and growth in weldments. They were obtained on stainless steel 316 L tubes where a defect was machined in a welded joint. The tubes were submitted to a low uniform tension (29 Mpa) and to thermal stocks (ex : in the case of a circunferential defect ΔT in the thickness 180°C and 10 cycles ; in the case of a semi-elliptic defect ΔT in the thickness 300°C and 1500 cycles).

Interpretation of the data are in progress. It seems that for the low primary stresses the influence of residual stresses is small because of large cycling strain conditions. This work will be presented at the SMIRT 14 Conference. It is planned to complete the TERFIS 5 tests with lower strain conditions.

- The use of limit loads of structures containing defects is widely spread in flaw assessment. In the case of an axially cracked pipe, tension and bending do not contribute to the elastic opening of the crack lips, but increase the level of plasticity, and a safe J estimation requires a full expression of the limit load, including all applied loadings, even when they have no influence on the elastic J estimation. It was proposed a lower bound estimation method for the global collapse of an axially cracked cylinder, based on the shell Von Mises yield criterion, under pressure tension and bending.

8.5.2.3. Fatigue damage assessment in weldments

Weldments are the preferential sites of crack initiation in elevated temperature structures. In the RCC-MR code (§ 8.5.2.1), when the design is based on an elastic analysis this behaviour is taken into account by use of admissible stresses specific to the joints. These admissible stresses of welded joints depend on the quality of the weld and its mechanical characteristics. They are obtained by multiplying the allowable stresses in the weld by an efficiency coefficient which takes account of the extent of the non-destructive examination performed. The allowable stresses for the weld are deduced from those of the base metal by factoring using the coefficients J_f and J_r for fatigue and creep damage respectively. Experimentally, J_f can depend on the number of cycles and is defined as the ratio of the strain range for base metal failure at n cycles divided by the strain range for weldment sample at n cycles.

With this objective, flat plate specimens have been cycled at 550°C and 600°C in bending at AEA using displacement control (SOUFLE tests), and thin cylinder - thick cylinder junctions have been cycled at 600°C to combined thermal and mechanical loads at CEA (SOFA experiments). Results of these experiments will be presented at the SMIRT 14 Conference.

8.5.2.4. The Leak Before Break concept (L.B.B)

In order to validate the extension of the Leak Before Break concept to very long internal surface cracks, an experimental data base is made with CHARFIS program (Fig. 8.6) [ref 2]. The tubular specimen has an internal axisymetric surface crack located in a weldment, which initiate and grow under a creep-fatigue loading. Last year two more tests were made or in progress, CHARFIS 4 and CHARFIS 5 (six tests are planned for the whole programme). The influence of a thermal gradient load in the thickness has been tested with the fourth campaign. For interpretations a Line Spring Finite Element approach was used.

Test	*Depth of defect*	*Bending Mpa*	*Tension MPa*	*thermal load*	*Cycles when breakthrough*
CHARFIS 1	0.5 x thickness	120	10	no	32
CHARFIS 2	0.5 x thickness	90	10	no	224
CHARFIS 3	0.5 x thickness	90	10	no	386
CHARFIS 4	0.5 x thickness	90	10	yes**	170
CHARFIS 5	3.8 x thickness	110	10	no	

*thickness of tubes : 8 mm

** cold shock in air, initial temperature 600°C.

Flowrates after breakthrough were measured in water tests.

8.5.3. Dynamics

8.5.3.1. Seismic response of LMR cores

- The SYMPHONIE experimental programme was presented last year with some details (ref. 2). After modification of neutronic shield elements spikes a new series of tests on the three row configurations were carried out. Tests were performed in air or in water and in the free-standing core or barrel restrained core. Part of this programme is performed in collaboration with Japan.

It was shown that the non-linear beam model gives good results regarding displacements and impact forces but some improvements are necessary regarding variation of the diameter of the core.

- Seismic response of the main reactor vessel and internal. 3D effects are being studied taking into account fluid-structure interactions. Comparison with 2D calculations are in progress.

8.5.3.2. Primary containment response to a mechanical energy release.

The PLEXUS code is used to calculate dynamics mechanics structural response due to various accidents such as sodium-water reaction or mechanical energy release following a core melt-accident.

These last years many calculations were performed with PLEXUS to study the primary containment response of EFR in case of a mechanical energy release.

In 1996 in order to have a better modelization of internal structures such as primary pumps, IHX, ... theoretical studies on a porous model in Eulerian formulation was achieved. This model was implemented in the PLEXUS code.

Additional development were achieved to take into account "thin" structures with apertures (i.e lateral neutronic shields) : flow continuity, effort on the structure.

Moreover dynamics loads data at high temperature on 316 L stainless-steel were obtained and documented.

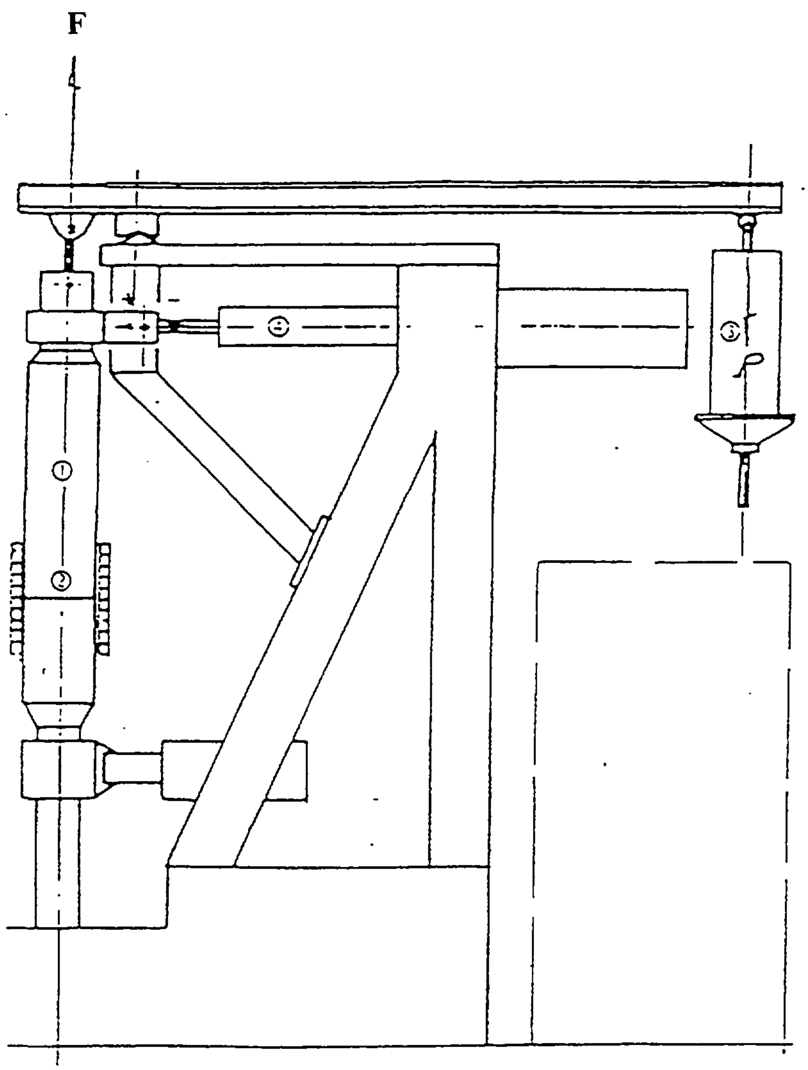


Fig. 8.6. CHARFIS Testing bench

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STATUS OF FAST REACTOR RESEARCH IN GERMANY

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Abstract

The paper gives a short survey of fast reactor activities in Germany. The fast reactor activities of FZK are part of the Nuclear Safety Projects. The R&D program include neutron physical and safety calculated, and post-irradiated examination of structural materials. The key issues and tasks of the program concerned safety and transmission of minor activities and fission products.

I. Nuclear Energy in Germany

The general situation of Nuclear Energy in Germany remained unchanged in 1996 and can be characterized as follows:

- Electricity production of about 160 TWh/a, i.e. about 1/3 of the total production
- Extremely good availability of running power stations despite hostile political environment
- One power station (Mülheim-Kärlich) is still „sitting on the ground". The state of Rheinland-Pfalz was charged to pay compensation to the owner. The state has lodged an appeal.
- Strong opposition against transport of spent fuel to the intermediate storage of Gorleben. Request of Niedersachsen (where Gorleben is located) to have additional intermediate storages in southern part of Germany. Objection by states in southern Germany
- New attempt to start consensus talks on
  - spent fuel transportation
  - operation of existing plants
  - possible construction of one new plant
- consensus failed
- Funding of German utilities for further development of EPR assured for the next phase (optimisation of Basic Design).

II. Fast Reactor activities at FZK

II.1 Neutron physical calculations of selected core configurations for burner reactors (CAPRA type)
During the determination of the neutronic input data for safety related SAS4A studies systematic deviations in the power distribution for the CAPRA reference core were recognised. The detailed investigations showed that for the more complex types of CAPRA cores more refined calculational treatment is required, i.e. either a fully 3-dimensional diffusion method or even the VARIANT transport method. In the latter case, the considerable influence on control rod worths and on the sodium void effect - especially for configurations with control rods inserted (i.e. not fully withdrawn) - must be studied.

Reactor physic studies for the transmutation of Am in separate subassemblies of a LMFR were made using a rather high Am-concentration (to restrict the number of the special assemblies), a typical fast reactor flux (3.61589 x 10^{15} K/cm^{2}s) and depleted UO_2 as a „matrix material". The calculations were done as fundamental mode burnup calculations with the KAPROS code system. The results show that for an irradiation cycle of 6 years the total amount of Am is halved, ¼ of the original mass was fissioned and the remaining ¼ transmuted mainly to Pu 238, Pu 242 and to Cm (242 and 244) and only a small percentage of the depleted UO_2 was fissioned. The power densities for varying Am and U contents could be adjusted between 500 W/cm^3 (pure UO_2) and about 1200 W/cm^3 (pure Am oxide). Also the study of Am multirecycling confirmed these results and based on that the possibility to transmute Am in the central part of a LMFR offers the chance to reduce the long-term risk of this element.

II.2 Safety-oriented studies of burner cores

Compared to the conventional cores the core melt-down behavior and the associated recriticality risk of burner cores are influenced by the specific burner properties, i.e. higher Pu- and MA-enrichment, lower fuel mass, no axial and radial blanket, etc. First calculations of the initiation phase of an oxide core were performed with the newest version of the SAS4A code and the unprotected loss-of-flow accident (ULOF) has been chosen as a representative initiator. The results show that energy releases as consequence of the initiation phase of ULOF transients in the CAPRA reference core design and their mechanical load potential remain rather small. Most probably consequences can be contained within the primary containment.
Analyses to assess the recriticality and core-melt energetic potentials for the CAPRA reference oxide and for the U-free core have shown that an enhanced fuel discharge from the core is required to prevent the formation of a neutronically active whole core pool with a high energetic potential. To study the potential for fuel removal the SIMMER III code has been applied as the main calculational tool and it could be shown that the heterogeneous structure of the CAPRA core as well as the diluent subassembly system might be optimized for a maximum potential of fuel release and by that decreasing criticality and reducing the energetic levels.

II.3 Post-irradiation examinations of cladding and wrapper materials

To determine swelling and in-pile creep of FR cladding material a pressurised tube experiment was carried out in the PFR reactor. Samples of various alloys (model plain Fe-15Cr-15Ni, commercial German 1.4970, etc.) were tested in the temperature range between 420°C and 600°C and doses reached between 61 and 106 dpaNRT. After shut-down of the PFR the samples were transported to FZK for PIE. The non-destructive measurements of diameter and length show the large effect of minor alloying elements upon swelling resistance; most effective is the addition to Si.

These measurements show also the correlation between swelling and in-pile creep according to the theory of stress-induced preferential absorption (SIPA) and the creep model proposed by Gittus.

Since the length increase of the pressurized tubes is due only to swelling, these measurements are used to determine the stress-induced swelling. Problems arise by the influence of the endplugs and the heat affected zone, so that density measurements at the end of the irradiation are necessary to confirm the length measurements. The density measurements of a series of different lots of the DIN 1.4970 are finished. The stress-induced swelling measured by the two methods agree rather good. The values of the linear stress-induced swelling are approximately 0,7% at 420°C, 80 dpa_{NRT} and 120 MPa hoop stress. It should be noted that the different lots show similar stress-induced swelling and that there is only a very weak temperature influence.

To determine the influence of stress on the microstructure, TEM investigations are in progress. First results on the average void diameter and the void concentration seem to show that the stress-induced swelling is a result of the accelerated void nucleation rather than the accelerated void growth. But this result needs further confirmation.

The post-irradiation examinations of a wrapper made of martensitic X18CrMoVNb12 1 steel were performed within the EFR (European Fast Breeder Reactor) project in collaboration with the Centre d'Etudes Nucléaires de Cadarache and members of the German-Belgian AGT 1 working group. The examinations were aimed at inserting into the French Phénix reactor a wrapper made in Germany and checking the latter for impacts of irradiation on the service life and integrity. Complex mechanical tests and structural analyses were carried out in order to determine the material behavior following two years of irradiation at the Phénix reactor. During 719 EFPD (Equivalent Full Power Days), a maximum dose of 105 dpa_{NRT} was accumulated in the center of the wrapper. The irradiation temperature varied from 380°C entry temperature of the sodium at the foot of the wrapper up to the top of the wrapper with about 630°C. Particular attention was paid to the welds connecting the martensitic wrapper material with the austenitic (AISI 316) top and bottom sleeves. Another area of importance was that of the highest dose in the center of the wrapper.

The investigations of the wrapper showed the material behavior after non-destructive tests, tensile, creep and charpy-V-notched impact tests. All these tests were performed with base material as well as with welds, at room temperature and at the different reactor temperatures.
Compared to the unirradiated wrapper, no macroscopic changes in dimension could be found, a deformation of the component was not observed. By means of transmission electron microscopy, the effects of neutron irradiation on the material could be made visible. In the temperature range of 380 up to about 410°C, irradiation-induced coherent α'-precipitates were formed in the martensitic laths. Void formation took place in the range of the highest dose at temperatures from 400 up to 480°C. Counting of the pores yielded a maximum swelling of 0.05% in the high-dose range and, hence, confirmed the good swelling behavior of the material. At higher temperatures (>430°C) He bubbles were distributed homogeneously in the matrix with their density increased considerably.

The irradiation induced microstructural changes did not strongly effect the mechanical properties of the wrapper material. Radiation hardening by coherent α'-precipitates was observed in the temperature range of 380 up to 430°C. The strength of the material with the highest dose (480°C) did not change compared with the unirradiated material. But the welded material showed at 380°C an increase and at the top, at 630°C a loss of the tensile strength combined with a loss of elongation, too. In the welded samples the material separation under tensile tests took place at 380°C in the melting zone and at 600°C between the martensitic base material and the martensitic heat-affected zone.
Creep tests of the base wrapper material were performed at 550 and 600°C. The results of base material, of welds and irradiated material are in the same scatter band.
The impact tests of the wrapper material after irradiation showed a small decrease of the upper shelf energy. The DBTT (Ductile to Brittle Transition Temperature) was hardly influenced.

Summarizing, the martensitic 10% Cr-steel showed good mechanical properties as wrapper material, especially in the regions of high doses. The mechanical properties and the fracture behavior will be discussed by the microstructural changes.

II.4 Irradiation experiments in HFR (TRABANT)

In the frame of the international CAPRA programme new fuel types are under discussion, either based on a very high Pu content (>40 at %) in the mixed oxide type or considering U-free fuels (PuO_2 + inert oxides or Pu N). In the irradiation experiment TRABANT informations for these new fuels are to be received with respect to thermodynamic properties, compatibility and solubility behavior.
The first experiment was started in 1995 and consisted of 3 pins with different fuel columns, namely with $(U_{0.55}Pu_{0.45})O_2$, $(U_{0.55}Pu_{0.40}Np_{0.05})O_2$ and PuO_2/CeO_2).

The pins were irradiated in a TRIOX capsule of the HFR and two pin failures occurred at burnup levels of 6 at % (PuO_2/CeO_2 pin) and at 10 at% (pin with 45% atPuO_2). At present only results of the non-destructive examinations are available which show the existence of molten fuel in the case of the U-free pin (PuO_2-CeO_2) and fuel dislocation and temperature increase in the upper part of the pin with the high Pu content. The destructive examination is planned in the second half of 1997.

For the continuation of the TRABANT series it is planned to irradiate again mixed oxide pins with a high Pu-content which will be fabricated by different procedures (mechanically mixed or by a SOL-Gel route).

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STATUS OF FAST REACTOR DEVELOPMENT IN INDIA (April 1996 – March 1997)

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Abstract

India generated 395 TWh of electricity during April 1996 to March 1997. Oil import bill during the year wss $ 9.3 billion. The operating performance of the thermal power reactors has considerably improved during the year and has enhanced the confidence level in nuclear energy in the government and the public. Construction of 4x220 MWe PHWR is continued at two locations. Start of construction of 2x220 MWe PHWR, 2x500 MWe PHWR and 2x1000 MWe VVER (Russian collaboration) and 500 MWe PFBR have been proposed in the IX Plan (1997-2002). The 13 party coalition government is discussing the IX plan proposals in the power sector.

Operation of FBTR at 10.5 MWt is continued. The maximum fuel burnup reached is 32,000 MWd/t without any failure. Targeted burnup is 50,000 MWd/t. Post irradiation examination has been completed on one fuel subassembly taken out at 25,000 MWd/t. The performance of the fuel is very good. Turbine was rolled upto synchronous speed of 3000 rpm several times during the year and operation was found to be smooth. TG synchronisation with grid will be achieved during the reactor operation at 12.5 MWt, with the addition of fuel subassemblies in the core.

All the activities related to the revision of conceptual design from 4 loop to 2 loop concept are almost complete for the 500 MWe Prototype Fast Breeder Reactor. The main options for the reactor are sodium coolant, pool type, MOX fuel, 2 primary sodium pumps, 2 secondary loops with 4 SG in each loop. The important design activities carried out during the year are plant dynamic studies, decay heat removal analysis ,design of pump to grid plate pipe, scram and LOR parameters, location of secondary sodium pump in the secondary sodium circuit and design of fuel handling machines.

R&D in the domain of engineering development, thermal hydraulics, structural mechanics, metallurgy, non-destructive examinations, chemistry and safety are continued. Important experimental R&D works carried out during the year were testing of prototype primary sodium pump in water, operation of a large sodium test rig to study the heat and mass transfer in the cover gas, testing of dummy fuel subassembly in water test rig for pressure drop and vibration measurements.

1.0 INTRODUCTION

1.1 Energy Scenario

The economic reforms initiated since 1991 in India have resulted in GDP growth rate of 6.8%. It is being increasingly realised that this growth rate cannot be sustained unless the infrastructure is expanded correspondingly. There is a feeling now that the Government's over reliance on the private sector participation in the power sector has misfired due to various reasons. The slackening of public sector investment during the eighth plan (1992-97) on electricity generation capacity addition, resulted in poor realisation of the target. Against the target of 31 GWe, only 15.5 GWe could be realised. Annual addition of capacity has been dwindling with an addition of 1.6 GWe in 1996-97 against 2 GWe in 1995-96. Large scale power cuts have been imposed in many states. Capacity additions upto 10 GWe in public and private sectors are expected in a short period and most of them will be liquid fuel based. This again raises issues of energy security as the fuel has to be imported. A comprehensive national policy on the energy front is yet to emerge. The target for the capacity additions during the ninth plan (1997-2002) is 40 GWe. The public sector National Thermal Power Corporation(NTPC) has been performing extremely well, meeting about 25% of national electricity demand from its thermal power stations. The corporation has plans to add 6.2 GWe during the ninth plan period in addition to the projects totalling 1.8 GWe underway. The major problem faced by public sector utilities is the policy of subsidies given to various user sectors resulting in bankruptcy and paucity of internal resource generation. There has been general increase in the electricity tariff to raise the resources.

Total electricity generation capacity was 84.9 GWe at the end of March 1997. Out of this 61.2 GWe is thermal, 21.5 GWe is Hydro and 2.2 GWe is nuclear. Total electricity generated by the utilities during the year was 395 TWh. This is about 4% more than the generation during the previous year. Sector-wise electricity generation during 1996-97 is as given below in TWh

Thermal	-	316.85	
coal	-		265.50
lignite	-		013.35
gas	-		027.80
oil	-		001.68
multifuel	-		008.52
Hydro	-	068.61	
Nuclear	-	009.01	
Total	-	394.47	

The installed wind power capacity connected to the grid, is 800 MWe

1.2 Economic Outlook

Economic reforms initiated in 1991 in India continued in 1996-97 in spite of the change in the political setup indicating that the policy of globalisation has come to stay in the country under the general political consensus. The performance in the foreign trade front was slightly discouraging with exports rising only by 8% to $ 35 billion. There was a decline in imports too resulting in a trade deficit of $5.3 billion. The total value of oil imports increased to $ 9.3 billion. The situation in foreign exchange reserves continued to be bright with reserves touching $ 22.36 billion. Though India is a developing country, as per World Bank data, it is the fifth largest economy in the world ($ 1318 trillion) after USA, China, Japan and Germany, when purchasing power parity is considered. Budgetary allocation by the Government for R & D and professional education has been increased to $ 2.5 billion in the year 1997-98, which is 14% higher than in 1996-97. This allocation is roughly 4% of the total expenditure by the Government.

1.3 Nuclear Energy

The performance of nuclear power plants has improved considerably over the last two years. The operating performance of the eight plants out of the 10 existing plants is tabulated below. Total electricity generated by nuclear power in the year 1996-97 is 14% more than in the previous year though the capacity has not increased. An interministrial committee has carried out a comparative cost study between nuclear and coal, and it is found that the nuclear energy is cheaper if the location is 900 km from the coal pit. This difference improves in favour of nuclear energy for the life time cost.

TABLE I

Operating Performance (1996-97)

Unit	Generation - 10^6 kWh		Availability Factor (%)	Capacity (%)
	Target	Actual		
TAPS -1	665	424 *	38	30
TAPS -2	975	648 **	59	46
MAPS -1	500	752 ***	53	51
MAPS -2	810	1233	84	83
NAPS -1	1209	1377	75	71
NAPS -2	1218	1449	78	75
KAPS-1	1201	1593	90	83
KAPS-2	992	1593	90	83
Total	7570	9069		67

* TAPS-1 was under annual shutdown from July 1996 to mid January 1997 for refuelling, carrying out mandatory regulatory inspection of core shroud and other systems.

** TAPS-2 started generation from June 1996 after annual shutdown from September 1995 to May 1996 for refuelling, carrying out mandatory regulatory inspection of core shroud and other systems.

*** MAPS-1 started generation from September 1996, after annual shutdown from April 1995 to August 1996 for mandatory regulatory inspection of coolant channel.

The two plants that are not included in the above table are the twin reactors of Rajasthan Atomic Power Station(RAPS). RAPS-1 has been brought on stream recently after prolonged shutdown for repairs. RAPS-2 is undergoing coolant channel replacement. Removal of existing channels has been completed ahead of schedule and retubing is being taken up.

It may be seen that the performance has exceeded the targets set for six plants. This excellent performance of nuclear plants has given rise to renewed enthusiasm from the Government to go in a big way for nuclear power. Construction of 2x220 MWe at 2 sites is in progress and will be commissioned in 1999-2000. Projects consisting of 2 x500 MWe PHWR, 2x220 MWe PHWR and 500 MWe PFBR are proposed to be started in the ninth plan period. In addition to these, proposal of constructing 2x1000 MWe Russian VVER are expected to be finalised during this period. Recent visit to Russia by India's prime minister has speeded up the proposal. The 13 party coalition government is discussing the IX plan proposals in the power sector. The target set at present is 20 GWe by the year 2020. An industrial forum called Indian Atomic Industrial Forum(IAIF) has been launched in October 1996, consisting of major Indian industries to serve as a common platform for the industries, institutions etc., for exchanging information and views in relation to the nuclear technology, with an ultimate aim to promote the cause of nuclear power.

Six MOX fuel subassemblies have been loaded in the Tarapur BWR, demonstrating the feasibility of the fuel substitution. Good experience in the fabrication of MOX fuel has also been acquired. Another important target achieved was cold commissioning of the 100 tpa capacity Kalpakkam Reprocessing Plant. This plant will meet the fuel requirement of the 500 MWe PFBR proposed to be built at Kalpakkam. The construction of PFBR is expected to commence in 1999 for which work on the preparation of PSAR has been started at the Indira Gandhi Centre for Atomic Research. IGCAR has a scientific and technical manpower of about 1500 and budget allocation for the year 1997-98 for IGCAR is Rs. 800 million ($ 22.2 million).

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 turbine generator (TG) is not available. The reactor achieved its first criticality in October 1985. Since then it has been operated at various power levels in stages upto 10.5 MWt. The small carbide core with 26 fuel subassemblies has been licensed to operate upto 10.5 MWt with 320 W/cm peak linear heat rating of fuel. All the works in TG system have been completed and the turbine has been rolled upto its synchronous speed of 3000 rpm. In March 1997, clearance from safety authorities has been obtained for further loading of the fuel subassemblies and to operate the reactor upto a power level 12.5 MWt. Synchronisation of TG with grid will be done shortly.

2.1 Reactor Operation

During the year reactor operated for 2,550 h, out of which 1,900 h has been at high power (fig 1) and the following were achieved.

- Test fuel pins irradiation programme for Mark I and Mark II (55% PuC-45% UC) compositions was completed and discharged for PIE.
- Maximum fuel burnup of 30,390 MWd/t for Mark I (70% PuC-30% UC) core without any fuel failure was achieved. The central fuel subassembly with a maximum burnup of 25,030 MWd/t was discharged for post Irradiation Examination (PIE). PIE is mostly completed (see para 4.5.3).
- Irradiation campaign no.03 with 25 & 04 with 26 fuel subassembly core were completed.
- Rolling of turbine at 3000 rpm for ~ 100 h continuously was achieved and all electrical tests on generator were completed.
- No reactivity transient incident recurred during the reactor operation in the year

FBTR has logged 17,000 h of cumulative operating time, out of which 4,200 h has been at high power. During the year, reactor tripped 14 times (3 scrams and 11 LOR). In addition, there were 5 forced manual shutdowns and 5 controlled shutdowns. During the shutdown,

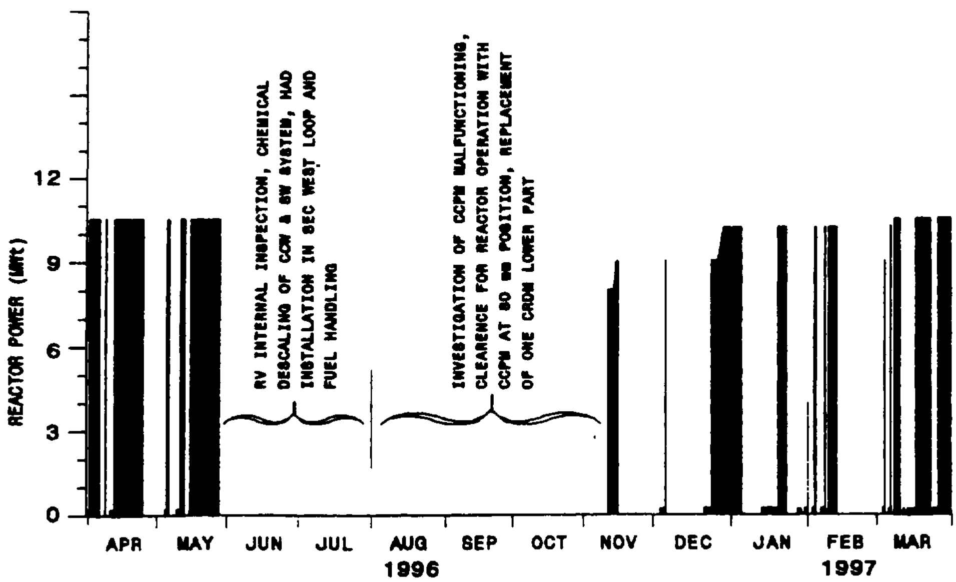


Fig 1 Histogram of reactor operation

reactor vessel internal inspection, ultrasonic testing of secondary sodium circuit weld joints, installation of hydrogen in argon detector and carbon meter circuit in secondary sodium loop, chemical descaling of condenser cooling water system, replacement of leaky(bellows failure) CRDM lower part, various modification works for the improved performance of different systems, major preventive maintenance works and surveillance activities were carried out. In October 1996, during exercising of one of the hydraulically operated SG isolation valves, a ferrule joint in the oil inlet line to the valve actuator failed. The leaking oil splashed & seeped through the insulation of hot sodium pipe and caught fire. After inspection of sodium lines, rectification & checking of oil circuit, the system was normalised.

After completion of fuel handling campaign in July 1996 for loading the 26th fuel subassembly, mobile plate of core cover plate mechanism (CCPM), housing core thermocouple sleeves, could not be lowered to its normal working position. Similar incident took place in July 1995 and normalcy was then restored by applying safe jacking down force. Normalcy could not be restored, with safe jacking down force, this time. Clearance was obtained to operate the reactor with CCPM at 80 mm position after reviewing the safety implications, arising due to dilution of temperature measurement at the outlet of fuel subassembly, by suitably modifying the trip thresholds. The core temperature data indicated an average dilution of outlet temperature for all the fuel SA by about 7% with respect to the measured values with CCPM at 15 mm normal position and the individual rings were found to retain their identities. Mock-up studies are being carried out to devise rectification methods to restore CCPM to normalcy.

The performance of reactor systems, sodium systems, control rod drive mechanisms and other safety related systems and auxiliary system were generally satisfactory. The primary and secondary sodium purity was maintained below the plugging temperature of 105° C. The four sodium pumps and their drives are operating well and have logged 83,293 h, 70, 615 h, 88,878 h & 69,201 h.

The heat transport parameters achieved during turbine rolling at 10.5 MWt operation in March 1997 are as follows:

- Reactor inlet/outlet temperatures : 334/420 deg C
- Primary sodium flow through core : 375 cu m/h
- Central fuel subassembly sodium outlet temperature : 474 deg C
- Sodium temperature at SG inlet/outlet : 417/287 deg C
- Secondary sodium flow (2 loops) : 280 cu m/h
- Feed water temperature : 188 deg C
- Feed water flow : 16 t/h
- Steam temperature : 398 deg C
- Steam pressure : 120 kg/sq cm

2.2 Evolution of Hydrogen in the Secondary Sodium Circuit

The SG were valved in with water on 21st January 1993 for the first time and were in service, operating at 10.5MWt, for 4,250 h till 31st January 1997. The hydrogen concentration in the secondary sodium with the SG in service is mainly a function of hydrogen diffusion rate, steam and sodium temperature. During the first valving in of water, the hydrogen concentration increased rapidly to a high value on the virgin surface of the SG tubes (without magnetite layer) due to high diffusion of hydrogen which liberated during the process of magnetite layer formation. This came down subsequently as the operating hours increased. It is also observed that the increase in background is comparatively less whenever the operation is followed after a controlled shutdown in the previous operation, as the magnetite layer formed was not disturbed due to the gradual decrease of temperature in SG.

In March 1997, an experiment was conducted with bypassing the cold trap in the secondary west loop with the reactor operating at 10.5 MWt to study the increase in hydride/oxide impurities and also to check the sensitivity of the SG leak detection system. On bypassing the cold trap, the hydrogen signal increased at a rate of 20 ppb/d till the cold trap was put back into service after 4 days. The hydrogen value reached is 160 ppb. Fig 2 shows the different hydrogen evolution profiles as recorded by the quaropole mass spectrometer over the period of SG operation.

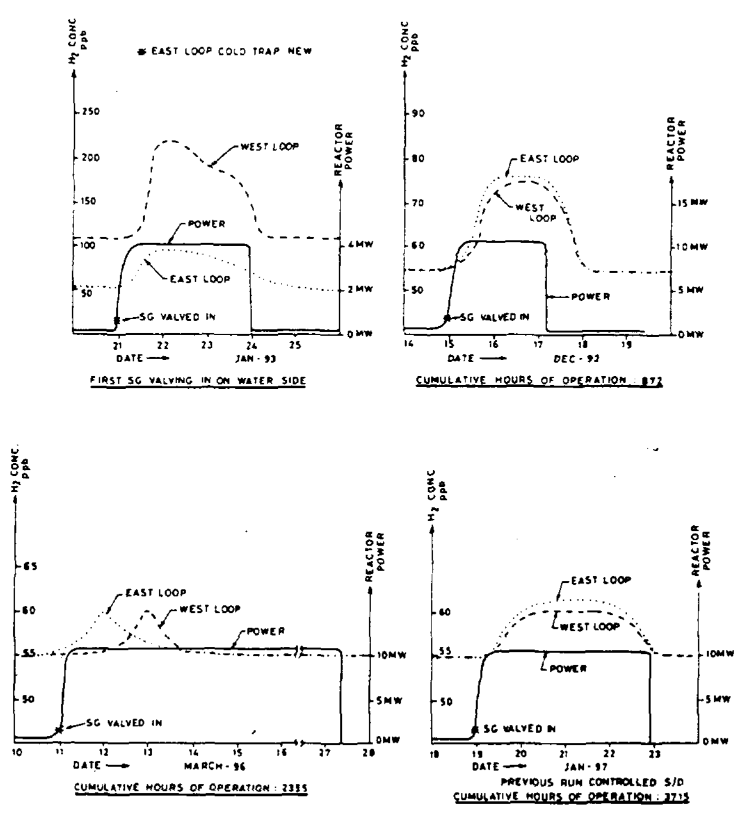


Fig 2 Hydrogen evolution during SG operation

2.3 TG Commissioning

During this period, TG was rolled several times. Of these, two trials were done for cleaning the system to prevent degradation of steam water chemistry. During other occasions, TG trips were due to the following reasons:

- LH CIES valve was stuck open
- Heavy sparking at excitor brushes
- Failure of trip reset valve during oil injection test
- Oil leak from front pedestal leading to a minor fire.
- LOR due to loss of suction in CEP.
- LOR due to DLP trip when CPU got choked.
- Feed water flow fluctuation
- On low vacuum in main condenser as auxiliary header pressure collapsed due to inadequate steam quantity

After modification works in steam-water circuit, such as diversion of steam trap drains to dump condenser, connection of LPFT recirculation line upstream of gland steam condenser, provision of spray cooling arrangement for exhaust hood of main condenser, revision of operating level of LPH - 1 and rectification of mechanical & electrical deficiencies in TG system, TG was rolled continuously for ~ 100 h at 3000 rpm in March 1997. Salient features of TG rolling trials were as follows:

• Turbine inlet steam pressure (kg/sq cm)	-	117
• Turbine inlet steam temp. (deg. C)	-	390
• Main condenser vacuum (mm Hg)	-	640
• Critical speed of turbine (rpm)	-	1700
• Governor takeover speed (rpm)	-	2750
• No load steam consumption (t/h)	-	4 - 6
• Max. eccentricity at critical speed (μm)	-	32
• Vibration on bearing No.3 (limit 50μm)	-	22
• Overall expansion (limit 9.2 mm) mm	-	4
• Differential expansion (-2.4 to +3.2 mm)	-	+ 1
• Coast down time from 3000 to 0 rpm (min)	-	32

All mechanical commissioning checks were completed. During the continuous run at 3000 rpm, all electrical tests including drying of alternator winding were performed.

2.4 Future Programme

Reactor will be operated upto the power level of 12.5 MWt to achieve the following

- TG synchronisation with the grid
- Maximum fuel burnup of 50,000 MWd/t
- Irradiation programme for Zr-Nb alloy for PHWR programme.

Development and fabrication of second generation neutronic channel has been completed and the channels will be received at site in May 1997. The first generation channel had noise pickup, obsolescence of components and maintenance problems. Hence, these will be replaced during a long shutdown of about 3 months. The UPS is also being replaced. A spare boiler feed water pump has been manufactured indigenously, as procuring spare parts for the imported pump was found to be difficult. Procurement of materials for the manufacture of spare SG is in progress.

The reliability analysis of the DHR systems was carried out and it is found that the non-availability of DHR for a mission time of 1 week is 1.1×10^{-7}.

Radioactivity release to atmosphere was 37.4 Ci and man-rem consumed was nil.

3.0 PROTOTYPE FAST BREEDER REACTOR - DESIGN

3.1 Conceptual Design

Prototype Fast Breeder Reactor (PFBR) is a 500 MWe pool type sodium cooled reactor. The 4-loop concept which was taken as the reference for the preliminary design and analysis during 1985-1992 has been reviewed thoroughly mainly from economic considerations. The outcome of these studies helped to arrive at a compact 2-loop concept which is expected to be the fore-runner of commercial FBR in the country. For the 2-loop concept, all the major conceptual issues have been finalised. The main options selected are given below: (fig 3)

• Reactor coolant	Sodium
• Primary circuit concept	Pool
• Thermal power	1250 MWt
• Electrical power	500 MWe
• Fuel	PuO_2-UO_2
• Core height /diameter	1 / 2 m
• Fuel pin dia / No of pins per SA	6.6 mm / 217
• Number of PSP	2
• Number of IHX	4
• Number of sec loops	2
• Number of SG per loop	4
• Number of TG	1
• Na temp at reactor inlet	670 K
• Na temp at reactor outlet	820 K
• Steam condition at SG outlet	763 K at 17 MPa
• In-vessel fuel handling	2 rotatable plugs + 1 TA
• Spent fuel storage	In water pond
• Reactor shutdown systems	2
• Decay heat removal systems	2
• Containment building	RCC cylindrical shape
• Reactor site	Kalpakkam
• Reactor life	30 y

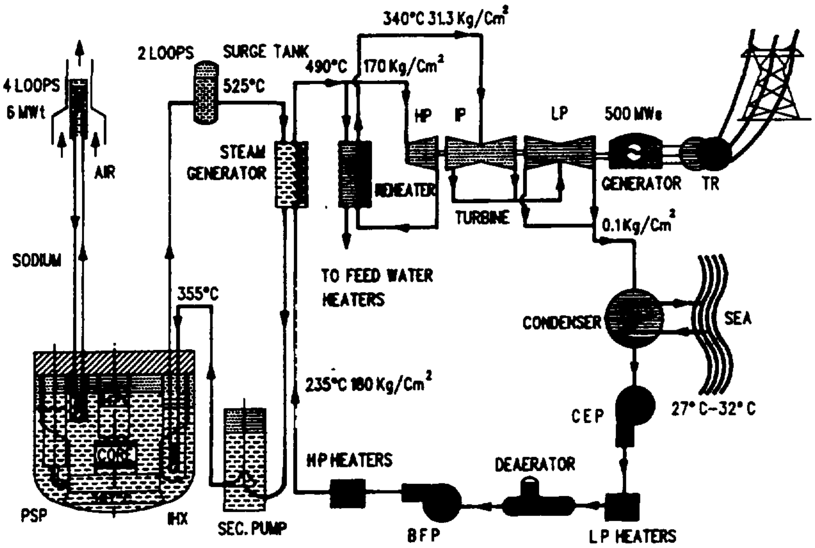


Fig 3 PFBR flow sheet

3.1.1 Core

The equilibrium core configuration, fresh subassembly and peak power distribution and linear power/burnup have been obtained using the newly developed in-house code 'FARCOB'. The static and dynamic reactivity coefficients under various core conditions are evaluated. The value of power coefficient is 0.462 pcm/MWt and there is 12 pcm decrease in reactivity for 10 % decrease in flow. The contribution of differential thermal expansion between control rod and main vessel to dynamic power coefficient is 0.125 pcm/MWt. Reactor physics aspects of the protected plant incidents analysis were completed, from which permissible time delay in initiation of safety action for different incidents was established. The design basis events that were considered in setting the limits on time delay are the rupture of pipe connecting pump to grid plate and total instantaneous blockage of fuel subassembly, resulting in melting of top one third fuel subassembly and subsequent falling under gravity. The shielding design of the core is reviewed to satisfy the new occupational dose criterion of ICRP-60. Streaming paths through roof slab are provided with supplementary shielding. Neutron flux at axial detector locations below main vessel is estimated using alternate methods of analysis. Tritium pathways and tritium release in the reactor is evaluated and the tritium release to the environment is found to be negligible. The decay power, n-source and fission product activity of discharged subassembly is evaluated from handling considerations. Inventories of fission product concentrations are generated for fuel subassembly as a function of burn-up.

Design specification and detailed drawings for core subassembly, control & safety rod drive mechanism (CSRDM) and diverse safety rod drive mechanism (DSRDM) have been completed. Design of DSR subassembly has also been completed. Modifications required in the core subassemblies to facilitate handling by transfer arm type IVTM were finalised and incorporated in the design. Flow zoning of core was finalised for the computed power data. Based on the parametric studies on the core restraint system using the code 'NUBOW-2D', it was found that the difference in the bowing behaviour between a single pad and two pad systems is negligible for 100,000 MWd/t burn-up. Hence a natural core restraint concept with a single contact pad at 150 mm above fissile column top is recommended.

3.1.2 Reactor Assembly

The bottom dished end shape of main vessel with a geometrical discontinuity at its junction to the cylindrical portion, has been finalised after detailed investigations on buckling under various operating conditions. Fluid-elastic instability analysis of the weir shell indicated that the weir shell is stable under all operating condition except during fuel handling operation which is being investigated.

Core support structure design with square grids at the central portion and radial stiffeners around the periphery has been recommended. Structural integrity assessment has demonstrated that even with the absence of the most critical radial stiffener, the reduction of rigidity of the structure is within the acceptable limits from reactivity insertion considerations.

Detailed creep-fatigue analysis was done for control plug, the most critical component in the reactor assembly and the analysis indicated that the maximum permissible reactor outlet temperature in order to satisfy the high temperature design criteria of RCC-MR (1993) is about 780 K as per `elastic route' and 830 K (very close to the operating temperature of 820 K) as per detailed viscoplastic analysis using Chaboche Viscoplastic Model. Hence it is decided to reduce the thermal load. Accordingly, it is proposed to reduce the primary and secondary sodium flows to 20% as a sympathetic safety actions following a reactor scram. With these, structural integrity assessment of control plug has been made which indicates that it is possible to satisfy the RCC-MR `elastic route' itself at a temperature of 820 K. The reduction of primary sodium flow is also essential to limit the hot shocks on the IHX bottom tubesheet and main vessel bottom dished end during one secondary sodium pump trip or seizure incidents.

The conceptual design of grid plate and primary pump discharge pipe have been completed. The primary sodium is pumped into the grid plate from each primary pump through 2 discharge pipes. The primary sodium pump discharge pipes which are critical for the 2-loop concept are analysed in detail both from plant dynamics, structural mechanics and consequences of pipe rupture considerations (fig 4). Although overall mechanical and thermal loadings are low, the pipe layout and thickness (15 mm) have been chosen so as to have minimum stress (<50% of design code allowable values), thus increasing its reliability. The layout is finalised to meet the contradictory requirements of higher flexibility in the horizontal direction for accommodating the thermal expansion of the pipe during hot shocks (~50 K on the metal wall) and adequate rigidity during seismic excitations. Detailed fracture analysis is also done with pessimistic assumptions on the crack size (crack length is 100 and depth is 4 mm) orienting at the highly stressed locations. Assessment made as per RCC-MR (1993) (Appendix 16) indicates that the propagation of crack is negligible under the load cycles involving plant shutdowns, hot shocks and seismic events. Sodium leakage is possible only after the component is subjected to more than 120 times of one plant life load cycles. With these arguments, it is worth considering that a guillotine rupture of one pipe can be taken out from the design (a level equivalent to LBB justification without leak being monitored). However, as a philosophy under this situation, wherein ISI is not possible, effect of one pipe ruputure has been analysed for the thermal hydraulics considerations. It results in a flow reduction to about 23 % in 2 s. Detailed analysis shows that this is detected by 2 diverse parameters (higher flow through unaffected pump and reactivity) and reactor is scrammed before acceptable limits on fuel integrity is reached.

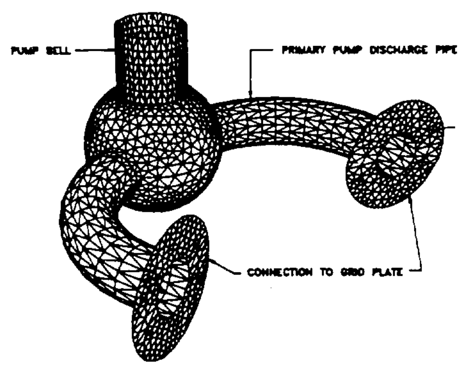


Fig 4 FE mesh of discharge pipe

At the IHX penetrations through the inner vessel, mechanical sealing arrangements are opted and conceptual design of the this sealing system is finalised. The thicknesses required for the tubesheets (120 mm), inner shell (16 mm) and inner shell of the outlet header (16 mm) have been determined for IHX, based on detailed structural optimisation studies at the junction of tubesheet and outer shell, effect of tubesheet thickness on the buckling behaviour of straight tubes (19 mm OD and 0.8 mm thick) and thermomechanical behaviour under various flow zoning options. Further, from these studies, option of 40% more flow in 6 outer rows has been recommended.

Design of top shield design cooling system, inflatable seals and support arrangement for rotatable plugs, vertical support system for reactor assembly, reactor vault and its thermal insulation are completed. Design validation studies for the mechanical seals at the IHX penetrations and inflatable seals at the rotatable plugs have been taken up. Based on the detailed structural optimisation studies, the stiffness arrangements for the roof slab have been decided.

3.1.3 Secondary Sodium Circuit

The primary sodium circuit consists of 2 centrifugal pumps without non-return valve (NRV). Each one is a single stage, top suction pump delivering a flow of 4.25 m^3/s at a head of 75 m, and an operating speed of 680 rpm. The pump is designed for a cavitation margin of 1.4 at an available NPSH of 15.24 m of sodium. An axial diffuser has been employed to restrict the maximum dimension of the removal hydraulic part to less than 1800 mm. Each pump is powered by a 3500 kW solid state AC variable speed drive with speed variation from 20% to 100% of normal speed. Each of the two secondary loops is provided with one pump which delivers 3.5 m^3/s at a head of 60 m of sodium and an operating speed of 1450 rpm.

For deciding the location of secondary sodium pump in the secondary circuit, three alternative concepts , viz.(i) pump in the cold leg at higher elevation with a surge tank in the hot leg as in FBTR, (ii) pump in the cold leg located at the lower elevation, incorporated with a piston ring seal in the overflow line to avoid flooding of the pump vessel with a surge tank at the hot leg and (iii) pump in the hot leg without surge tank and pump cover gas acts as cushion space, are being studied with reference to operational and mechanical design aspects (fig 5). Surge tank is essential for absorbing the pressure shocks on the component following a large sodium water reaction in SG. Locating the surge tank in the hot leg is favoured because it acts very effectively. Pump in the hot leg has less operational problem and a marginal cost saving with the absence of surge tank. But there is no operating experience (high operating experience

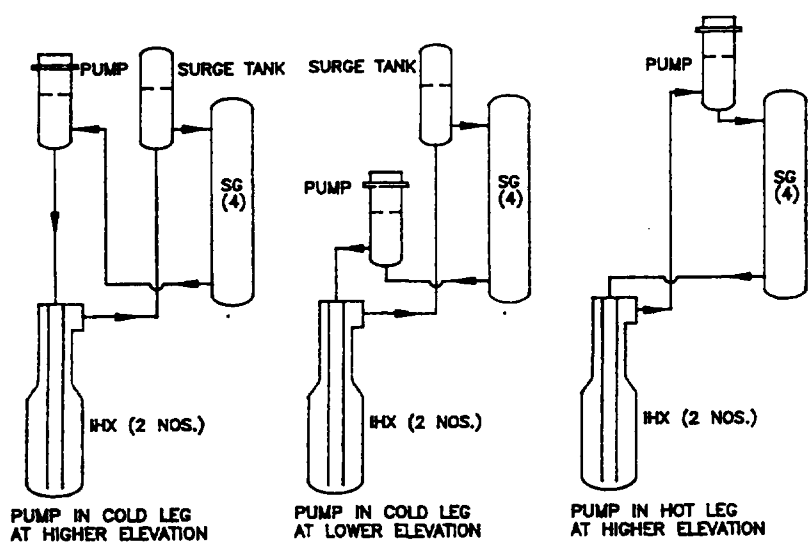


Fig 5 Options for locating secondary sodium pump

is available for the primary sodium pump). Pump in the cold leg at higher elevation option involves higher cost because of increased pipe length and sodium inventory. Hence pump in the cold leg at lower elevation is preferred. There are minor operational and control problems because of continuous operation of EM pump which can be handled easily.

Structural mechanics analysis indicates that the acceptable temperature difference between the primary temperatures from IHX outlets is about 40 K. Thermal hydraulics analysis indicates that with one module isolated in one loop, the temperature difference is 10 K at 90 % power. With 2 modules isolated, this temperature difference is 24 K at 67 % power. To achieve this, it is necessary to reduce the core flow (to 67 %) in proportion to the power as well as the sodium flow in the unaffected loop (50 %) as in the affected loop (50 %) The geometry of SG tubesheet junctions are profiled so as to minimise the basic tubesheet thickness and also to have improved thermomechanical behaviour under thermal shock during transients. Fig 6 shows the improved profile of the SG tubesheet junction with its header. The recommended tubesheet thickness is 145 mm for SG.

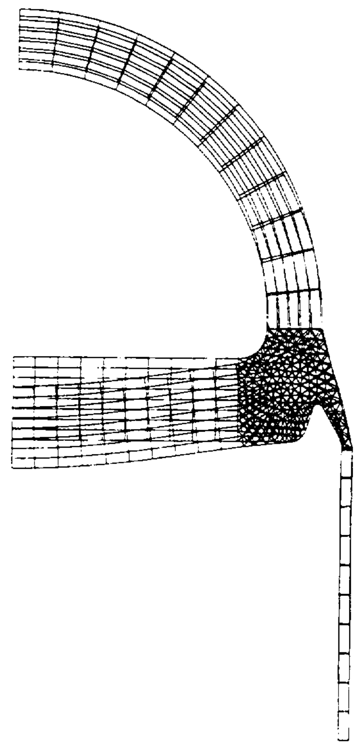


Fig 6 Optimised shape of SG tubesheet

3.1.4 Decay Heat Removal Systems

Two diverse DHR systems, one through the normal heat transport path, referred as Operational Grade Decay Heat Removal System (OGDHRS) and another through dip heat exchangers in pool referred as Safety Grade Decay Heat Removal System (SGDHRS) are provided. Continued long term DHR through OGDHRS is found to be essential for maintaining the required system temperatures during fuel handling and maintenance. In order to realise this, additionally, a small heat exchanger is incorporated in the steam-water circuit.

DHR through SGDHRS is envisaged during the plant upset (category 2), emergency (category 3) and faulted (category 4) conditions. Temperature limits for the clad (1073 K for category 2, 1173 K for category 3 and 1273 K for category 4) and cold pool components (813, 873 and 913 K for category 2, 3 and 4 events) have been established under various SGDHRS operating conditions. In order to satisfy the temperature limits, SGDHRS consists of 4 independent loops each with 8 MW capacity. Each SGDHRS comprises of one Na-Na heat exchanger dipped in reactor hot pool and one Na-air heat exchanger with motorised dampers on air side. From the analysis it is noted that the clad hot spot temperature of the storage subassembly exceeds the limiting value of 923 K, under situation of off-site power failure. Based on this preliminary analysis, it is recommended to provide emergency power supply for the primary pump alone to run at their lowest allowable speeds. However, detailed 3D thermal hydraulics investigations taking into account the inter-wrapper flow phenomenon are in progress.

3.1.5 Component Handling

Considerable progress has been made in the design of component handling systems. For handling core subassemblies within the main vessel, transfer arm type In-vessel Transfer Machine (IVTM) has been selected. For transporting core subassemblies towards the external storage, Inclined Fuel Transfer Machine (IFTM) with rotatable shield leg has been selected. Spent fuel subassemblies after washing are directly stored under water for cooling. Conceptual designs of IVTM, IFTM, fresh and spent fuel storage bays and their machines, and component handling flask have been completed. Detail design of IVTM including detail drawings has been completed (fig 7). Shielding for spent fuel storage bay is designed.

Criticality studies have been conducted for fuel storage bay to reduce lattice spacing. Temperature evaluation is made for spent fuel during its journey from the reactor to the spent fuel bay. Fig 8 shows the temperatures of the clad in a fuel subassembly with 5 kW decay powerat various stages. Clad stages. Clad temperatures are estimated in case the sodium filled pot carrying subassembly gets struck. It is found that the nominal central pin clad temperature exceeds the limit (923 K) when the pot is struck in roof slab region, rotating shield, secondary ramp and ex-vessel transfer position. Hence cooling is required. However, it remains to be seen whether temperature limits can be raised from the considerations of economy and avoidance of complex cooling arrangement.

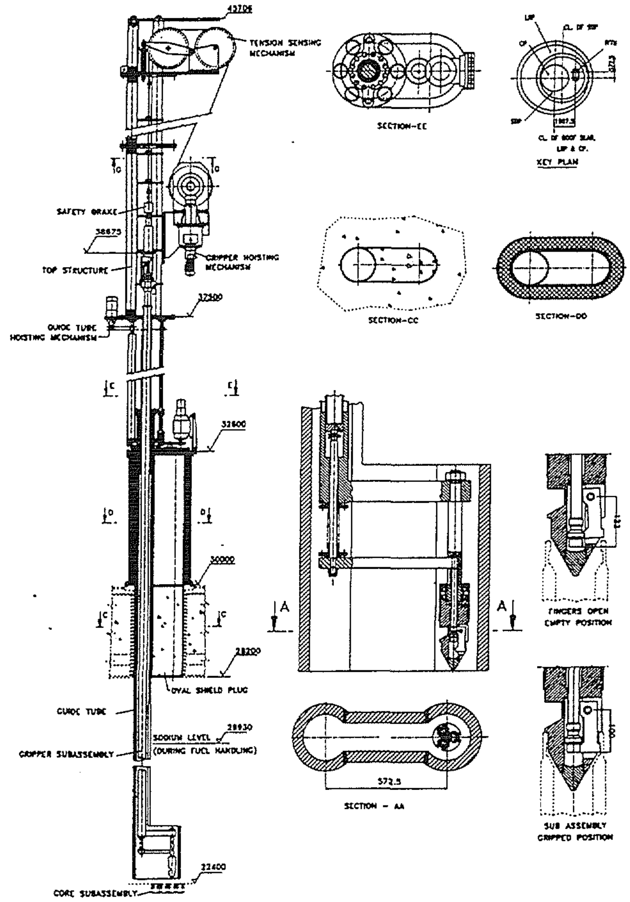


Fig 7 Invessel Transfer Machine

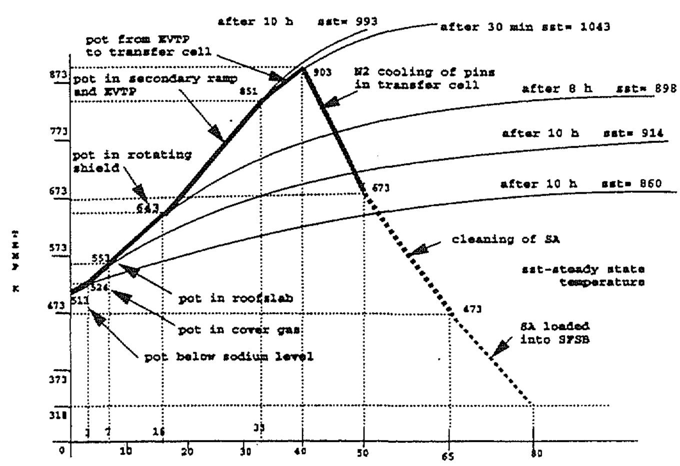


Fig 8 Clad temperature evolution during 5 kW spent fuel handling

3.2 Plant Dynamics and Shutdown Systems

The purpose of the plant dynamics study is to determine power, flow, temperature and pressure transients under various design basis events (DBE). These parameters are essential for deciding on the requirement of NRV, flow halving time and for arriving at reactor shutdown parameters along with the corresponding threshold values. The evaluation of temperature, pressure and flow are the input to the structural integrity assessment.

Various transients analysed are one primary or secondary or boiler feed water pump trip, one primary or secondary pump seizure, rupture of one primary pump discharge pipe, offsite power failure, uncontrolled withdrawal of a control and safety rod, total loss of feed water to SG, one primary or secondary pump acceleration from 20 % power and feed water flow increase to 125 % in one loop. Based on these studies reactor scram and LOR parameters are identified. Reactor is scrammed, i.e., by gravity drop of all control & safety rods (CSR) and diverse safety rods (DSR), only for events involving fast transients and flow blockage in the core. For all the other events LOR (lowering of all the control and safety rods) is used for the reactor shutdown. The safety criteria is to ensure the availability of two diverse reactor trip parameters for every DBE (fig 9).

Reliability analysis of reactor shutdown system is being carried out for various combinations of safety parameters and combination of drive and rod mechanisms.

It has been established that absorber rod drop time should be less than 1 s so that the reactivity transient of a few dollar per second resulting from coolant voiding, fuel melting and slumping can be protected by the shutdown system. Analysis of reactivity transient shows that absorber rod speed can be in the range of 1-4 mm/s. 2 mm/s speed has been decided for CSR and 4 mm/s for DSR. Scram delay time on reactivity events should be less than 250 ms.

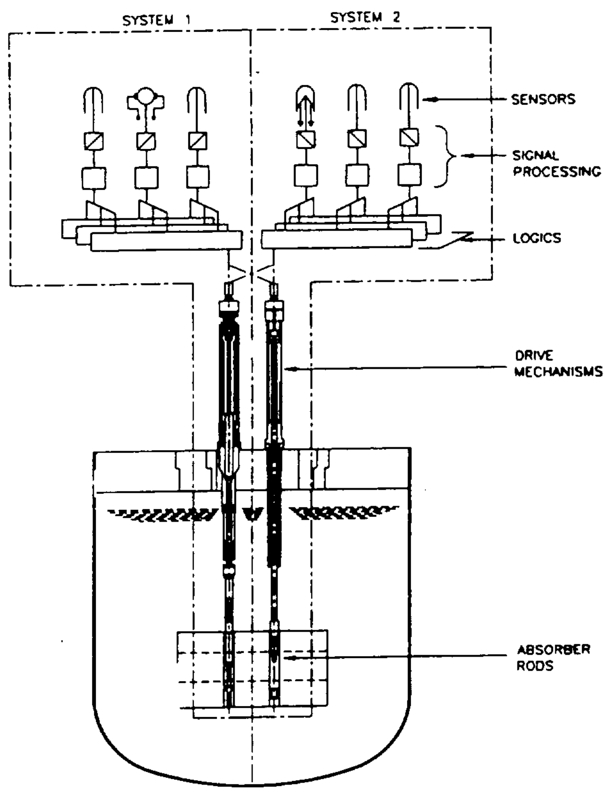


Fig 9 Shutdown system schematic

3.3 Instrumentation and Control (I&C)

The general philosophy to be followed and major features of I&C systems has been decided. Design of control room is under review and man-machine interface is under study. Conceptual design of systems for core temperature monitoring, flow measurement in primary sodium, reactor protection, failed fuel detection and SG leak detection have been prepared. Flux monitoring and failed fuel detection systems are under discussion.

3.4 Core Disruptive Accident

Sodium release to RCB under CDA has been estimated at about 1.5 t, based on the approach followed for FFTF reactor. The important input to this analysis are the transient and quasi-static pressure of the sodium after slug impact beneath the top shield and the fraction of the sodium mass in the reactor assembly which has potential to get ejected. These parameters are obtained from the detailed fast transient fluid-structure interaction analysis using an in-house computer code called 'FUSTIN'. A preliminary estimate is also made on the transient pressure and temperature rise in the RCB for the 1.5 t of sodium release and the values are 30 kPa and 80 K respectively.

3.5 In-service Inspection

A task force has reviewed the Inservice Inspection requirement of NSS systems/ components and recommendations have been made for ISI.

3.6 Technology Development

Technology development of main vessel, inner vessel, SG and primary sodium pump are progressing at manufacturing works. Evaluation of quotation for CSRDM is in progress. Tender drawings and specifications for roof slab, grid plate are under preparation. Tender documents for large diameter pipe bends, Tees and tube to tubesheet joints for 0.8 mm thick IHX tubes are prepared. A 1:5 working model of the transfer arm has been fabricated to demonstrate the fuel handling operations. Grade 91 welding consumable and T91 tubes for SG have been taken up for indigenous development.

3.7 Seminars

In continuation of a series of seminars on the designs involving interdisciplinary aspects, a workshop was held on rector shutdown system during 4-6 March 1997 at the centre. The scope of the workshop includes regulatory aspects, design philosophy, design concepts, neutronic & engineering aspects, existing designs, operating experience, reliability, manufacture, quality assurance and surveillance. This workshop provided a forum for effective exchange and dissemination of information among all those connected with nuclear reactor shutdown system (PHWR, BWR, FBR).

3.8 Conceptual Design Report

Conceptual design report has been prepared except chapters on I&C, plant layout and civil structures. Various plant design data viz. plant life, design basis events, safety classification, seismic categorisation, applicable design and construction codes, computer codes, material data and site data have been compiled.

Detailed design of NSSS is in progress. R&D works to be completed before start of construction have been identified and priorities have been decided for completion within 2-3 years. For this, collaborative works have been established with other R&D institutes. A Project Design Safety Committee has been constituted by the AERB. Safety evaluation is likely to be completed in about 2 years.

4.0 RESEARCH AND DEVELOPMENT

4.1 Reactor Physics

A new 25 group cross section data set has been prepared for FBTR Mark II core analysis for following isotopes : Fe, Cr, Ni, Th-232, Li-6, Li-7, N, Sn, Sb-121, 123, 124 and Bi. The new 25 group cross section data has also been prepared for two pseudo fission product lumps (including 111 fission products) for PFBR core burnup analysis. A computer program to refine the calculation of self shielding factors in a group partly covering resonance cross section region and partly covering smooth cross section region has been prepared. The errors made in the self shielding factors model, due to the assumption of constant background cross section and due to interpolation on the 2D grid of temperature and back ground cross section, are quantified for typical fast reactor cores. The multiband method of self shielding is investigated and required subgroup parameters is evaluated for several fast reactor materials.

Hexagonal geometry power distribution has been calculated for FBTR core to estimate precisely heat generation rates. The variation of power coefficient and burnup reactivity loss rate as a function of burnup in FBTR has been measured.

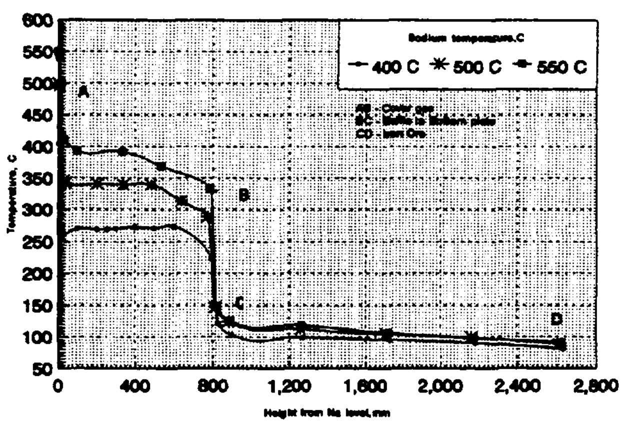
Benchmark ULOF of Modified BN-800 reactor with near zero sodium void coefficient of reactivity was analysed and results were presented at IAEA Consultancy Meeting of December 1996. Based on this, improvements to the pre-disassembly phase accident calculation codes have been made.

Beam forming technique for SG leak localisation has been successfully developed from noise data recorded on PFR steam generators during end of life experiments.

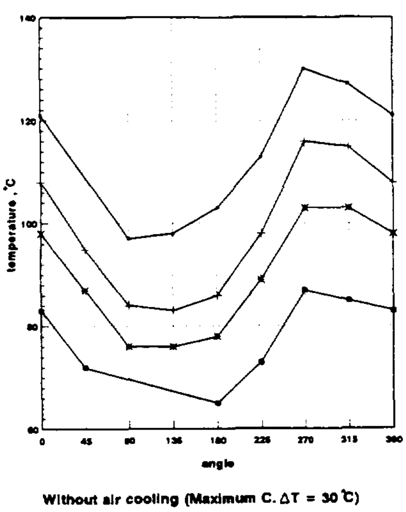
4.2 Engineering Development

4.2.1 Large Component Test Rig (LCTR)

The LCTR was operated at a maximum temperature of 773 K. The rig will be operated at higher temperature (~820 K) after carrying out improvements to the cooling circuit of an EM Pump. During the year the roof slab cooling circuit was commissioned and the heat flux on the



Cover gas space and roof slab



Annulus in roof slab

Fig 10 Temperature distribution in cover gas space and roof slab annulus

Table II : Heat flux to roof slab (W/m^2)

Na temperature (K)	Convection heat transfer	Radiation heat trasfer	Total heat trasfer
673 773	357 472	321 345	678 817

roof slab was studied with varying cover gas heights (table II). For 80 cm cover gas height, which corresponds to the prototype, heat flux at the sodium temperature of 773 K was 817 W/m^2. Fig 10 shows the temperature distributions in the cover gas and annulus of the roof slab.

The indigenously designed and fabricated Annular Linear Induction pump (ALIP) of 5 m^3/h, at 5 kg/sq cm rating was tested in the LCTR loop for hydraulic performance and endurance at 673 K for 1000 h. The measured head flow characteristics was slightly below prediction, probably due to the deviations in manufacture and assembly. Five more ALIP required for the in-sodium material test facility, have been manufactured.

4.2.2 Testing of Primary Sodium Pump (PSP)

The full size pump (hydraulic parts) of the 4 loop-PSP concept was tested at 500 and 600 rpm at the supplier's works (fig 11). The head developed, efficiency and NPSH have been met fully with the specifications, although there was a small deviation in the head vs flow characteristic curve vis-a-vis predicted one. A full scale hydrostatic bearing designed and manufactured at IGCAR was also tested along with the pump and its performance was found satisfactory. A bypass type flow meter was installed in the pump discharge and calibrated. The performance of the bypass flowmeter is close to the prediction over a large range of flow. This type of flowmeter is proposed for PFBR.

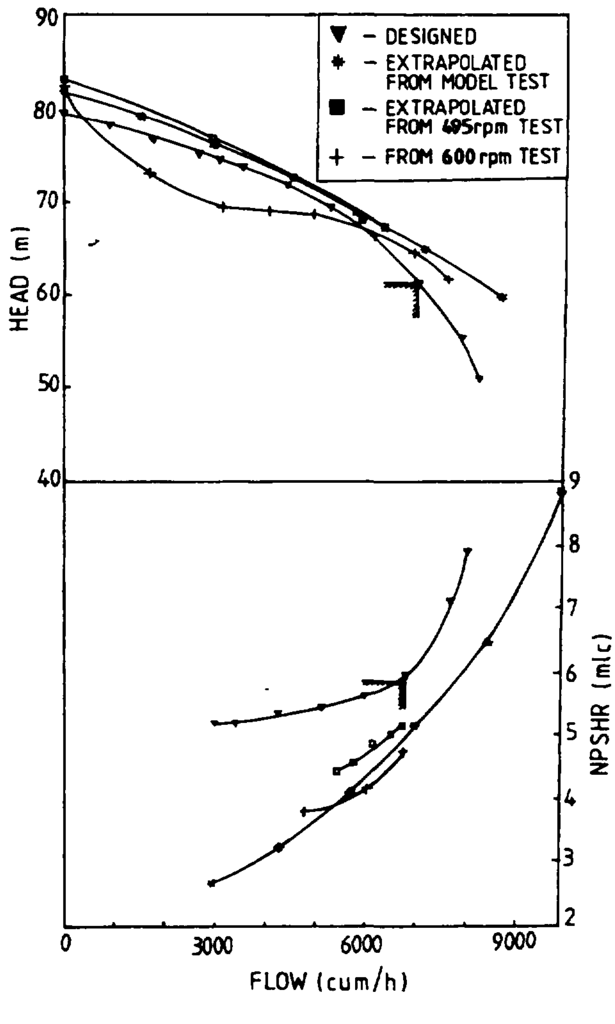


Fig 11 Head, NPSH and flow characterictics

4.2.3 Testing of CSRDM Subassemblies

Development of a quick release electromagnet has been completed. The release time of the electromagnet is reduced by slotting the solid core parts. Slotting requirements have been finalised based on a theoretical analysis using an equivalent circuit methodology. The new design is tested and the response time is found to be ~ 100 ms for the expected load range. A bellowless concept has been considered or the CSRDM with elastomer seal between the translation tube and the outer tube sheath. Different types of seals, viz floating "O" rings, oil seal without spring and custom-built "V" seals were evaluated for friction force and leak tightness. "V" seals of two different configurations have been qualified for further evaluation.

4.2.4 Air Tests on Grid Plate Model

Experiments have been completed on the 1/3 scale air model of the grid plate with and without baffle plate. The tests show that the flow distribution amongst the various subassemblies is ensured for the air flow range from 20% to 100% simulating operation with 2 to 4 pumps. The flow distribution is found to be unaffected even under simulated condition of a single inlet pipe rupture. The pressure drop across the grid plate has also reduced due to the absence of the baffle plate. Hence it is recommended not to have baffle plate in the grid plate.

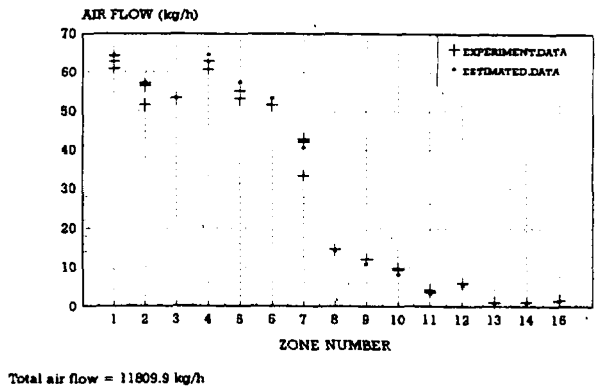


Fig 12 Air flow distribution in 1/3 scale model

4.2.5 Dummy Fuel Subassembly Tests in Water

A full scale dummy fuel subassembly with 217 pins and 150 mm wire wrap pitch has been tested in water for pressure drop. Additional tests are planned with higher wire wrap pitches of 200 and 250 mm. Fabrication of blanket and absorber subassembly is in progress.

4.2.6 Flow Induced Vibration Studies on SG Model

Hydraulic tests to assess the flow distribution and measure vibration of tube bundle are carried out on a 60 degree sector model of SG (fig 13a). The velocity distribution measured in the inlet plenum was found matching with prediction of the 3D hydraulic calculations (fig 13b). Vibration measurements are carried out in the straight spans and expansion bend regions. Although the vibration level is found higher in the inlet span where cross flow takes place across the tube bundle, it is within the acceptable values based on structural mechanics analysis (fig 13c).

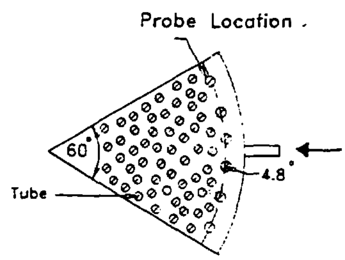


Fig 13a 60° sector model of SG

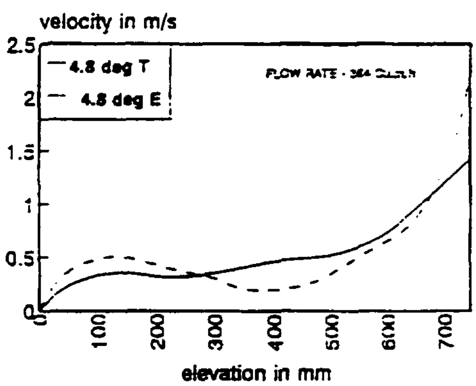


Fig 13b Velocity distribution in the inlet plenum

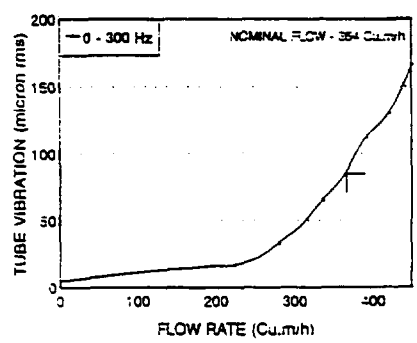


Fig 13c Tube vibration behaviour

4.2.7 Instrumentation

Development of sodium proof ultrasonic transducer was pursued. Tests conducted earlier had indicated that the couplant gets dried up at 523 K. To overcome this, soldering of the crystal to diaphragm was successfully carried out with a special solder alloy. Testing was carried out upto 493 K in silicone oil and it was found to work satisfactorily, though with reduction in echo amplitude. Further development and long term testing is in progress.

4.3 Structural Mechanics

For the seismic analysis using random vibration theory, floor response spectra in power density functions have been generated at all the important locations.

For the structural integrity assessment of control plug mockups, 75 thermal shocks have been given so far, based on the preliminary estimation of crack initiation. No visible cracks have been noticed due to the possible statistical variations in the creep-rupture properties. In order to continue the tests, the copper cylinder which was originally provided to simulate the temperatures on the outer shell of mock up, is to be replaced by mild steel cladded with SS. Theoretical analysis has been completed to confirm the use of mild steel in place of copper.

For establishing thermal stripping limits, the experimental set-up to simulate thermal striping on the horizontal plate has been built including arrangements for cooling of water to about 280 K and heating to about 360 K.

For the thermal ratchetting studies on main vessel under moving axial temperature gradient, a loading frame based on composite lever principles has been fabricated. Heating and cooling systems to simulate axial thermal gradient is being installed.

LBB analysis has been completed for SG - Sodium inlet nozzle using ANSYS code as per the conventional approach. Analysis is being repeated as recently documented in RCC-MR, appendix A 16.

3 concrete test beds (reaction floor) have been constructed to conduct a series of CDA experiments using TNT and low density pentolite chemical explosives. First series of preliminary tests on cylindrical shells will start by July 1997.

Theoretical and experimental investigations are made on a 1/6th scale model of primary pump discharge pipe to estimate the collapse load with and without cracks. Crack length is varied in the studies. Further provisions are being made to carryout LBB studies in this setup itself.

2 SG straight tubes (7 span full size mockups) have been tested for buckling under axial compression. The support plates are stiff in the present study. The experiment has demonstrated that the tubes can withstand a temperature difference of 60 K as per RCC-MR (1993) buckling design. In order to repeat the tests with realistic support grids, the support grids are being fabricated. One stainless steel vessel of 330 mm dia and 0.5 mm thickness is tested for buckling under axial compression. Theoretical buckling strength predicted by 'INCA' code is 3.2 t and experimental value is 3.8 t. The buckling mode shapes are predicted satisfactorily by CASTEM 2000 code.

Towards validating `SODSPIL', experiments have been conducted to generate the preliminary data on ejection of water through the annular space in the rigid cover of the vessel filled with water subjected to sharp pressure pulse (the shape of the pressure pulse is correctly simulated in the test). The code predictions are satisfactory. Tests are further planned involving pressure measurements in the tank.

4.3.1 Computational Facilities

Very powerful computers, viz. Silicon Graphics Power Challenger (speed 125 mflops, R8000 RISC Processor), Indy (15 mflops and R4000 RISC Processor) and Indigo 2 (35 mflops and R4600 RISC processor) have been installed during the year at the centre. In these computers, commercial computer codes, viz. ABAQUS version 5.5, MSC-NASTRAN, EUCLID, MSC-PATRON and IDEAS have also been commissioned.

4.3.2 Structural Mechanics Laboratory

For carrying out specialised structural mechanics experiments in the domain of high temperature design including ratchetting and thermal striping, buckling, seismic and leak before break and fracture investigations, construction of a sophisticated structural mechanics laboratory has been started in February 1997. The first phase of the construction will be complete by December 1997.

4.4 Metallurgy

4.4.1 Tensile, Creep, Low Cycle Fatigue and Creep-Fatigue Interaction Behaviour

The creep properties of SS 316 LN base metal and weld metal are studied at 873 and 923 K in the range of applied stress from 120 to 315 MPa. The results are compared with those obtained for SS 316. Generally the creep rupture lives of the weld metals are found to be lower than those of respective base metals by a factor of 5 to 10. Both the base and weld metal of SS 316LN exhibited better rupture lives compared to SS 316 base metal and weld metal respectively, in identical test conditions.

At 923 K, the stress exponent for minimum creep rate is about 9.5 for both the base metals, whereas it is about 12 for the weld metals. However, in the case of SS 316, the weld metal exhibited higher creep rates compared to the base metal whereas for SS 316LN the creep rate of the weld metal was lower by a factor of 5 at high stress levels and 10 at lower stress levels, compared to that of its base metal. Both the weld metal exhibited lower rupture elongation compared to that of the respective base metals at 873 and 923 K. Comparison of the rupture lives of the two steels with the ASME Code Case N-47 curves for the base and weld metals showed that in the case of SS 316 LN, the codal allowable stresses are over-conservative.

The tensile properties of forged 9Cr-1Mo-Nb-V steel have been determined at three strain rates (3×10^{-4}, 3×10^{-5}, 6×10^{-5} 1/s) in the temperature range 297-973 K. In general, at all the strain rates, the yield and the ultimate tensile strength values decreased gradually as the test temperature was increased from room temperature to 498 K. The alloy exhibited strength plateaus in the temperature range 498-648 K. Above 648 K, the strength decreased marginally with decreasing strain rate. The alloy exhibited a broad minima in ductility in the range 498-648 K. The yield and ultimate tensile strength values obtained for the forged alloy at different temperatures have been found to be only slightly lower than those given by ASME Sec-VIII average curve for rolled bars. The forged alloy, however, displayed much inferior values of ductility compared to the average ductility curve reported by ASME.

A detailed understanding of the creep deformation and fracture of 9Cr-1Mo base and weldments at 823 K has been developed with a view to optimizing the performance for steam generator applications. Creep rupture lives of weldments has been found to be significantly inferior compared to the base metal in the range of stresses between 130-250 MPa.

Optical micrography and micro hardness investigations are carried out in as-welded, post-weld heat treated and creep tested conditions. The heat affected zones (HAZ) of 9Cr-1Mo steel weldments consist of coarse grain martensite with δ-ferrite, coarse grain martensite, fine grain martensite and intercritical structure. A hardness trough is noticed in the intercritical regions of HAZ which becomes more predominant after PWHT and creep test.

A project has been initiated to study the effect of Ti/C ratio for optimum creep-fatigue interaction properties of indigenously developed Alloy D9. As part of this programme, baseline LCF data has been generated at 823 and 923 K on this alloy with a Ti/C ratio of 4. The alloy is found to obey the Coffin-Manson relationship at both these temperatures.

The effect of temperature on the low cycle fatigue properties of this alloy has been studied. Low cycle fatigue life peaks around 573 K and then gradually decreases. Evidences for the occurrence of dynamic strain ageing, such as increased cyclic hardening have been observed at 823 K. The alloy with a Ti/C ratio of 4 has also been tested in the 20% prior cold worked condition. While at 823 K the cold worked alloy exhibited a linear strain-life relationship, at 923 K, the strain-life relationship exhibited a two slope behaviour. The solution annealed material exhibited better life than cold worked material at both 823 and 923 K. Preliminary hold time results indicate that the cold worked material has better creep-fatigue interaction resistance. Tests with longer hold times are in progress.

4.4.2 Fracture Mechanics Material Properties

Reference nil-ductility transition temperature (RT_{NDT}) of 9Cr-1Mo welds fabricated using different diameter electrodes (2.5, 3.15 and 4 mm) were determined. There was no significant dependence of RT_{NDT} on the electrode diameter. Load-time curves from drop-weight and Charpy V- notch (CVN) specimen tests were analysed for obtaining conservative K_{Id} estimates. These were compared with the ASME K_{IR} curve (ASME Section III, Div. I, Appendix G). The ASME K_{IR} curve is applicable to carbon and low alloy ferritic steels with minimum room temperature yield strength of 345 MPa or less. Hence ASME K_{IR} curve is applicable to other high alloy/high yield strength steels like 9Cr-1Mo steels (the weld has a yield strength of 550-580 MPa at room temperature) needs further validation. In the present case, the estimated minimum dynamic fracture toughness (Kid) values obtained for the 9Cr-1Mo welds upto 323 K were lying above the ASME K_{IR} curve. The conservatism of the ASME K_{IR} curve at higher temperatures needs further validation.

The CVN transition and Kid properties of 9Cr-1Mo plate steel in normalised and tempered (N+T) and in two simulated post-weld heat treated (PWHT) conditions were also evaluated. PWHT had no significant effect on the properties which is in conformity with the trend in literature where this steel is stated to be remarkably tolerant to wide variations in heat treatment conditions. Moreover, the cleavage strength of this steel was higher than that for the weld reported above, indicating comparatively poorer fracture resistance for the weld material.

4.4.3 Weldability

The weldability of several austenitic stainless steels was studied with particular reference to the effect of nitrogen on cracking of type 316L stainless steel. The results showed that in the fusion zone, cracking was greatly enhanced when nitrogen was present in the range 0.06-0.12% with an austenitic-ferritic (AF) solidification mode, than when a fully austenitic solidification mode is present. Analysis of the composition data from the literature and the present results showed a possible role of phosphorus segregation in enhancing cracking in the mixed AF mode. In compositions with low P levels (< 0.025) and for similar nitrogen contents, the cracking is always higher in the fully austenitic solidification mode.

Weldability of alloy 718 (material for intermediate support grids of SG tubes) was investigated using the Varestraint hot cracking test. The calculated heat input did not show any correlation with the hot cracking test results. However, the corresponding center line cooling rate, calculated using Rosenthal heat flow equation, showed very good correlation with the hot cracking test results, according to which the cracking in the fusion zone decreased with increasing cooling rate. Fabrication weldability tests using a fillet geometry showed that Inconel 718 was more cracking-sensitive than Inconel 82 filler for welding alloy 718.

The aluminizing process for alloy 718 was scaled up from laboratory scale to actual scale for treating the strips for manufacture of PFBR steam generator tube bundle support structures. Aluminizing was carried out using the low activity pack process. The process cycle was standardised and uniform aluminide layer of 80 m thickness was obtained over 600 mm long strip.

Diffusible hydrogen (HD) measurements in the welds of three different Cr-Mo steels namely, 2.25Cr-1Mo, 9Cr-1Mo and 0.5Cr-0.5Mo were carried out. It was found that for given vol.-% of hydrogen in the shielding gas, HD was maximum for 0.5Cr-0.5Mo steel and minimum for 9Cr-1Mo steel. However, lower HD content in the 9Cr-1Mo steel did not result in lower susceptibility to hydrogen assisted cracking; among the three steels studied; it showed maximum susceptibility to cracking. Effect of preheat temperature on HD was also studied. It was found that there is substantial reduction in HD content with preheat temperature. The results indicate that the weld prepared with preheat can tolerate higher HD levels without resulting in cracking than those prepared without preheating. It is inferred that, in addition to external sources of hydrogen, alloy content, microstructure and defect density of the weld also influence the HD content in the welds.

As a part of the developmental effort towards the trimetallic transition joint (316LN SS/Alloy 800/9Cr-1Mo steel) for the steam generator shell-nozzle junction, the effect of aging on the interfacial microstructure and tensile properties of as-welded and post-weld heat treated Alloy 800/9Cr-1Mo steel joint (welded with Inconel 182) was investigated. These joints were post-weld heat treated for 1 h at three different temperatures, viz. 973, 998 and 1023 K and then aged at 848 K for 100, 500, 1000 and 5000 h for accelerated simulation of long-term elevated temperature service exposure. It was observed that aging at 848 K up to 5000 h had only a marginal effect on the tensile properties of this joint, with the Inconel 182 weld/9Cr-1Mo steel interface and the 9Cr-1Mo steel HAZ exhibiting excellent resistance to aging induced microstructural instability. The ductile fracture toughness parameters from room temperature tensile tests of transverse-weld specimens (corresponding to the weakest region in the 9Cr-1Mo steel HAZ) lead to unambiguous identification of the optimum post-weld heat treatment temperature of this dissimilar metal weld joint as 823 K.

4.4.4 Ultrasonic Velocity - New tool for Characterising Annealing Behaviour

Ultrasonic velocity measurements were found to characterise the recrystallization behaviour of 20% CW alloy D9 (Ti-modified austenitic stainless steel). Ultrasonic velocity measurements at different frequencies of 2, 5, 10, 20 MHz were carried out on the cold worked and annealed samples of alloy D9. Ultrasonic velocity measurements could distinctly identify the recovery, recrystallization and grain growth regions, during annealing of cold worked austenitic stainless steel. Compared to hardness testing, this technique was found to be more accurate in characterising the annealing behaviour of cold worked austenitic stainless steel.

4.4.5 Microstructural Characterisation and Environmental Effects

The mechanical properties of solution annealed SS 316 LN were studied after ageing at 1123 K for 10 h. Initially both strength and ductility increased and this could be attributed to the formation of Cr-N clusters. On further ageing, while strength increased, ductility decreased because of intragranular chromium nitride precipitation. SCC resistance is increased twice by ageing the solution annealed type SS 316 LN at 1123 K for 500 h. The increase in SCC resistance was marginal thereafter.

Several factors influencing localised form of corrosion in austenitic type SS 316 LN were studied. These include temperature of the medium and surface treatment by N-ion implantation. In the case of the welds, the role of solute elements have been independently examined through their influence on the ferrite content. In all the cases, electrochemical techniques like linear polarization measurements were employed to determine the corrosion sensitivity range and establish the corrosion resistance regimes.

The characterisation of the as-welded microstructure in different regions of the 9Cr-1Mo weldment has been carried out by extensive cross section electron microscopy. The observed microstructures at varying distances from the heat source have been correlated with the temperature isotherms predicted for each region. The repartitioning of solutes, namely Cr and Mo between the weld metal and heat affected zone has been established.

The post weld ageing behaviour of 9Cr-1Mo weldments in the temperature range 823-1023 K has been extensively studied by analytical transmission electron microscopy. The secondary phases that evolve have been identified, their chemistry established and the concept of phase evolution diagram extended to the weldments.

Oxidation behaviour of 9Cr-1Mo steel was examined under a tensile stress of 40 MPa and temperature of 973 K for different durations in the range of 25-140 h. The integrity of the oxide scale was monitored in-situ by recording the acoustic emission activities associated with the cracking. Specimens were also oxidised under similar conditions without the application of the external stress.

4.4.6 Materials for Reprocessing Applications

An ongoing study on materials for reprocessing including surface treatment of electrodes used in electrochemical processes has yielded important results. Vapour phase corrosion tests in boiling HNO3 were carried out and data has been obtained for advanced construction materials (high Cr alloys like Uranus 65 and Uranus SIN). New improved Pt-Ir containing metal oxide coatings were developed for Titanium anodes to be used in electrolytic dissolver in reprocessing plants. A nitric acid loop has been designed and its safety review is being done.

4.4.7 Biocorrosion of Materials in Fresh Water

Effect of surface finish on the biofouling rates of different materials was studied by standardized harmonic mean (SHM) of drop spread determination method. It was observed among different materials like titanium, stainless steel, admiralty brass etc., carbon steel was the most affected (highest value of SHM). Here the corrosion causing slime formers were gram-negative bacteria like Pseudomonas species. The influence of water quality parameters on the tuberculation of carbon steel (ASTM A-106-Grade B) was examined. Tuberculation of carbon steel was initiated under the influence of iron oxidizing bacteria (IOB) and the growth of tubercles was found promoted under sufficient supplies of oxygen and flowing water. Detailed studies indicated that in the presence of IOB, poorly crystalline ferri hydrite formed which then initiated tubercles and consequently promoted corrosion reactions.

Electrochemical studies on the effect of biofilms on SS 304 were carried out. It was observed that algal dominated biofilms cause ennoblement in stainless steels. Also it was shown that while biofilms enhanced the pitting potential (and thus decrease susceptibility to pitting), they also increased the passive current rendering the sample susceptible to crevice corrosion.

4.4.8 Fuel Fabrication

Studies were continued at Radiometallurgy Division, BARC, Mumbai, to optimise the process parameters for fabrication of $(U_{0.8}Pu_{0.02})O_2$ fuel pellets of 85% T.D using UO_2 and PuO_2 powders with thermally stable pores suitable for use as driver fuel. The parameters optimised are low temperature oxidative sintering in N_2+air atmosphere, high temperature reductive sintering in Ar+8%H_2 atmosphere, sintering time, quantity and particle size of pore former.

Compatibility studies of the $(U_{0.45}Pu_{0.55})C$ with SS 316 clad at 973 K and 1123 K for the time duration of 1000 and 2000 h has been completed. Compatibility studies of this fuel with sodium coolant is being carried out.

4.5 Irradiation, Post irradiation Experiments and NDT Development

4.5.1 Fuel Irradiation Experiments

Irradiation of seven fuel pins has been completed. The pins were irradiated in the fifth row of FBTR for periods ranging from 16 to 100 d at the power level of 10.1 MWt. The aim of the experiments is to study the evolution of fuel structure at the beginning of life of the fuel in the reactor. The linear power at the location of irradiation was about 180 W/cm. Three of the fuel pins have 70% PuC-30% UC pellets. The other four fuel pins have two short stacks of fuel pellets - one stack of pellets with the Mark I composition and the other stack of pellets with the Mark II composition (55% PuC-45% UC). All the experimental fuel pins have been filled with a mixture of argon and helium as the bond gas, to enable generation of higher temperatures at the fuel center corresponding to irradiation at higher linear powers. The equivalent linear power generated due to this was about 270 W/cm. Three of the experimental fuel pins have been received for post irradiation examination in the hot cell facility. These have been irradiated for 16, 16 and 26 d. Non-destructive examination of the fuel pins has been completed. The techniques used were eddy current inspection of clad, X-radiography of the fuel pins, leak testing and profile measurement. The examination has not revealed any defects. There is no swelling and the fuel-clad gap has not yet closed.

4.5.2 Irradiation Experiments on Alloys of Zirconium

In order to determine in-reactor creep rates of Indian made Zirconium alloys, it is decided to irradiate the pressurised capsules made of Zircaloy-2 and Zircaloy-4 Niobium alloys in FBTR. Towards this, six pressurised capsules have been fabricated and ready for irradiation.

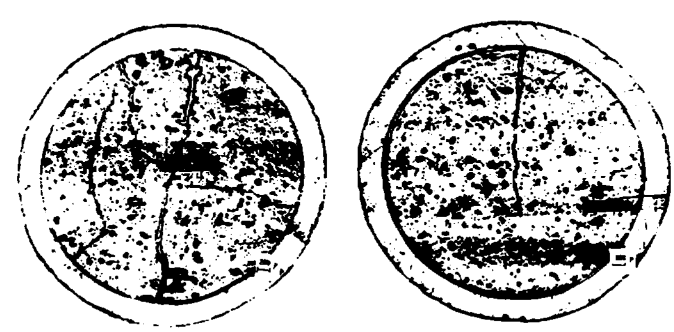
4.5.3 PIE on FBTR Fuel Subassembly

The central fuel sub-assembly (~ 2.1 kg mass, 25,036 MWd/t burn-up, 320 W/cm linear heat rating and ~15 dpa) discharged on July 1996 was taken to the hot cells of Radiometallurgy Laboratory for Post Irradiation Examination (PIE). In the hot cells, highly irradiated advanced fuels are handled and examined under inert nitrogen atmosphere, where the temperature, pressure, and purity are closely controlled. The examinations carried out on SA are visual examination, removal of sodium, temperature and dimensional measurements, and dismantling. The fuel pins were subjected to leak testing, ultrasonic cleaning in alcohol, diameter measurements, eddy current testing, X- radiography, and metallography of cut sections of the fuel. Visual examination of the fuel SA did not show any signs of corrosion or surface defects. The SA was washed in the alcohol recirculatory system to remove the sodium.

Dimensional measurements of the fuel SA indicated that the fuel region of the SA did not indicate any swelling or bulging of sheath. The width-across-flat was found to be generally within the original limits of tolerance. Fuel pins did not show any corrosion or deformation on visual examination. However, the central fuel stack region appeared discoloured. Diameter measurements of nine pins were carried out in two different orientations, using a measurement bench which holds the pin vertically and scans the diameter along the length using two LVDTs fixed diametrically opposite to each other. Accuracy of measurements is within 1μ. Results indicate that the diameter is generally within the original limits of tolerance and that there is no clad deformation.

Leak testing of the pins was carried out by evacuating a closed chamber partly filled with ethylene glycol and the pins submerged inside it. While bubbles from a calibrated test pin with standard defects of 0.1 mm and 0.2 mm diameter through-holes could be identified, the fuel pins did not indicate any leak.

Eddy Current Testing (ECT) did not indicate any abnormality. Analysis of signals indicated that the spring support inside the pin is touching the clad and contact location is varying for the nine pins. The accuracy of location is within 2 mm. The location of spring support determined using ECT indicated that the pellet stack length has increased. These results co-relate well with that of X- radiography which followed ECT. X- radiography revealed that the pellet-clad gap and the pellet-to-pellet gap are too small to be measured. (Original pellet-clad gap is within 0.11 to 0.25 mm). An increase in the fuel stack length was noticed. This varies from 2.171 to 5.350 mm (0.68% to 1.67%). No gross abnormalities were observed.



a)Centre of fissile column 300W/ cm, 25,000MWd/t b) Top of fissle column 210W/cm,18,000MWd/t

Fig 14 Micrographs of the fule cross sections

One of the pins showing a stack length increase of 0.825% was first taken up for carrying out sectioning and metallography. The fuel was found to be cracked at the centre of the fuel column, whereas at the end, the cracking was relatively less (fig 14). The fuel-clad gap was found to be closed due to cracking, as well as swelling of the fuel. The fuel-clad gap was less at the centre of the fuel column, compared to the end, as expected. The presence of marginal fuel-clad gap, as well as the gaps available in the fuel due to cracks, indicate that, space is still available to accommodate further swelling of the fuel. However, the rate of swelling is expected to come down, since the centre-line-temperature of the fuel will be lower due to fuel-clad gap closure.

4.5.4 Eddy Current Inspection of FBTR Cladding Tubes

ECT was carried out at Nuclear Fuel Complex, Hydrabad and it was observed that a large number of continuous indications were obtained throughout the length of the tubes. In view of the large amplitude indications (poor signal to noise ratio, SNR) meaningful ECT was not possible. This poor SNR was attributed to banding (variation in diameter of the tubes at intervals) of the tubes. In order to solve this problem, a novel technique called Phased Array ECT (PAECT) was carried out on 205 cladding tubes and of which, 30 tubes were rejected due to the presence of defects.

4.5.5 Commissioning of KAMINI Reactor

30 kWt U-233 fuelled KAMINI reactor was made critical in October 1996 and will be used for activation analysis and neutron radiography. It is located below the hot cell of radiometallurgy laboratory. Fig 15 shows a view of the KAMINI reactor.

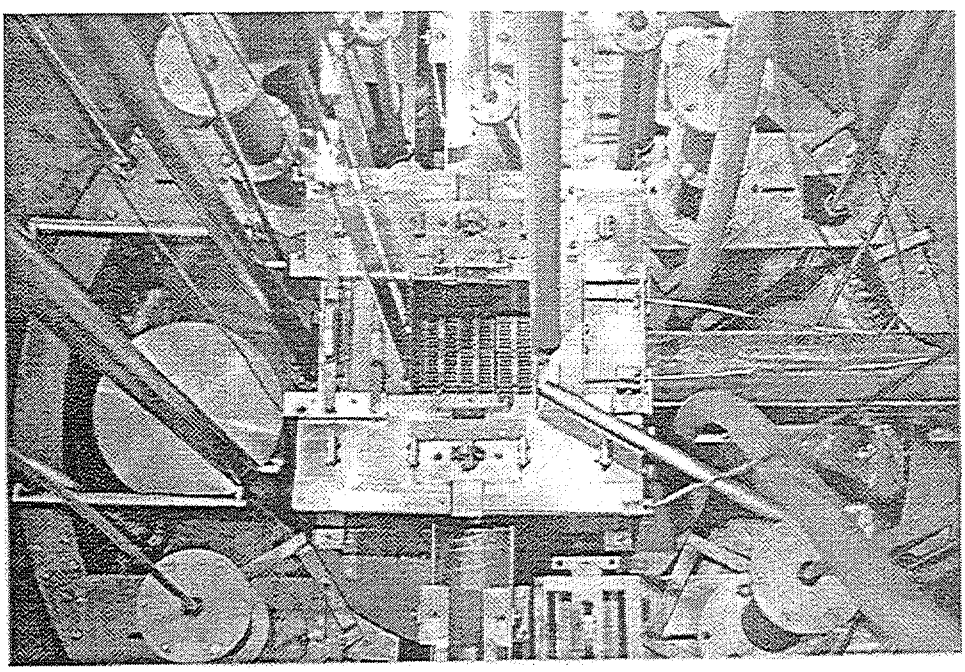


Fig 15 A view of the KAMINI reactor

4.6 Chemistry

4.6.1 Thermochemistry

Ca-Ga alloys are used for the recovery of actinides from molten salts which have been used in pyrochemical processes. The formation actinide-gallium alloys greatly facilitates the extraction of actinides from the salts. Hence thermodynamic investigations on U-Ga alloys have been carried out. The enthalpies of formation of the intermetallic compound, UGa_3 at 298.15 K and at 1038 K were determined by gallium solution calorimetry and precipitation calorimetry respectively. The enthalpy of formation of UGa_2 at 1563 K was determined by precipitation calorimetry. A molten salt emf cell technique employing $CeCl_3$/(LiCl+KCl) as the electrolyte was employed to determine Gibbs free energies of formation of $CeNi_5$ which is of interest for understanding the fuel-clad interaction mechanisms during the irradiation of (U-Pu-Zr) fuel.

Oxygen potentials of $(U_{0.9}Th_{0.1})O_{2+x}$ $(U_{0.77}Th_{0.23})O_{2+x}$ and $(U_{0.54}Th_{0.46})O_{2+x}$ were measured in the temperature range of 1073-1173 K using a gas equilibration method. H_2/H_2O, CO/CO_2 and CO_2/H_2 gas mixtures were used for fixing the oxygen potentials in these studies. The oxygen potentials measured were in the range of -450 to -220 $kJ.mol^{-1}$ and the oxygen to metals ratios ranged from 2.000 to 2.040.

The formation of solid solutions of $(U_{1-y}\,Gd_y)\,C_2$ $(0.2 \leq y \leq 0.8)$ in the high carbon region, have been studied in the temperature range 1063-1673 K. The solid solutions were prepared by blending the stoichiometric amounts of oxides with graphite and heating the mixture in the form of a pellet to the reaction temperature in a high vacuum chamber. The effusion pressures of CO in the vacuum chamber were correlated to the equilibrium pressure of the system, according to a new method developed in our laboratory. From the equilibrium CO pressures calculated, the Gibbs energies of formation of the different solid solutions were determined, by taking the appropriate free energy data of the oxides, graphite and the CO phases from standard thermodynamic tables. The temperature dependence of free energies of formation was also determined.

The thermodynamic study of rare earth dicarbide systems is essential in understanding the chemical state of fission products in the mixed carbide fuels used in fast breeder reactors. Thermodynamic properties of one such carbide, lanthanum dicarbide (LaC_2) in the temperature range 1263-1543 K have been determined by using the new method referred to above.

The carbon potential corresponding to the two-phase mixtures LaC_2-La_2C_3 and CeC_2-Ce_2C_3 were measured for the first time, in the temperature range of 973 K to 1173 K, by using the methane-hydrogen gas equilibration technique. The measured values of the chemical potential of carbon are in good agreement with the evaluated data.

4.6.2 Development of Nuclear Techniques for Non-destructive ASSY OF SNM

Assay of plutonium bearing waste is essential both from the point of view of nuclear material accounting as well as waste disposal. The standard technique for this purpose is high resolution gamma spectrometry using segmented scanning of the sample. A new easy-to-use segment correlation method has been developed to correct for the contribution from the neighbouring segments of the sample to segment under assay. Correction for attenuation of the gamma radiation due to the high density of the sample was carried out by using a point source of ^{133}Ba which has gamma lines near the lines of ^{239}Pu. The total plutonium inside the pins was estimated correctly to within 5%. The plutonium and all fission products were confined to the fuel zone and no signature of any fission product or fuel was seen outside this zone. The fission products seen were ^{106}Ru, ^{137}Cs and ^{144}Ce. The burn up seen by the two pins was computed using the total amounts of these fission products and was around 1 MWd/t. Radionuclides seen from the clad were ^{54}Mn, ^{60}Co, ^{125}Sb and ^{182}Ta. The activity profile along the pin for ^{54}Mn reflected the fast flux while the profile of ^{60}Co was seen to reflect the thermal flux profile. Comparison of the experimentally determined activities of ^{54}Mn with calculations revealed reasonable agreement while the experimentally determined activity of ^{60}Co was much lower than calculations.

4.6.3 Reprocessing Chemistry

Tri-n-amyl phosphate (TAP) is a promising extractant for use in fast reactor fuel reprocessing. The effect of diluent on the enthalpy of extraction of uranyl nitrate by tri-n-amyl phosphate was studied in the temperature range 283 K to 303 K employing a number of diluents including isooctane, n-dodecane, cyclohexane and benzene. Except o-dichlorobenzene, all the diluents gave almost comparable enthalpy values.

The effect of solvent structure on the enthalpy of extraction of americium (III) nitrate was studied using various trialkyl phosphates such as tri-n-butyl phosphate (TBP), tri-isobutyl phosphate, tri-n-amyl phosphate (TAP), tri-isoamyl phosphate and tri-n-hexyl phosphate. The results indicated that the enthalpy of extraction decreased in the order TBP>TsBP>TAP>TiAP THP. This behaviour was in contrast to the variation of the enthalpy of extraction of uranium (VI) with the alkyl carbon structure of the trialkyl phosphates reported in literature.

Studies on extraction of Am (III) from high active waste (HAW) solution have been initiated. The extractant CMPO (Octyl, Phenyl - N,N-diisobutyl Carbamoyl Methyl Phosphinoxide) has been synthesised and its extraction behaviour characterised. Studies have been taken up on third phase formation in the extraction of Nd(III) by CMPO, with TBP and as modifiers. The data measured employing TAP indicate that this modifier can permit high organic loadings without third phase formation.

4.6.4 Development of Hydrogen Meters

An electrochemical hydrogen meter based on ternary molten salts with $CaCl_2$ and CaHCl as two components of the electrolyte was constructed and tested in a bench-top sodium loop. Further tests regarding the long term performance of the meter are in progress.

A hydrogen sensor based on a proton conducting polymer (Polyvinyl alcohol - phosphoric acid) working on the amperometric principle was developed and tested. This sensor responds well even for 1 ppm of hydrogen in argon. The long term stability of this sensor is being evaluated with a view to use it in place of the thermal conductivity detector in cover gas hydrogen monitor.

4.6.5 Sodium Removal and Decontamination

A comparative study of different solvents used for sodium removal from the structural components was made. Experiments were carried out in the temperature range of 303-343 K to find out the effect of solvent, orientation and exposed area of sodium surface on the dissolution rate in solvents such as ethyl carbitol, butyl cellosolve and Jaysol-SS vary in the ratio of 1:1.2:2.4 respectively. At low temperatures (~303 K), the rate constant increased 2.5 times when the exposed surface area of sodium increased by 2.5 times, but at higher temperatures this increase was only marginal, indicating that the reaction temperature has greater role to play than the surface area. The orientation of the sodium sample has very little effect on the dissolution kinetics.

4.6.6 Experimental studies on Equation of State of UO_2

Equation of state and in particular, vapour pressure data on nuclear fuel materials at very high temperatures (2500-5000K) are especially required to analyze accident conditions. A laser induced vaporisation mass spectrometric (LIV-MS) facility has been developed to measure these high temperature vapour pressures of fuel materials. Initial experiments are carried out on UO_2 and the vapour species observed are U, UO, UO_2, UO_3 and O and their partial pressures are measured over a temprature range of 3,300 K to 5,500 K in which the total pressure changes from 0.05 MPa to 6.5MPa.

4.6.7 Studies on Vaporization behaviour of TeO_2 and MnTe

To generate thermodynamic data which could lead to a better understanding of the role played by tellurium in fuel-clad chemical interactions in fast reactors, Knudsen effusion mass spectrometric studies on compounds of tellurium were continued. Vapourization behaviour of solid TeO_2 and solid MnTe were investigated.

4.6.8 Studies on Phosphate Ceramics for the Use in Nuclear Waste Disposal

A series of phosphate ceramics of the formula A $M_2(PO_4)_3$ [A=Ca, Sr; M = Ti, Zr, Hf and Sn] belonging to the calcium titanium phosphate family have been synthesised and their thermal expansion behaviour investigated. These compounds are candidate matrices for nuclear waste immobilization. The thermal expansivity and expansion anisotropy of the compounds in the temperature range of 300 - 1273 K have been found to depend on the nature of the metal ions in the interstitial space and structural framework. The observed expansion behaviour has been explained in terms of the crystal chemistry of the compounds.

4.6.9 Determination of Boron and Antimony in Sodium

The conventional sodium distillation method(sodium distilled off in vacuum and the residue after suitable dissolution analysed for the impurities) was found to be suspect for the determination of boron. Standard addition studies did not yield quantitative recoveries. Hence an alternative method was standardised which involves dissolution of sodium in high purity water under argon atmosphere and the resulting sodium hydroxide solution was passed through an ion exchange column to selectively remove the B and Sb from the sodium. The eluate was concentrated and analysed by ICP-MS. The method was standardised using the standard addition technique. The detection limits of the method for B and Sb was found to be 0.1 μg/g of sodium. Primary sodium samples received from FBTR were analysed using the above method.

4.7 Fast Reactor Reprocessing

Construction of fast reactor fuel reprocessing plant (1 kg/d) is in progress. Reprocessing of experimental thermal reactor irradiated thorium rods is being started for recovery of U^{233}.

4.7.1 Reprocessing Flowsheet Studies

The first core of FBTR uses 70% PuC-30% UC. Reprocessing of this fuel with high Pu content requires special attention and computer simulation becomes essential. Towards this, **SIMPSEX** (**SIM**ulation **P**rogram for **S**olvent **EX**traction) code has been developed, which has an efficient model for PUREX distribution equilibria, based on activity considerations for key components and can handle 13 components. As data required for thermodynamic analysis is available only for 30% TBP/diluent system, SIMPSEX model is applicable to only to 30% TBP/n-dodecane. The code was used for the analysis of various U/Pu ratios enabling the prediction of concentration profiles for FBTR and PFBR fuels. Fig 16 shows a comparison between experimental, SIMPSEX and SEPHIS simulated Pu stage profiles for a typical run.

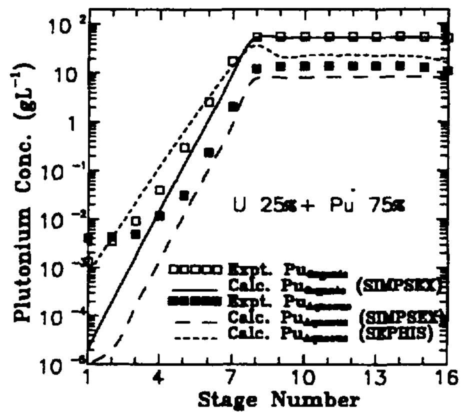


Fig 16 Prediction of Pu stage profiles

4.7.2 Prediction of Densities of Aqueous Solutions Containing Electrolytes

It was observed that the densities of aqueous solutions containing $UO_2(NO_3)_2$, $Pu(NO_3)_4$ and HNO_3 could be correlated better as a function of individual solute concentrations and the coefficients in turn can be correlated as a function of temperature. The published data containing about 705 points with Solute concentrations - U : 0 - 604x10^3 kg.m^{-3}, Pu: 0 - 730x10^3 kg.m^{-3} and nitric acid 0 to 16 kmol m^{-3} and temperature range of 283.15 K to 348.15 K were correlated with an error of 0.75%.

4.7.3 Removal of Solubilised Extractant from Aqueous Streams

Experimental studies for the removal of dissolved TBP in aqueous solutions by adsorbing on a fixed bed containing Amberlite XAD-4 resin were conducted. Break through curves were established for different flow rates and feed concentrations of TBP in aqueous solutions. Break through capacity, saturation capacity and mass transfer zone length (MTZ) were estimated and the MTZ length was correlated. The distribution data of TBP on resin were measured and the equilibrium data were fitted to Freundlich isotherm model.

4.7.4 Development of Fluidic and Metering Devices

Fluidic devices are gaining more importance in reprocessing plants and works have been done on vortex diodes, by which , a resistance ratio of about 50 has been achieved so far. Design codes are developed for the design of reverse flow diverter (RFD) pumps and constant volume feeders are developed for very low capacity (1-5 l/h) using bent pipes and cylindrical buckets.

4.8 Safety Research

4.8.1 Sodium Concrete Interaction Studies

An experimental run on sodium-concrete interactions was carried out by bringing 2 kg of sodium at 773 K in contact with limestone concrete block (area exposed to sodium 190 sq.cm, thickness 175 mm and mass of 7 kg) for a duration of 45 min. The concrete block was kept in a leak tight vessel of 20 l capacity under argon atmosphere. Hydrogen release reached a value of 8% (by vol) over a period of about 77 min. and maximum rise in pressure of the chamber was about 70 kPa over the duration of the experiment. Ultrasonic investigations of post-test specimen indicated internal damage upto 95 mm of the thickness. However, no gross axial or radial deformation of the block was noticed. Characterisation of sodium concrete reaction products as a function of depth of sodium penetration in concrete is in progress.

4.8.2 Air Cleaning Systems for Sodium Fire

An experimental set up of a submerged bed scrubber has been designed (with provision of 1.3 l bed volume and about 6 l water and with overall size of 220 mm dia 560 mm height) for trials on its performance of removal of sodium aerosols in sodium fire containments. Two experimental runs were carried out by burning about 100 gm of sodium in one and 150 gm in another. The sodium fire duration was about 90 min. and throughout the period air flow rate was maintained at 20 l/min. For pebble bed size of 8 mm to 13 mm, the sodium aerosol removal efficiency was in the range of 85 to 90%.

4.8.3 Simulated Fuel -Coolant Interaction Studies

Experimental trials were carried out with tin-water, lead-water and zinc-water systems, keeping the melt temperature at 873 K. The melt to water mass ratios were 0.1 and 0.3. The results brought out the suppressive role of melt surface tension and viscosity on the explosivity.

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A REVIEW OF FAST REACTOR PROGRAMME IN JAPAN

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Abstract

This report describes the development and activities on fast reactor in Japan for the period of April 1996 - March 1997. During this period, the 30th duty cycle operation has been started in the Experimental Fast Reactor "Joyo". The cause investigation on the sodium leak incident has completed and the safety examination are being performed in the Prototype Fast Breeder Reactor "Monju". The three years design study since FY1994 on the plant optimization of the Demonstration FBR has been completed by the Japan Atomic Power Company (JAPC).
Related research and development works are underway at several organizations under the discussion and coordination of the Japanese FBR R&D Steering Committee, which is composed of Power Reactor and Nuclear Fuel Development Corporation(PNC), JAPC, Japan Atomic Energy Research Institute(JAERI) and Central Research Institute of Electric Power Industry(CRIEPI).
In November 1996, the Japan Atomic Energy Commission(JAEC) established a Social Gathering Meeting to discuss generally the significance of FBR development in Japan for the future.

1. GENERAL REVIEW

(1) Aiming to discuss various problems of the basis of the policy on the nuclear energy in Japan, and to reflect various opinions of the nation to future policy on research, development and utilization of the nuclear energy, JAEC decided to institute the round-table meeting in March, 1996, and the discussion was started from April. The total of 11 meetings were held during April to September, 1996. Following subjects and matters were discussed in these meetings:
- Safety ensuring of the nuclear energy including disaster prevention
- Total energy and the nuclear energy
- Nuclear fuel recycling
- Relation between the nuclear energy and the society

(2) In succession to the above meeting, the Social Gathering Meeting was established in November 1996 by JAEC to discuss generally the significance of FBR development in Japan for the future. This Meeting will also discuss how to deal with Monju including its roles for the development of FBR.

(3) On the status of Joyo and Monju, Joyo started the 30th cycle operation in March 1997 after the completion of the 11th periodical inspection. Subsequent to the sodium leak accident occurred on December 1995 in Monju, PNC carried out investigations of the cause of the accident which were completed in March, 1997. A total review of the Monju plant is now being performed in order to improve its safety.

(4) As for the demonstration fast breeder reactor (DFBR-1) of Japan, the three years design study since FY1994 on the plant optimization of the DFBR-1 has been completed by JAPC. Related research and development works are underway at several organizations under the discussion and coordination of the Japanese FBR R&D Steering Committee.

(5) Recent major emphases on the PNC's R&D are placed on the research and development of technologies specific to FBRs aiming at establishment of FBR technological system by the year of about 2030. R&Ds are being conducted in the following fields:
- core and fuel
- high temperature structural design
- safety

2. EXPERIMENTAL FAST REACTOR JOYO

2.1 General Status

This report covers the activities of Joyo from April 1996 Through March 1997. The operating history of Joyo is shown in Fig. 2.1.

The 11th periodical inspection of Joyo has finished on March 24th 1997. In this period enforcement of some countermeasures against sodium leak has been carried out based on the lessons learned from the sodium leakage accident of Monju, as follows:

(1) Modification of the sodium leak detection system
(2) Modification of the fire alarm system
(3) Setting up of building interior monitoring function(Setting industrial TV camera)
(4) Replacement of the thermocouples with compression fitting type
(5) Installation of additional smoke dampers

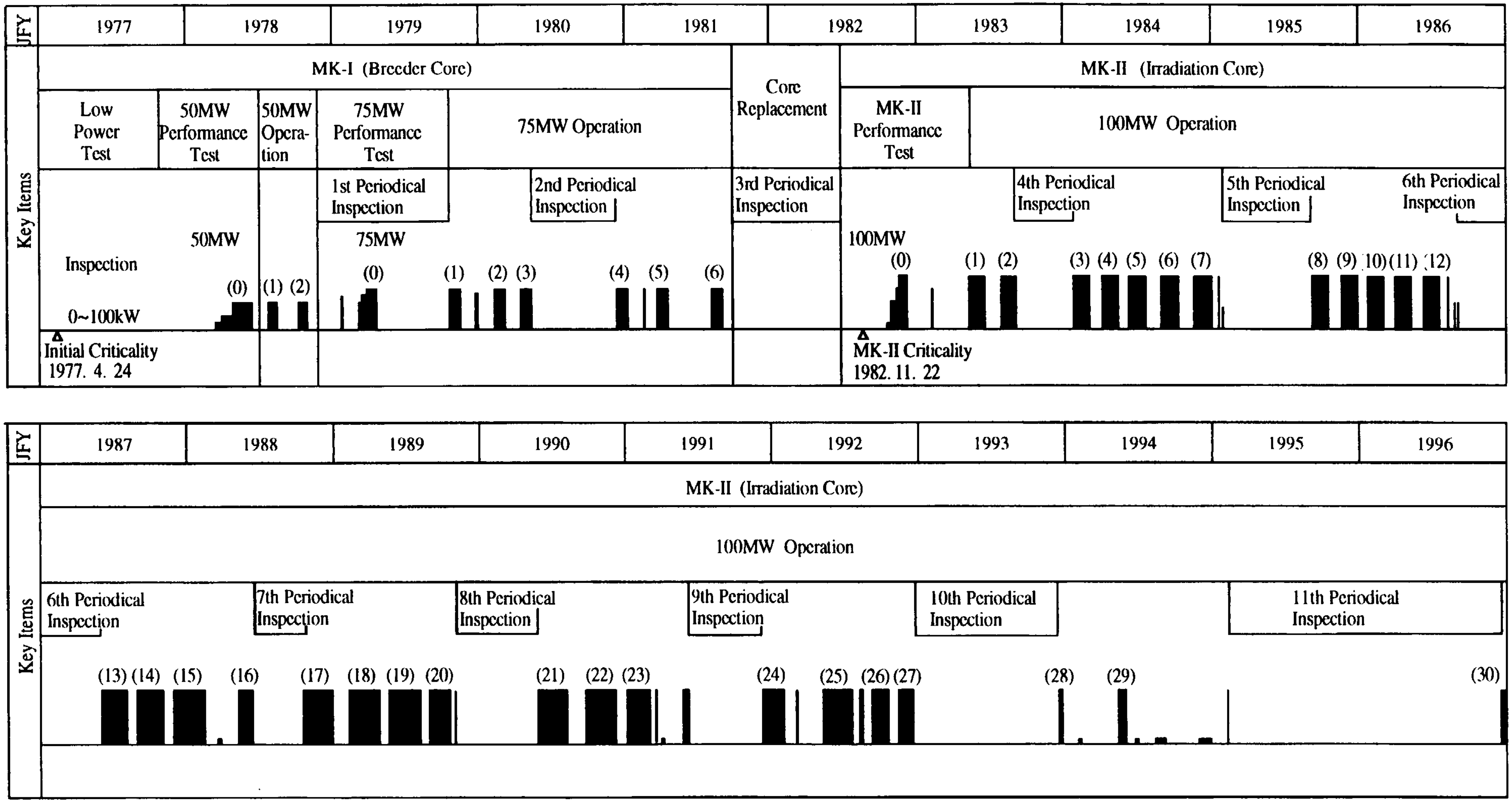
Joyo started the 30th cycle operation on May 24th.

The core configuration is shown in Fig.2.2. The irradiation test in the 30th cycle is summarized as follows:

(1) The creep test of fuel cladding materials under irradiation with MARICO(Material Testing Rig with Temperature Control).
(2) Sodium bond type control rod test.
(3) New fuel test(mixed-carbide fuel, nitride fuel)

2.2 Upgrading Program of Joyo(MK-III program)

The Joyo upgrading program named the MK-III program is planned to improve its irradiation capability. The safety review for the MK-III program by government was completed in September 1995. The plant upgrading schedule will be discussed because of the influence of the Monju secondary sodium leak accident.



(as of March, 1997)

Figure 2.1 Operating History of Experimental Fast Reactor JOYO

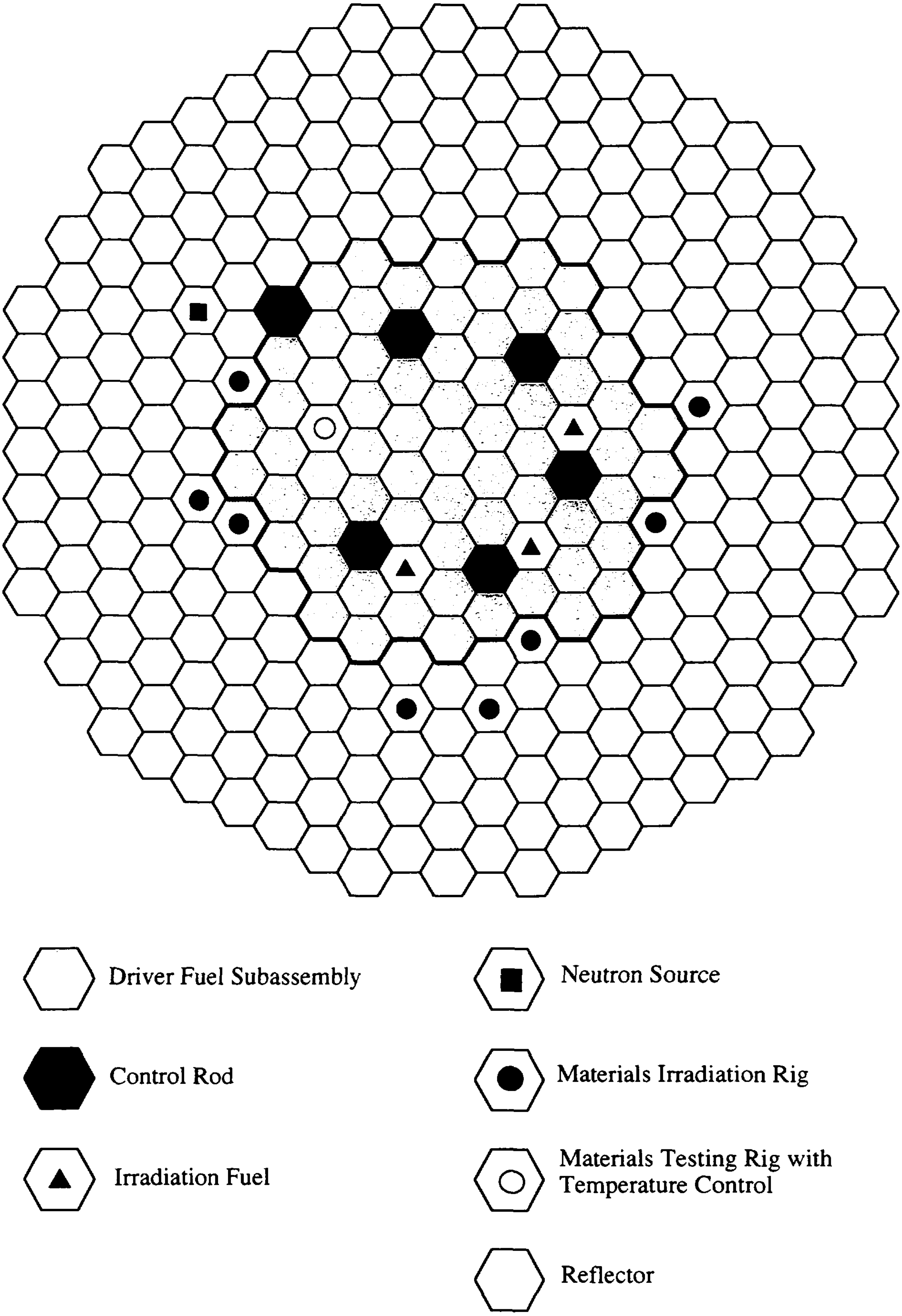


Fig.2.2 Core Configuration for Nuclear and Plant Characteristics Test

3. PROTOTYPE FBR, MONJU

3.1 Introduction

Monju is a prototype fast breeder reactor designed to have an output of 280MWe (714MWt). Monju is fueled with mixed oxides of plutonium and uranium and cooled by liquid sodium. The principal data on plant design and performance are shown in Table 3.1.

The construction of Monju began in 1985 on the north side of the Tsuruga Peninsula in the central Japan, facing the Sea of Japan. The FBR is now being developed to play a major role in Japan's future nuclear power program. Monju achieved initial criticality in April 1994, and generating electricity and connecting to the grid was started in July 1995.

3.2 Outline of Accident

On December 8, 1995, a leakage of sodium occurred in the piping room (C) of the Secondary Heat Transport System (SHTS) while the output of the reactor was being raised for a plant trip test at 40% output as part of a series of performance tests. The nuclear reactor was shut down manually after the accident, and sodium was drained from the SHTS in which the accident occurred and also from the Loop C of the Primary Heat Transfer System (PHTS). The plant is currently in a low-temperature shutdown state. The plant conditions of Monju at the time of the sodium leak occurrence are shown in Fig. 3.1.

As a result of investigations conducted at the location of the leakage, it was found that about 0.7 ton of sodium had leaked out from a breakage at the tip of a thermocouple well installed on the Intermediate Heat Exchanger (IHX) outlet pipe of the SHTS/Loop C (see Fig. 3.2).

There was no effect on the general public or employees due to radioactive materials.

3.3 Cause Investigations

Cause investigations of the sodium leak accident have been carried out. The schedule of the cause investigations is shown in Fig. 3.3.

The outline of the investigation works is as follows;

(1) Reasons why only one thermocouple well was affected
The results of the investigations concluded that the breakage of the thermocouple well was caused by the high frequency fatigue as a result of flow-induced vibration parallel to the direction of sodium flow. The thermocouple well broke at the point of reduction of radius. It has also been clarified that the design of the thermocouple well was inadequate in that it did not take into account the phenomenon of parallel vibration.
However, since no other thermocouple well in the Monju secondary loops displayed any

evidence of damage, an investigation was launched into the reasons why only well was affected. Using a water rig, the factors leading to flow induced vibration were examined. These water rig tests showed that if the thermocouple sheath inside the thermocouple well was bent, then the damping of the well vibrations was reduced and their amplitude was increased. The effect was most severe in the case where the thermocouple sheath was bent at the point of reduction of radius of the well. The thermocouple sheaths used were found to have rubbing marks at the points of contact with the well. Similar marks were found only on the sheath of the Monju thermocouple in the well which failed in the accident. It was therefore concluded that the reason for only one thermocouple well having failed was that its thermocouple sheath was initially bent.

(2) Evaluation of the Damage caused by the Lost Part

The tip of the thermocouple well which broke off, the so-called "lost part", was carried by the sodium flow to the superheater inlet distributor where it lodged and from which it was later recovered. The collision points of the lost part with the internals of the loop were deduced and the damage to the loop was estimated from the corresponding damage to the surface of the lost part. It was concluded that the integrity of the loop had not been impaired.

(3) Effects of Sodium Aerosol on Equipment and Structures

All the mechanical, electrical and electronic equipment to which sodium aerosol had adhered was cleaned (see Fig. 3.4). Visual inspection and electrical testing were carried out as appropriate to ascertain whether any deterioration had occurred. It was found that there had been no damage to mechanical structures as a result of exposure to aerosol. After comprehensive cleaning and rebuilding, the function of electrical and electronic components was found to be unimpaired but this will remain subject to a program of continuing inspection and monitoring.

(4) Effects of the Sodium leak on the Floor Liner and other Nearby Structures

Tests on high temperature chemical corrosion were carried out and the results were compared with the damage from the Monju accident. It was concluded that the damage to the ventilation duct and walkway grating and the thinning of the floor liner were caused by Na-Fe compound oxide type corrosion.

(5) Clarification of the Damage Mechanism of the Floor liner and other Structures

Two mock-up tests were carried at the O-arai Engineering Center(OEC), using full-scale simulation of equipment around the leak site, to examine the sodium leak combustion behavior and damage mechanism. The first test was terminated early due to exhaust gas filtration problems. In the second test (Sodium Leak Combustion Test II) the floor liner was perforated; this had not occurred in the Monju accident.

A series of tests was therefore carried out including: chemical analysis and X-ray diffraction of the combustion residues and metal structure microscopy of the floor liner. The corrosive thinning mechanism of the liner in the Monju accident and Sodium Leak Combustion Test-II were compared. An important aspect was the measurement of the moisture released from heated concrete, since the quantity of moisture determines the concentration of sodium hydroxide (NaOH) which is a major factor in corrosion. In addition, a chemical thermodynamics examination of the corrosion mechanism was also carried out.

It became clear that there was little moisture present during the period of combustion in the Monju accident and in Sodium Leak Combustion Test-I (which was carried out in a steel cell without concrete). In both these cases, the deposit on the floor was mainly sodium oxide (Na_2O) with very little NaOH. Hence, a high temperature Na-Fe compound oxide type corrosion occurred by reaction between the Na_2O and the steel liner plate (Fe), resulting in a small and uniform thinning of the liner. By contrast, because the volume of the concrete cell used in Sodium Leak Combustion Test-II was small, the temperature became higher and a larger amount of water was released from the concrete. The proportion of NaOH in the molten pool (Na, Na_2O, Na_2O_2) on the floor liner was therefore also higher. Sodium peroxide (Na_2O_2) in the pool dissociated to give the peroxide ion (O_2^{2-}) and molten-salt type corrosion occurred on the steel floor liner plate. Thus, the corrosion mechanism in the Monju accident was completely different from that in Sodium Leak Combustion Test-II. In laboratory tests the corrosion rates of the Na-Fe compound oxide type corrosion and the molten salt type corrosion, were compared; it was shown that the rate of molten salt type corrosion was approximately five times higher (see Fig. 3.5).

3.4 Safety Improvement

In order that the countermeasures against secondary sodium leaks are effective, PNC will carry out the following research:

(1) Sodium Leak Combustion Behavior

In addition to sodium leak combustion behavior essentially based on the current assumption of the largest scale pipe rupture, the combustion behavior of medium and small scale leaks is to be thoroughly examined. An experimental program focused on the chemical composition of sodium combustion products (aerosol and residue deposits) is also to be completed.

(2) Integrity of the Floor Liner and other Structures

The integrity of the floor liner in the case of thermal expansion of its total area was confirmed at the Monju design stage; however, an examination of its integrity in the case of localized temperature rises during medium-scale and small-scale leaks will now also be performed. Further work is required on damage due to high temperature corrosion by combustion residues.

(3) Safety Evaluation

Up to now the safety evaluation of a secondary sodium leak accident has been based on the severest temperature-driven pressure rise, assuming the largest scale of sodium leak. This was chosen since the loss of integrity of the building due to the pressure rise in the room has a critical effect on loop separation and therefore accident limitation.

Table 3.1 Principal Design and Performance Data of Monju

Reactor type	Sodium-cooled / loop-type	Reactor vessel	
Number of loops	3	height / diameter	18 / 7 m
Thermal output	714 MWt	Primary coolant systems	
Electrical output	280 MWe	Coolant sodium mass	760 ton
Fuel material	PuO_2-UO_2	Inlet / outlet reactor temperature	397 / 529 ℃
Core dimensions		Coolant flow rate	5.1×10^6kg /h / loop$\times$3loops
Equivalent diameter	1,790 mm	Coolant flow velocity	6m/s(inlet), 4m/s(outlet)
Height	930 mm	Secondary coolant systems	
Plutonium enrichment (inner core / outer core)		Coolant sodium mass	760 ton
	(Pu fissile %)	Inlet / outlet IHX temperature	325 / 505 ℃
Initial core	15 / 20	Coolant flow rate	3.7×10^6kg /h / loop$\times$3loops
Equilibrium core	16 / 21	Coolant flow velocity	5 m/s
Fuel inventory		Water - steam systems	
Core (U+Pu metal)	5.9 t	Feed water flow rate	113.7×10^4kg/h
Blanket (U metal)	17.5 t	Steam temperature (turbine inlet)	483 ℃
Average burnup	80,000 MWD/ t	Steam pressure (turbine inlet)	12.7 MPa
Cladding material	SUS316	Type of steam generator	Helical coil
Cladding outer diameter/thickness	6.5 / 0.47 mm	Refueling system	Single rotating plug
Blanket thickness			with fixed arm FHM
Upper / lower / radial	30 / 35 / 30 cm	Refueling interval	6 months
Breeding ratio	1.2		

However, in future, the integrity of the floor liner and other structures must be confirmed for all sodium leak rates and ventilation conditions, focusing on the temperature rise of the floor liner and its thinning by corrosion.

4. DESIGN STUDY OF DFBR

JAPC conducted for the past several years conceptual design studies of the DFBR-1 in accordance with the basic policy of the Federation of Electric Power Companies (FEPC) and 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 in the early 2000's, and the major specifications.

The optimization study of the DFBR-1, which was conducted by JAPC for three years since FY1994 based on the major specifications and those design studies, has completed. These studies were focused on core safety enhancement including the application of gas expansion modules (GEM), increase of design margins for the structural tolerance of the containment facilities, feasibility and licensability of seismic isolation plant for introduction to the DFBR-1. Through these studies, the goal was to make the design of the whole DFBR-1 plant harmonious with both safety and economic viability, and to prove preparation for the basic design.

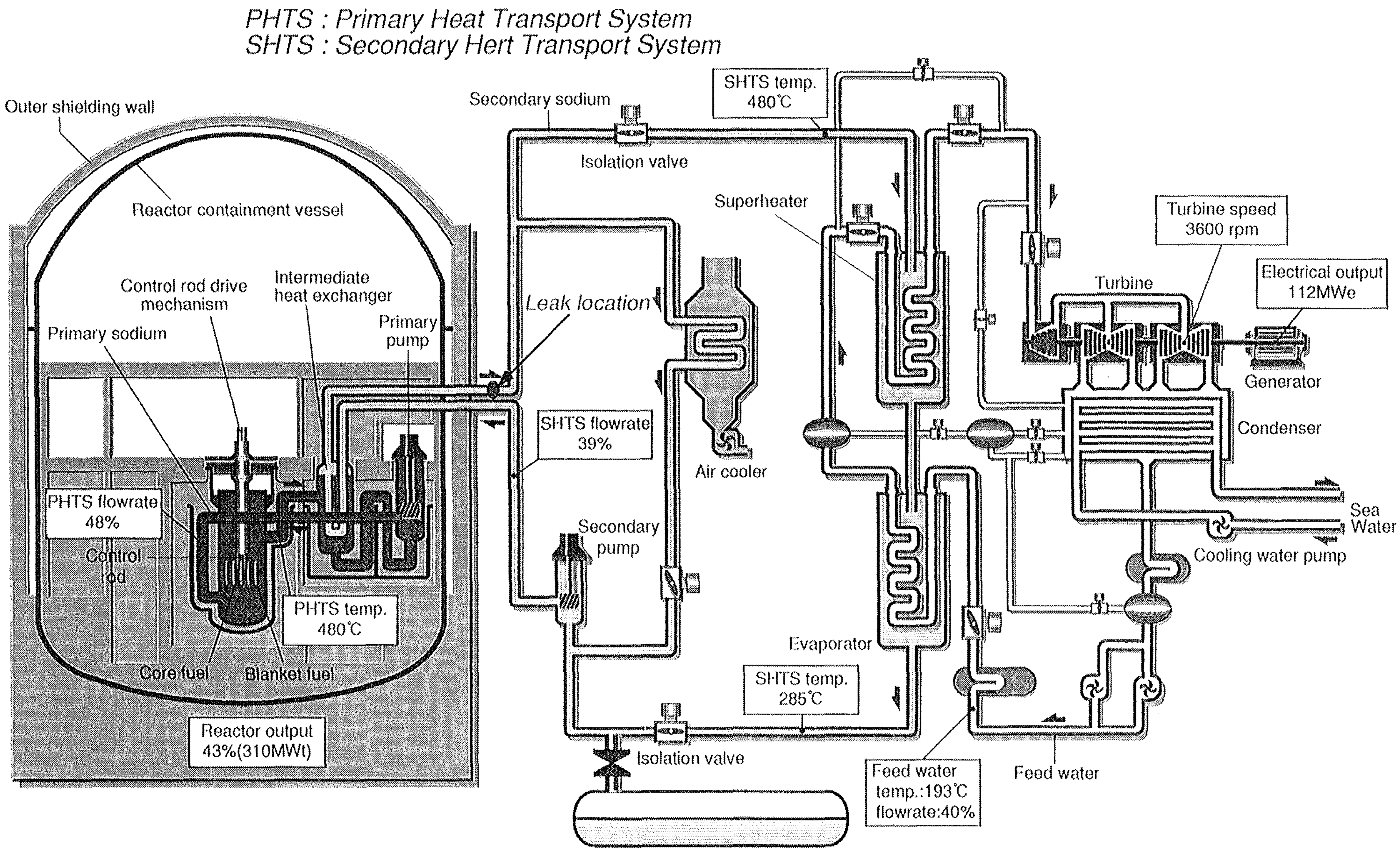


Fig. 3.1 The State of Plant before the Occurrence of the Accident

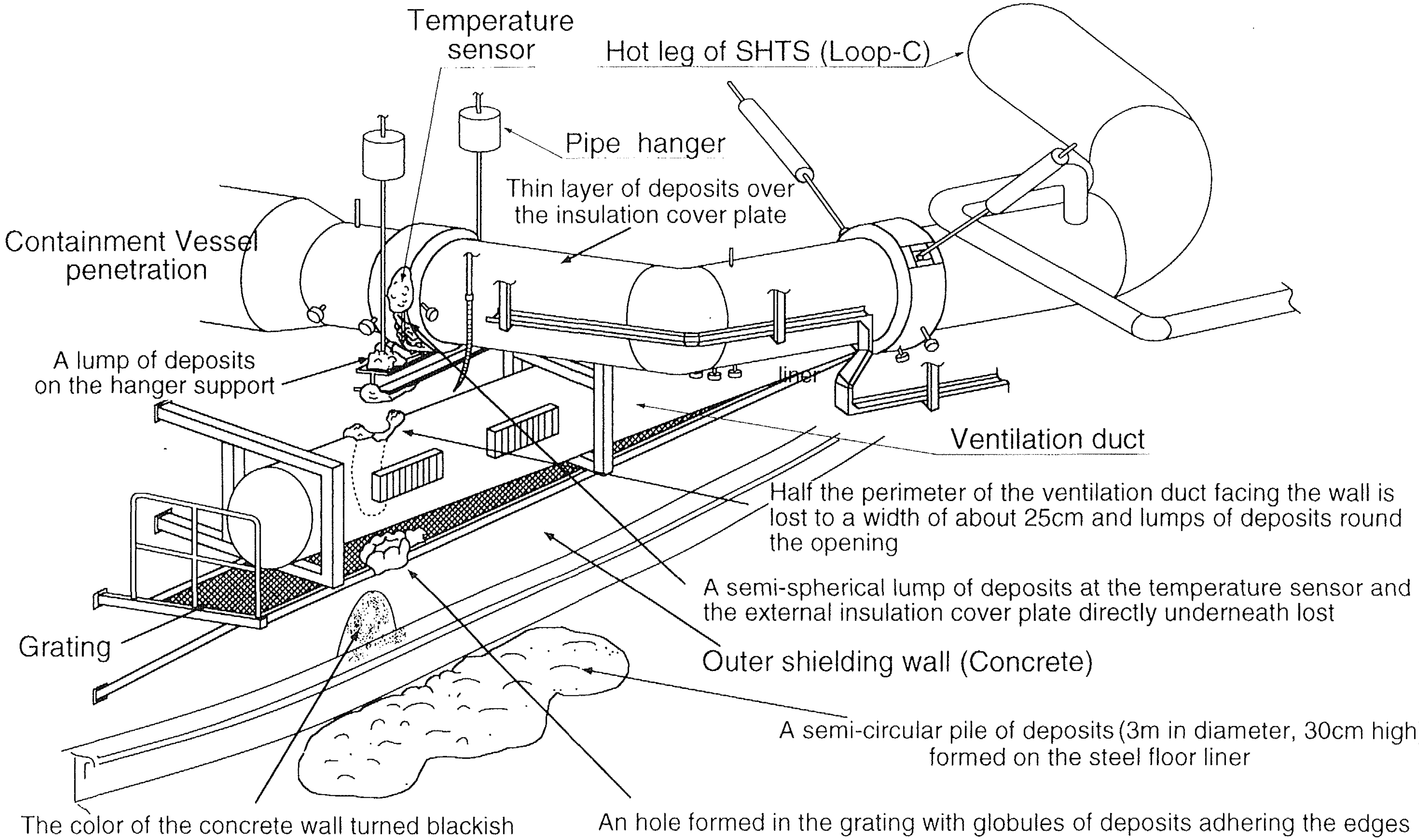


Fig. 3.2 Sketch of the Affected Area

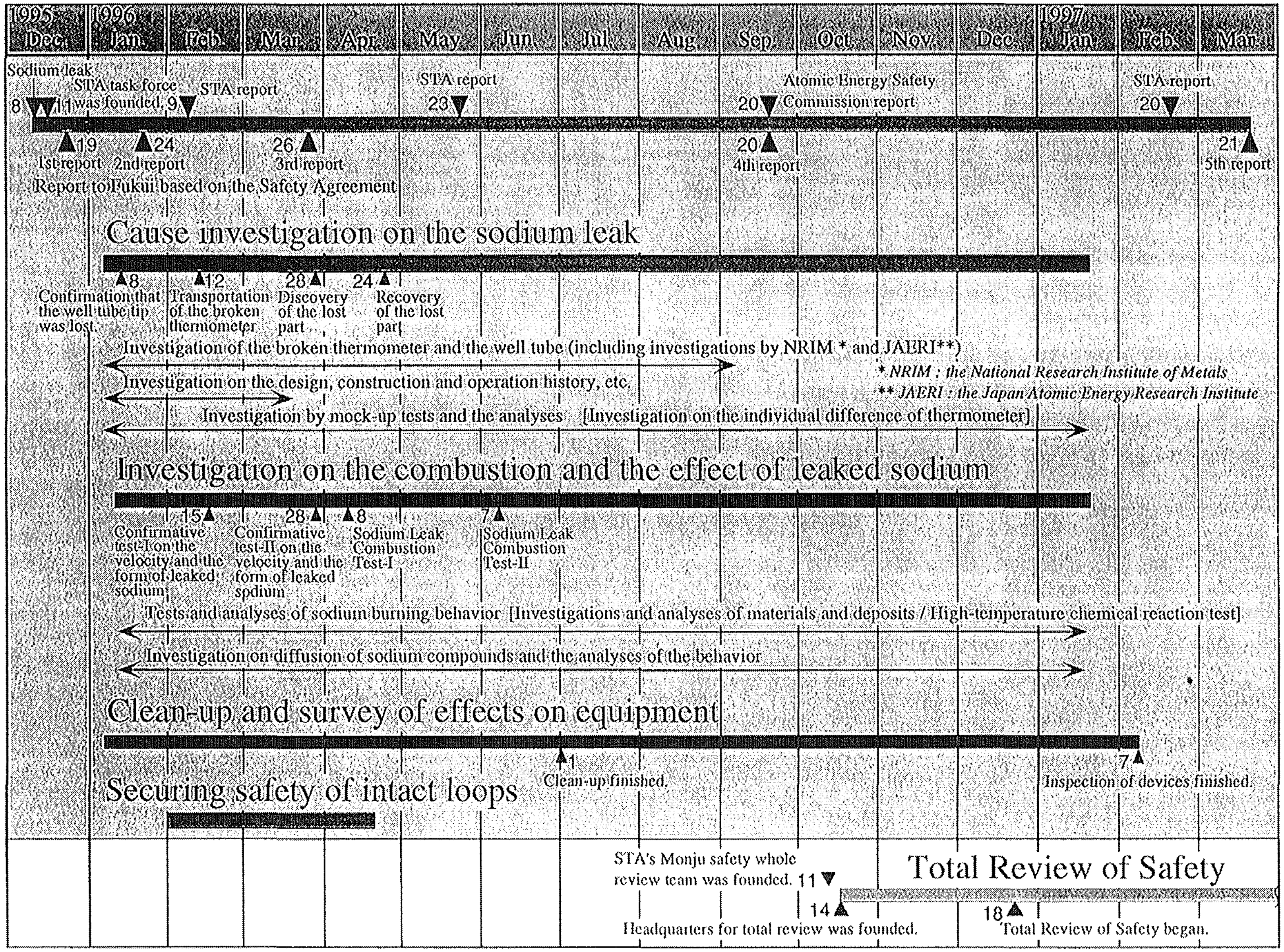
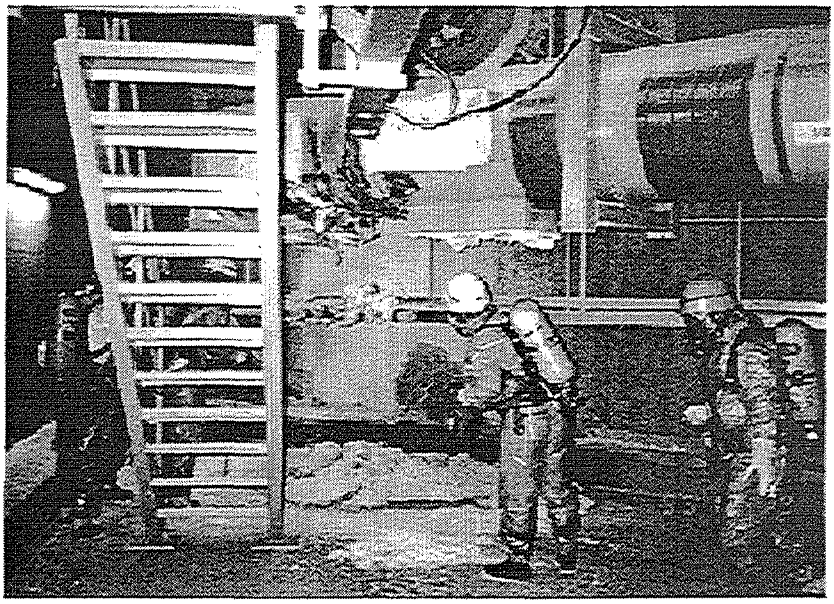


Fig. 3.3 Schedule of the Cause Investigation

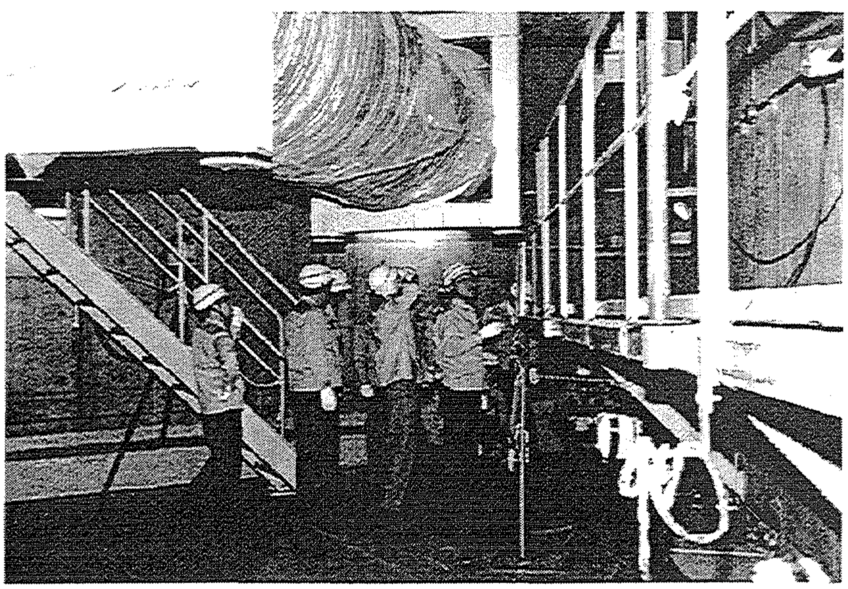
Just after the accident (photographed on December 9, 1995)



Sodium compound was piled up on the floor liner. At the same time, it spread to many rooms as an aerosol and adhered to the floor and the wall, etc.

Clean-up

The present condition



Clean-up finished in July, 1996. Now it is possible to visit and work at the leak site without protective clothing.

Fig. 3.4 Clean-up of the Floor and the Wall

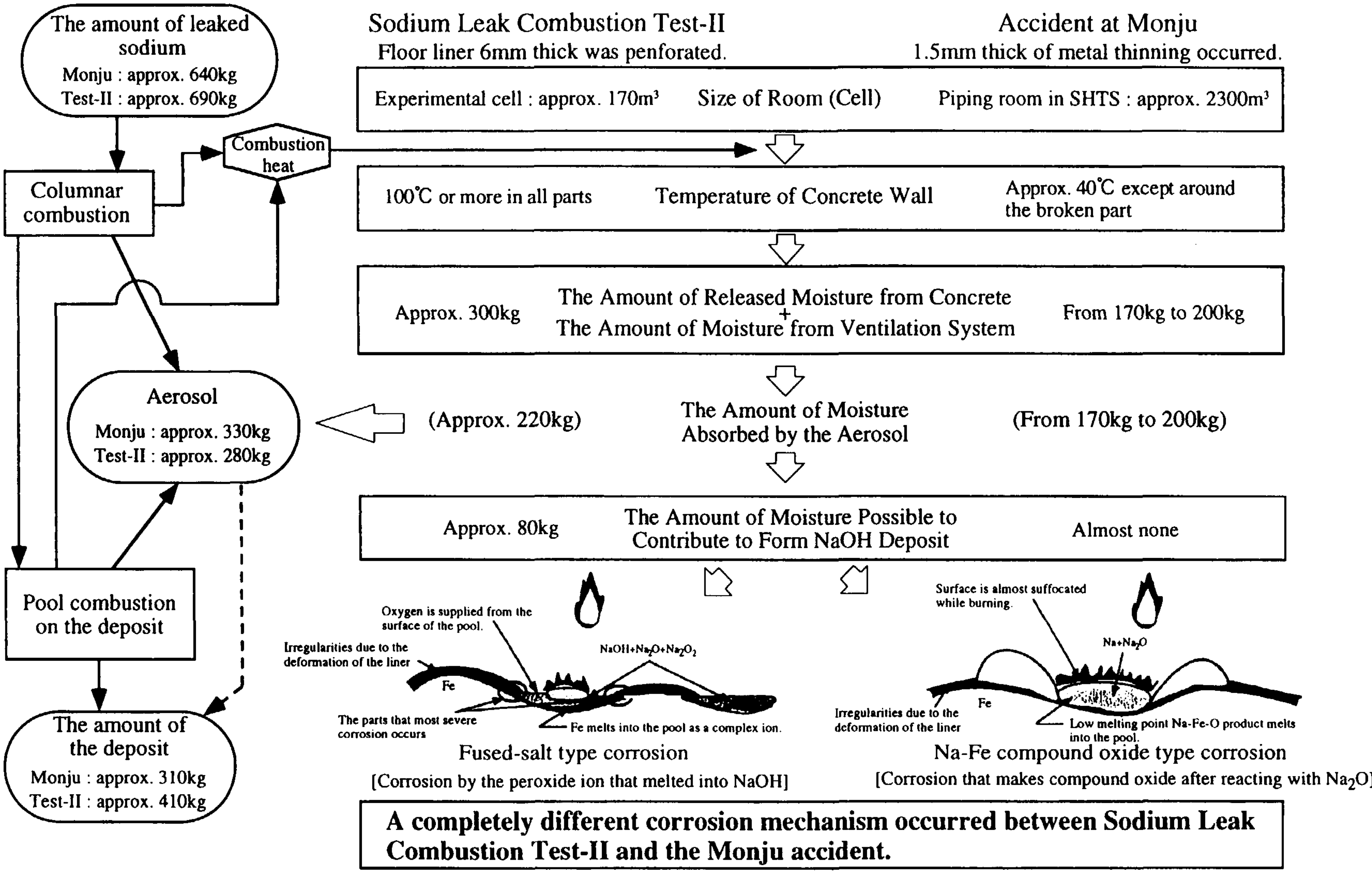


Fig. 3.5 Comparison of the Damage to the Floor Liner between Sodium Leak Combustion Test-II and the Monju Accident

5. REACTOR PHYSICS

5.1 Development of Analytical Method

The effort to develop a neutron transport code based on an improved nodal method has been continued in order to treat the Hex-Z geometry of FBR cores more accurately. In order to reduce truncation error of the code, a new method to treat the radial leakage has been developed, in which the distribution of node boundary fluxes is obtained from local two-dimensional flux distribution. The local flux distribution is evaluated from average fluxes at surrounding nodes and node boundaries by second-order polynomials. An FBR core model with extremely large-sized assemblies was calculated by the new method and the results were compared with those of a reference Monte Carlo calculation. While the previous method overestimated the criticality by 0.3% dk for a control rod-insertion case, the new method agreed well with the Monte Carlo result.

The continuous-energy Monte Carlo method, which has no analytical modeling errors in principle, has become more realistic in Japan because of both the release of an efficiently vectorized code and the rapid improvement of computer hardware. To enhance the applicability of Monte Carlo calculation, the theory of Monte Carlo perturbation method was investigated and successfully introduced into a Monte Carlo code. Although the perturbation method seemed to be promising for the calculation of small reactivity changes, there was room for further improvement about its stability and accuracy.

5.2 Critical Experiment and Analysis

In the core design of large FBRs, it is essential to improve the prediction accuracy of nuclear characteristics from the viewpoint of both reducing construction cost and insuring plant reliability. Extensive work is being performed in this context to accumulate and evaluate many results of reactor physics experiments in FBR field. As a part of the effort to develop a standard data base for large FBR core nuclear design, the physical consistency of JUPITER experiment and analysis was evaluated by full use of sensitivity analysis, effect of different nuclear data libraries and application of most-detailed analytical tools.

As a conclusion, JUPITER experiment and analysis was found to possess sufficient consistency on the whole, especially for the prediction of criticality, space-dependency of C/Es in core region, and sodium void reactivity, which were persistent problems in the past JUPITER evaluations. It was also recognized that there is, however, some room for further improvements about the C28/F49 ratio, reaction rate distribution in blanket region, and Doppler reactivity. Efforts are now being conducted from various viewpoints such as re-evaluation of experimental and analytical errors, application of new most-detailed analytical tools, comparisons with other experimental cores, and refinement of statistical tests for physical consistency.

To enhance the accuracy and reliability of large FBR design work in Japan, the quantitative knowledge obtained from the JUPITER experiments and analysis will be utilized as one of the most important databases for FBR core physics.

5.3 Cross-Section Adjustment

At PNC, a three-year project has been launched to prepare a new unified cross-section set for the use in the design work of Japanese demonstration FBR. The word 'unification' implies the combination of integral information such as critical experiments and analyses with the differential nuclear data in the Bayesian sense.

The special features of the unified cross-section are: (a) use of the latest JENDL-3.2 library-based cross-section set and corresponding covariance data, (b) application of various integral parameters besides JUPITER data, such as MOZART and FCA critical experiments, JOYO and MONJU reactor data which include burnup and temperature characteristics, and (c) extension of adjusted nuclear data to self-shielding factors and scattering matrix components.

The number of integral parameters prepared for the cross-section adjustment is increased to more than 200 from the previous 80 data of JUPITER. The preliminary results of adjustment by the whole integral data set showed quite a good consistency and reliability.

5.4 Shielding Experiment and Analysis

The development of a standard database for FBR shielding design was continued. In the database, the experimental and analytical information from JASPER and other shielding experiments is evaluated and compiled in a systematic and consistent manner.

As a part of the efforts, typical axial shielding experiments of JASPER are now being re-analyzed with the latest version of nuclear data library, JENDL-3.2, and compared with the results of the former JENDL-2. Further, a cross-section sensitivity code, SWANLAKE, was applied to survey the cause of the difference between the two JENDL libraries.

5.5 Study on Pu and MA Burner Core

One of the distinctive features of a fast reactor is its good neutron economy. Utilizing the excess of neutrons enables us to construct flexible cores such that they breed or burn plutonium in consideration of plutonium balance, incinerate MA and long lived fission products for reducing radio toxicity and improve safety.

There has been increasing focus on the research and development work necessary for the utilization of the excellent Pu burning characteristics of fast reactor cores. Studies on Pu burner fast reactor core have been performing to show the flexibility of plutonium utilizing characteristics of a fast reactor.

The following three approaches to burning plutonium efficiently in a fast reactor are considered :

(1) Enhancement of neutron leakage (high Pu enrichment MOX core)
(2) Introduction of neutron absorption material (high Pu enrichment MOX core)
(3) Core without uranium

Series of analyses were performed to investigate the basic characteristics of Pu burning in a fast reactor by changing various parameters including fuel pin specifications, smear density, core height, Pu vector, types of inert matrix without U, etc.

Highly enriched mixed oxide (MOX) fuels and Pu fuels without uranium were considered for Pu burning enhancement. It was found that Pu consumption rates essentially depend on Pu enrichment. Both burnup reactivity loss and Doppler coefficient are important criteria for highly enriched MOX fuel cores. Cores without uranium were found to consume the Pu at a very large burnup rate close to the theoretically maximum value of 110-120 kg/TWhe. The introduction of UO_2 in an internal blanket is effective in enhancing the Doppler coefficient, it causes a minor increase in the sodium void reactivity in non-uranium cores.

Some of the MA nuclides (Np, Am, Cm) contained in residual waste from reprocessing have extremely long-term radio toxicity. Means of reducing the radio toxicity of the MA nuclides are presently under investigation. The MA nuclides could produce useful energy if converted into short-lived fission products by neutron bombardment. From this standpoint, a nuclear reactor provides the obvious means for transmutation of MA nuclides. Among the various nuclear reactors, a fast reactor is considered to have the greatest potential to transmute MA effectively, because of its hard neutron spectrum.

Feasibility studies have been performed to investigate the basic characteristics (transmutation rate, burnup reactivity, Doppler coefficient, sodium void reactivity, maximum linear heat rate, etc.) of a fast reactor core with MA transmutation, the following items were considered:

(1) Study on loading method of MA in the core (homogeneous, heterogeneous, hybrid, blanket, etc.)
(2) Selection of fuel material for MA transmutation (oxide, inert matrices such as Al_2O_3, CeO_2, etc.),
(3) Study on the maximum tolerable amount of rare earth (RE) nuclides,
(4) Effect of MA recycling on core characteristics and fuel cycle system.

MA transmutation in a fast reactor core has no serious drawbacks in terms of core performance, provided that the homogeneous loading method can be employed with a small fraction of MA fuel (~5wt%).

Fast reactors have a strong potential for burning Pu and MA effectively.

5.6 Cross Section Measurement of MA and Rare Earth Nuclides

In the reactor transmutation studies on long-lived radioactive waste, nuclear data for MA nuclides and fission products are of primary importance. However, nuclear data for many MA nuclides are still not known to the desired accuracy. Accurate experimental data of neutron cross section for MA are indispensable to establish MA transmutation technology by reactors. Accurate neutron cross section data of RE nuclides become necessary for designing the MA burning core. The data, however, are quite inadequate both in quality and in quantity.

Fission cross section ratios of minor actinide nuclides (Np-237, Am-241 and Am-243) relative to U-235 in the fast neutron energy region have been measured at YAYOI fast neutron source reactor.

As a part of MA nuclear data evaluation, the analysis of a Np-237 sample irradiated in JOYO has been performed. Additional irradiation test of Np-237, Am-241,243 and Cm-244 samples in JOYO was started in August, 1994.

Measurements of keV-neutron capture cross sections of RE nuclides (Sm-147, Sm-148, Sm-150, Ce-140, Pr-141, Eu-153, Nd-143, Nd-145) have been performed to evaluate the accuracy of the nuclear data libraries using the 3-MeV Pelletron accelerator of the Research Laboratory for Nuclear Reactors at the Tokyo Institute of Technology.

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 preparation of system performance test on SASS by use of Joyo is underway.

6.2 Instrumentation

A development of velocity measurement system in sodium by using ultrasound is in progress.

Improvement of the ISI test equipment, which was developed and tested in the mock-up test rigs in OEC and then transferred to Monju from OEC, is now under discussion.

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

6.3 Large Scale Thermal Hydraulic Test

A fundamental design of Large Scale Thermal Hydraulic Test Facility was completed.

7. FUELS AND MATERIALS

7.1 Fuel Fabrication

The PFPF (Plutonium Fuel Production Facility) equipped with automated and remote handling fuel production systems started fabrication of Joyo and Monju fuels from October 1988. Total amount of MOX fuel for Joyo & Monju fabricated in PFPF reached 11 tons MOX.

7.2 Fuel Pin Performance

Fuel pin performance codes of MOX type pellet for analyzing the steady state and transient conditions have been improved since 1984, with the data of Joyo irradiation program and PNC/DOE EBR-II operational reliability testing program. The modification of codes for annular MOX fuel is underway with PFR irradiation data base from UK. A three dimensional deformation code of large fuel pin bundle under irradiation has been modified with out-of-reactor testing and PIE results of Joyo. Development of nitride fuel performance code is also in progress.

7.3 Core Material

Advanced austenitic steels of PNC1520 and PNC1425 have been developed for high burnup FBR fuels. Improved performance than the current PNC316 steel is demonstrated by the out-of-reactor and irradiation testing programs. PNC1520 is to apply Monju advanced core and DFBR core.

7.4 Irradiation Experiments

(1) Joyo

Fuel pin irradiation continues 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 also included in the test.

The fuel subassembly with CEA austenitic stainless steel cladding tubes has also been irradiated in MK-II core since 1988 and reached the final stage at 125GWd/t.

(2) Foreign Reactors

The PIE of fuel subassemblies of PNC316 steel and PNC1520 advanced austenitic stainless steel irradiated in FFTF has been successfully completed. The 2nd shipment of irradiated fuel materials to OEC was conducted to take a further investigation of high burnup fuel behavior.

7.5 Development of Advanced Fuels

Feasibility study of advanced nitride fuel has been conducted since 1986. PNC/JAERI irradiation test of nitride and carbide pins continues in Joyo since 1994.

7.6 Post Irradiation Examination

Construction of PIE facility of Fuels Monitoring Facility (FMF-2) adjacent to existing FMF-1 at OEC completed to handle the large fuel assemblies irradiated in FBR Monju and ATR Fugen. Partial hot operation of FMF-2 will start from late 1997 by testing Joyo fuel assembly.

8. STRUCTURAL DESIGN AND MATERIALS

8.1 Development of Structural Design Method

(1) FINAS nonlinear structural analysis program

Effort to entire the capability 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 engineers 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.

- Creep-fatigue design methods based on elastic analysis

  A new creep-fatigue method, which is based on the concept of a general elastic follow-up model, is being developed. The elastic follow-up equations to predict strain magnification and creep relaxation for structural discontinuities are established.
- Design rules for weldment

  A new design approach, taking into account the metallurgical and geometrical discontinuities inherent in weldment, is being pursued.
- 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 Analysis.

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 loop(STST)

A thermal creep-fatigue test, whose specimen was a new test model "a cylindrical shell with cross-section gradually step-changing" is continued with use of the test facility (STST).

(2) Thermal transient tests in large sodium loop (TTS)
A thermal transient test of a vessel with fillets model could not be continued by a trouble of the sodium valve and a maintenance of TTS.

(3) Distortion tests of fuel sub-assembly duct (CMDT, WFT)
The distortion behaviors of internal structures was started. The fundamental distortion tests of ducts of a fuel sub-assembly were performed with use of new equipment, CMDT and WFT. The distortion tests of several fuel sub-assemblies will be stared 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 in the framework of "IAEA/IWGFR Coordinated Research Program on Intercomparison of LMFBR Seismic Analysis Codes" was completed in 1995. From 1996 PNC joins SYMPHONY program on FBR core seismic tests and analysis at CEA, France.

(3) Seismic structural integrity test rig has been constructed. Piping dynamic test for evaluation of ultimate strength of piping components are under preliminary examination.

8.4 Fracture Mechanics Tests and Evaluation

Both deterministic and probabilistic fracture mechanics methodologies are being development for the integrity assessment of flawed or cracked structure.

Computer codes developed at PNC as CANIS-J for calculation of fracture mechanics parameters, CANIS-G for simplified crack propagation analysis were modified further.

Crack propagation test with a cylinder with an axial temperature gradient is 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 extrapolation 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 9Cr-1Mo steel and FBR grade 316 stainless steel

The creep, fatigue and creep-fatigue tests of modified 9Cr-1Mo steel and FBR grade 316 stainless steel under a long-term life condition were continued to develop the material strength standard for DFBR.

(2) Tests in Sodium and Water/steam

Mechanical strength (fatigue, creep fatigue) tests on 316FR (nitrogen controlled) in sodium are still continued in the program to evaluate the carbon and nitrogen transfer effects. It is resulted based on the creep-data in the sodium that the effect of a sodium environment on a creep strength is negligible.

(3) Tests in Irradiation Environments

Surveillance tests for the Class 1 components of Joyo were conducted 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 mechanical 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.

Coupling of irradiation tests in Joyo and JMTR for evaluation of neutron spectrum effects were started.

(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 fuel subassemblies, the core internal structures and the heat transport systems during the normal operation, scram transients and the early stage of postulated accidents. On this account, thermohydraulic experiments related to the decay heat removal by natural circulation have been carried out, and the development and validation of the thermohydraulic safety analysis codes is also in progress.

(1) Thermohydraulic Experiments

An integral sodium experiment has been carried out with a partial core model composed of seven subassemblies, inter-wrapper gaps, an upper plenum and dipped cooler. The tests for core-plenum and cooling system interaction were almost completed. A series of tests is under way focusing inter-wrapper flow under conditions of natural circulation decay heat removal.

(2) Development And Validation Of Analysis Codes

Subchannel analysis codes, ASFRE for single-phase flow and SABENA for two-phase flow, have been developed for the purpose of predicting fuel element temperature and thermalhydraulic characteristics in the FBR fuel assemblies. ASFRE has the detailed wire-spacer model called distributed flow resistance model, which calculates the effect of wire-spacer on thermalhydraulics. Also planer and porous blockage models are implemented for fuel assembly accident analysis. In this reporting period, three dimensional thermal conduction model was used for the evaluation of local blockage in a fuel assembly. In addition, the comparison of pressure losses in the assembly with the water experimental data has been performed. Regarding SABENA, based on the two-fluid model, no activity is reported.

Multi-dimensional thermalhydraulic code AQUA has been updated in terms of computing time. AQUA has been parallerized and resulting speed-up is as much as a factor of several tens. Practical applications are made during this reporting period. Examples are the analyses of the flow and temperature field of Monju during sodium leakage and fire incident; thermal stratification analysis during manual trip test from 40% rated power condition, etc.

A fluid-structure thermal interaction analysis code FLUSH have been developed to evaluate the temperature distribution in the shield plug. As for the pre-test analysis of Monju performance test, the temperature distribution in the shield plug was successfully predicted by the interactive calculation between structural components and cover gas fluid regions. In this reporting period, boundary-fit coordinates system has been incorporated so that the complex geometry of the shield plug is taken into account.

A three-dimensional arbitrary Lagrangian Eulerian finite element code SPLASH has been applied to the flow-induced vibration analysis of the thermocouple probe. Equations of

structural dynamic motions are developed and coupled with the SPLASH code. This application was made as a part of the investigation study of Monju sodium leakage incident in December 1995.
The whole plant simulation is performed using two computer codes, Super-COPD and SSC. Super-COPD has been applied to the whole plant behavior prediction when a sodium leakage takes place. It is performed to estimate the total amount of leaked sodium during the Monju sodium leakage. It has been confirmed that the sodium leakage rate and the transient course of the sodium level at various vessels, and piping are analyzed as a function of time and change of operation mode. SSC is mostly designed for the safety analysis of FBR plant. A space-dependent plant dynamics model has been developed to apply to the performance evaluation of passive safety features. Three dimensional calculations on neutronic characteristics have been performed to investigate the local power distortion that is expected to occur in the case of non-symmetric behavior of passive shut-down devices. The interface program between neutronic and thermohydraulic calculation is under development.

9.2 Degraded Core Research

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 under preparation. 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. Four distinct modes of interaction behaviors observed in the experiment were characterized fairly well by considering the minimum film boiling temperature. From extrapolation to reactor materials, it is predicted that energetic interactions are unlikely to be met. High-temperature experiments with steel and alumina are planned in the near future.

On the in-pile experiments jointly conducted with French CEA, all the planned tests have completed for the CABRI-FAST program, in which slow and fast transient tests have been conducted, mainly with high burnup, annular fuel pins. A joint synthesis work is underway. Starting in 1996, a next joint in-pile test program, CABRI-RAFT, is initiated, where the total of 7 tests are planned in the CABRI and SCARABEE reactors through 2000.

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 FZK and CEA, a model improvement effort was concentrated on the extended fuel motion behavior after a power transient. A special single channel code PAPAS-2S, which models detailed fuel pin mechanics, is also being elaborated through CABRI analyses. Version 2 of SIMMER-III including the neutronics model was completed and released to the European partners. The first phase of a joint assessment program participated by FZK and CEA was accomplished and its second phase was initiated. 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 being applied to CDA studies of D-FBR and future fast reactors.

Finally, a long-term research program SERAPH (Safety Experimental Reactor for Accident Phenomenology), has been undertaken over the last several years at PNC. 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 study of the new in-pile test facility with related R&D's on key elementary technologies is in progress to achieve required experimental performances, and to establish design concept.

9.3 Plant Accident Research

FBR plant accident research consists of two major activities. One is a study of a non-radiological sodium fire caused by sodium leakage from an intermediate heat transport system (IHTS), and the other is a study of a 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.

Most topical activity in FY1996 is two mock-up tests of the Monju sodium leak carried at OEC. Sodium Leak Fire Test-1 and -2 were conducted by use of SAPFIRE test facility at OEC/PNC to clarify the Monju sodium leak event. In the tests, simulated sodium pipe, thermometer, ventilation duct, grating were placed and sodium was leaked in a steel vessel with a volume of 100m3 in Test-1 and in a 170m3 concrete cell in Test-2. The profiles of sodium leak and combustion in Monju were clarified through the tests but a floor liner made of carbon steel had holes due to corrosion in Test-2 while no holes and slight thinning were observed in the Monju leak event, on the contrary. Further examination revealed that much smaller volume of the concrete cell in Test-2 compared with the Monju secondary piping room affected increased atmospheric temperature and concrete wall temperature and a large amount of water was resultingly released from the concrete wall. The newly obtained information from the tests is reflected on the sodium combustion analysis code, ASSCOPS.

In the source term study, a new experimental rig has been constructed for the investigation of FP release behavior from high temperature spent fuel under an accidental condition. Various performance tests are in progress to validate experimental procedure and measurements. A mechanistic model is developed to explain FP bubble behavior in a coolant sodium and FP release behavior from a sodium pool and gives a good agreement with the experimental data.

9.4 Steam Generator Safety Research

Current steam generator safety research is mainly aimed at the improvement of sodium-water reaction evaluation method for the large scale FBR demonstration plant. For this purpose, an overheating failure mechanism has been studied in details. TRUST-1 is a tube rupture simulation test where a test piece of heat transfer tube of mod. 9Cr-1Mo steel is pressurized by nitrogen gas and heated by induction. The overheating failure behavior is properly analyzed by use of the FINAS code. As the second step, a new test rig named TRUST-2 is under construction, where a water/steam up to 19MPa of pressure flows inside the induction-heated test tube. After those simulation test, a large scale sodium-water reaction test project is under way. In the test, overheating failure mechanism will be examined under prototypical steam generator condition and the analytical method being developed now will be validated.

Another ongoing item on the steam generator safety is a leak detection system development for the double wall tube that is being developed for the steam generator installed in a primary heat transport system of a future FBR plant. Basic performance data was obtained for an outer tube leak detection system to detect helium leak into sodium. An inner tube leak detection system using infrared rays is also under developing to quickly detect humidity at gas plenums.

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 is improving the systems analysis code network for level-1 PSA. The improvement aims at (1) reduction of analytical cost, (2) realization of dynamic reliability simulation, and (3) PSA application in a "living" mode. For the first purpose, PNC is developing a new software PIRAS (PNC Integrated Reliability Analysis System) which is able to analyze a large size of event trees and/or fault trees rapidly. The second recent efforts have focused on development of a dynamic reliability analysis program (DYANA) dealing with the Emergency Operating

Procedures. The third efforts have been made to develop a system configuration management program that evaluates the risk during the shutdown and to improve a Living PSA System (LIPSAS) for power operation. The LIPSAS is used at the site of Monju plant to examine the applicability of the system to safety management of a real plant.

Efforts are being made to develop the component reliability database and statistical analysis system for an LMFBR (CORDS) on an engineering workstation. The CORDS is originally based on CREDO (Centralized Reliability Data Organization), a cooperative project between PNC and the USDOE, which ended in 1992. As a result of enhancement of the user-interface, it became possible to use the CORDS as an on-line handbook of the component failure rate.

As part of the data analysis, we are estimating the occurrence rate of coolant leakage from sodium piping system. Failure of coolant boundary is a rare event and hence a large uncertainty is included in occurrence rate estimation of the failure based on the statistical analysis of the empirical data. The occurrence rate of crack penetration is being analyzed with the probabilistic fracture mechanics technique. In addition, efforts were made to collect the weld defect data on the basis of the radiographic testing records of various weld joints in sodium piping system. The weld defect data is statistically being analyzed.

A shutdown level-1 PSA has almost completed. The reliability of the decay heat removal system (DHRS) during the plant shutdown (refueling and maintenance) was evaluated. There are several system trains out of service because of the scheduled maintenance so that it reduces redundancy of the safety functions. We considered several candidates of accident management (AM) measures as an operator recovery action under accident condition. As a result of this study, it concluded that these AM candidates were effective for the DHRS to maintain high reliability in cooperation with a long grace period.

In application of a PSA method in a large LMFBR, we started a reliability evaluation of a passive safety device such as a self-actuated shutdown system and developed a reliability computational tool using the Monte Carlo method.

Level-2 PSA tasks (consequence analysis) are underway. The current effort for the in-vessel physical process includes a preliminary analysis of an event tree of ULOF accident in a large LMFBR particularly reflecting recent experimental and analytical knowledge. For the ex-vessel physical process, an analysis of source terms has been continued postulating a reactor vessel melt through event. A debris-concrete interaction has been mainly investigated by means of a sensitivity analysis of the relation between the initial conditions such as debris mass and temperature and the quantity of radiological nuclides to be released.

CURRENT STATUS OF THE BN-350 FAST REACTOR

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Abstract

In this paper described general situation with electricity production in Kazakstan and current status of the fast breeder reactor BN-350 for the period May 1996 - May 1997. Since the project lifetime of the reactor finished in 1993 the special annual procedure on continuing operation was established. All responsible organizations agreed that the reactor finally will be shutdown in 2003. Taking into account this decision the administration of the reactor already started the preparation on the decommissioning program. Such international organizations as the IAEA and European Commission are actively participating in this process.

1. INTRODUCTION

Production of the electricity in the Republic of Kazakstan for 1996 had decreased on 11 % in comparison with 1995. The reason is in common recession of production in the Republic. The production of the electricity on nuclear station BN-350 remained at a previous level of 0.18% of the total production. The loading factor for 1996 was 20 % as well as in 1995, whereas earlier it was never been below 70 %.

The reactor was educed to the capacity of 420 MW only on February 4, 1997 that is stipulated by long outage of the reactor for fulfilment of work under the program of safety upgrade and, partially, absence of the sufficient financial resources for duly payment for control roads and of delay with receipt of the annual sanction on operation.

2. MAIN WORKS CONDUCTED IN 1996

The program of works on reactor BN-350 safety upgrading was developed in 1988. For past period the program was discussed many times, supplemented and changed according to the experience received. The works such as investigation of the site seismicity, research of emergency situation, additional strengthening of the equipment and building designs, creation of additional safety systems, the replacement of the equipment have been completed.
The following works and events should be noted that took place last year:

- The protection system of the reactor vessel and jacket against increasing of pressure - hydro-seal - was constructed and entered into operation.
- Using of the calculation code, that substantiate the allowable power levels of the reactor BN-350 in conditions of absence of the seismic resistant system for emergency cooling of an active zone, the IPPE experts had conducted precursory calculations of the reactor's parameters in case of active zone cooling by natural circulation of the heat-carrier.
- During October 1996, 4 experiments on research of heat exchange and process of natural circulation of the heat-carrier of 1 and 2 circuits are conducted. In experiments results are obtained confirming an opportunity of long-duration of decay heat removal after drainage of the steam-generators for a level of capacity 520 MW. The experiment ' 4 proceeded more than 30 hours and allowable levels of temperatures 420 °C for cold part of the secondary circuit pipeline were not exceeded.

- The principle decision on the discontinuance of the reactor operation in 2003 has been accepted. Proceeding from it the process of the preparatory work and development of the particular decommissioning programs is beginning now. First of all these are the programs on the radioactive wastes and spent fuel management.
- Within the framework of works beginning on the realisation of the new NPP with the light water reactor project construction the special financial resources for the valuation of an opportunity BN-350 replacement to the other nuclear facilities has been founded.

3. INTERNATIONAL CO-OPERATION ON BN-350 SAFETY UPGRADING

The meeting of the representatives of MAEK, KAEA and the IAEA on a question of harmonisation of the technical assistance of the BN-350 safety upgrade was held in Vienna during 23-24 October 1996.

Being based on the recommendations of the meeting 4 projects for realisation within the framework of technical co-operation programme with the IAEA were prepared by MAEK and transferred to IAEA.

Within the framework of technical co-operation programme with the European Commission 2 projects for safety and decommissioning were prepared also.

During of the last autumn the experts of MAEK, the NNC RK, the KAEA and Argonne National Laboratory (USA) had analysed a situation and some possible projects on management of the BN-350 spent fuel were developed.

Taking into account the present status of the spent fuel assemblies cladding that had been damaged during operation the preparatory work on its stabilisation had been begun since January 1997.

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STATUS OF LIQUID METAL REACTOR DEVELOPMENT ACTIVITIES IN THE REPUBLIC OF KOREA: AN INTRODUCTION TO KALIMER

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Abstract

This paper presents general information on the Korea Advanced LMFR, especially devoted such basis issues as major design features, development schedule and current status of program.

1. INTRODUCTION

The dependence of electricity generation on nuclear energy is quite heavy in Korea and the heavy dependence eventually raises the issues of efficient utilization of uranium resources, which Korea imports from abroad, and of spent fuel storage. From the viewpoint that liquid metal reactors (LMRs) have the potential of enhanced safety utilizing inherent safety characteristics, the transuranics (TRU) reduction and resolving spent fuel storage problems through proliferation-resistant actinide recycling, LMRs will be the most promising nuclear power option. The LMR development program was approved as a national long-term R&D program in 1992 by the Korea Atomic Energy Commission (KAEC) which decided to develop and construct a liquid metal reactor of 150 MWe by the year 2011 with the goal of developing an LMR which can serve as a long term power supplier with competitive economics and enhanced safety. Based upon the KAEC decision, the Korea Atomic Energy Research Institute (KAERI) is developing KALIMER (Korea Advanced LIquid Metal Reactor), which will be the first LMR in Korea.

The objective of KALIMER Program is to develop an economically competitive, inherently safe, environmentally friendly, and proliferation-resistant fast reactor concept. Efforts of the KALIMER Program will be concentrated on the establishment of an advanced design concept in order to contribute to the worldwide R&D and commercialization of the LMR.

KALIMER has enhanced safety features with the use of metallic fuel, Self-Actuated Shutdown System (SASS), Gas Expansion Module (GEM) in the core, and Passive Safety Decay Heat Removal System (PSDRS). Utilization of these enhanced safety features eliminates the need for diverse and redundant engineered safety systems and KALIMER accommodates unprotected anticipated transients without scram (ATWS) events without operator action, and without the support of active shutdown, shutdown heat removal, or any automatic system without damage to the plant and without jeopardizing public safety.

Environmentally friendly KALIMER has extremely low probability and amount of accidental radioactivity releases. The KALIMER core is loaded with metallic fuel which can be recycled through pyroprocessing. Recycling of transuranic elements by this process would avoid the expense and potential long-term risk of their disposal in a geological repository, and would provide increased proliferation resistance. The high thermal efficiency of KALIMER also improves the economics and makes it possible to discharge reduced amount of heat to the environment.

2. KALIMER DEVELOPMENT SCHEDULE

The KALIMER Development Program is divided into four phases for its completion by 2011; Concept Study, Basic Design, Detailed Design and Construction Phases. The overall KALIMER development schedule is shown in Figure 1 which also includes milestones and major R&D activities for each phase.

A pool-type sodium cooled KALIMER is currently in Concept Study Phase during which efforts have been concentrated on the development of basic design and sodium technologies unique to LMR design and operational characteristics. In parallel with the technology development, preliminary KALIMER design concept has been established through the evaluation of performance and applicability of innovative design concepts to the KALIMER design. The technology development efforts have supported the design concept evaluation works. Substantial progress has been made in all elements of the technology development and will be vital for the subsequent Basic Design.

After establishing the main design features of KALIMER by July, 1997, the basic design will be carried out for the next 5 years, that is, by 2002. The goal of this phase is to produce a Preliminary Safety Analysis Report using the design tools and methodologies which were developed in the Concept Study Phase and will be developed during this period through the international collaboration. Supporting R&Ds for the design code development and validation will also be performed during this phase.

Through the successful completion of subsequent detailed design, construction, and safety test, the KALIMER will reach initial criticality by the year 2011.

3. MAJOR DESIGN FEATURES OF KALIMER

The main feature of the KALIMER concept is to be formally setup in July of 1997. To develop the concept, a comparative study has been carried out by evaluating various design alternatives. The concept study is focused on making the KALIMER safer, more economic, having less impact on the environment, and more resistant to nuclear proliferation.

The heat balance of KALIMER concept is shown in Figure 2, and Table 1 summarizes the key design features of KALIMER.

CORE

As for the core design, preliminary design concept study is being performed to achieve development goals of KALIMER. The core is fueled with metallic fuel which has a good negative reactivity characteristics to enhance inherent safety. The core design aims to improve economics, and to develop a minor actinide recycle option by utilizing pyroprocessing technology for proliferation resistance. Several passive and innovative design features are incorporated in the core design to give inherent passive means of negative reactivity insertion, and the simplification and compactness inside the reactor vessel.

The KALIMER core system is designed to generate 392 MWth of power. The reference core utilizes a homogeneous core configuration with two driver fuel enrichment (< 20% U^{235}) zones that can allow a compact core and fuel shuffling. The core, shown in Figure 3, consists

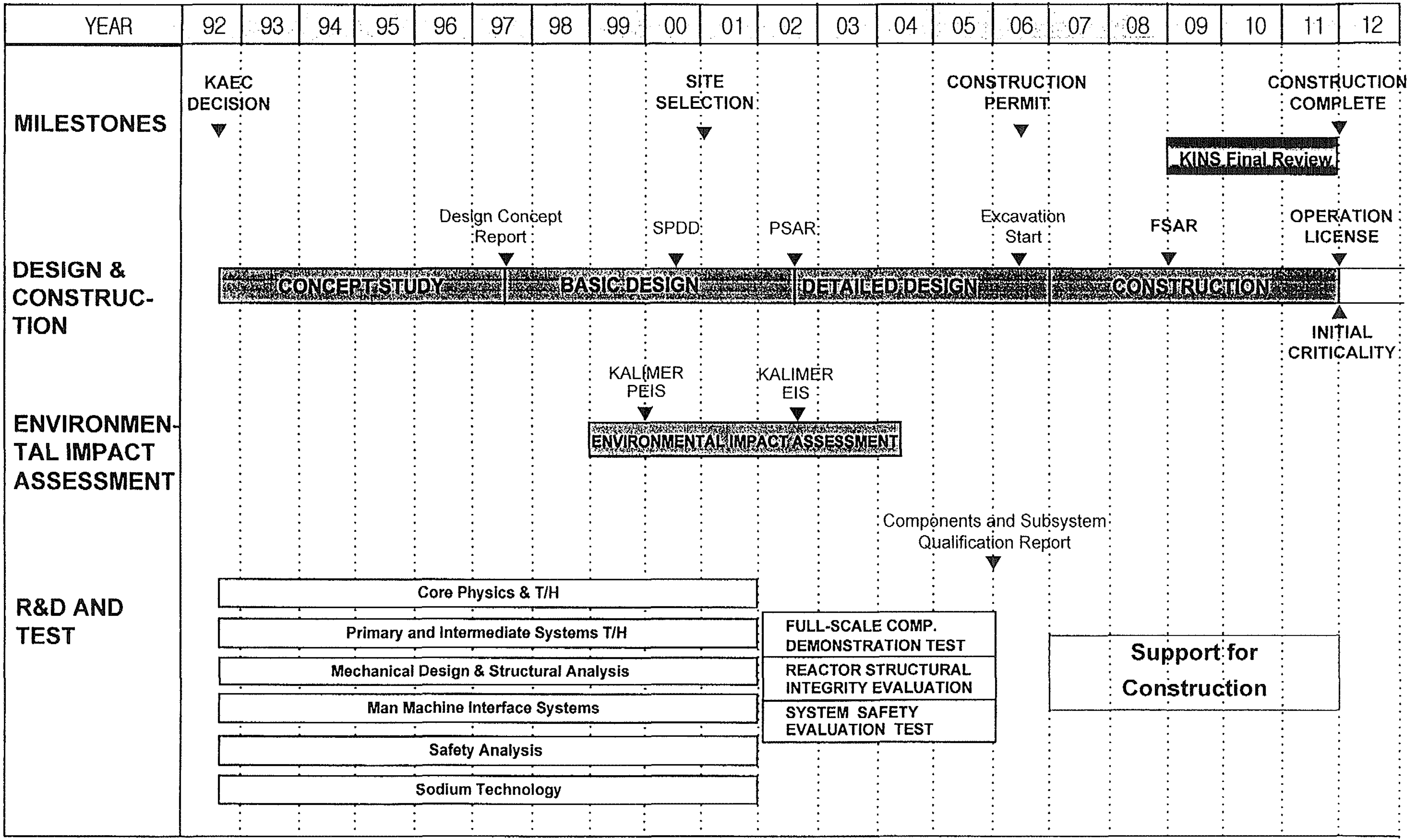
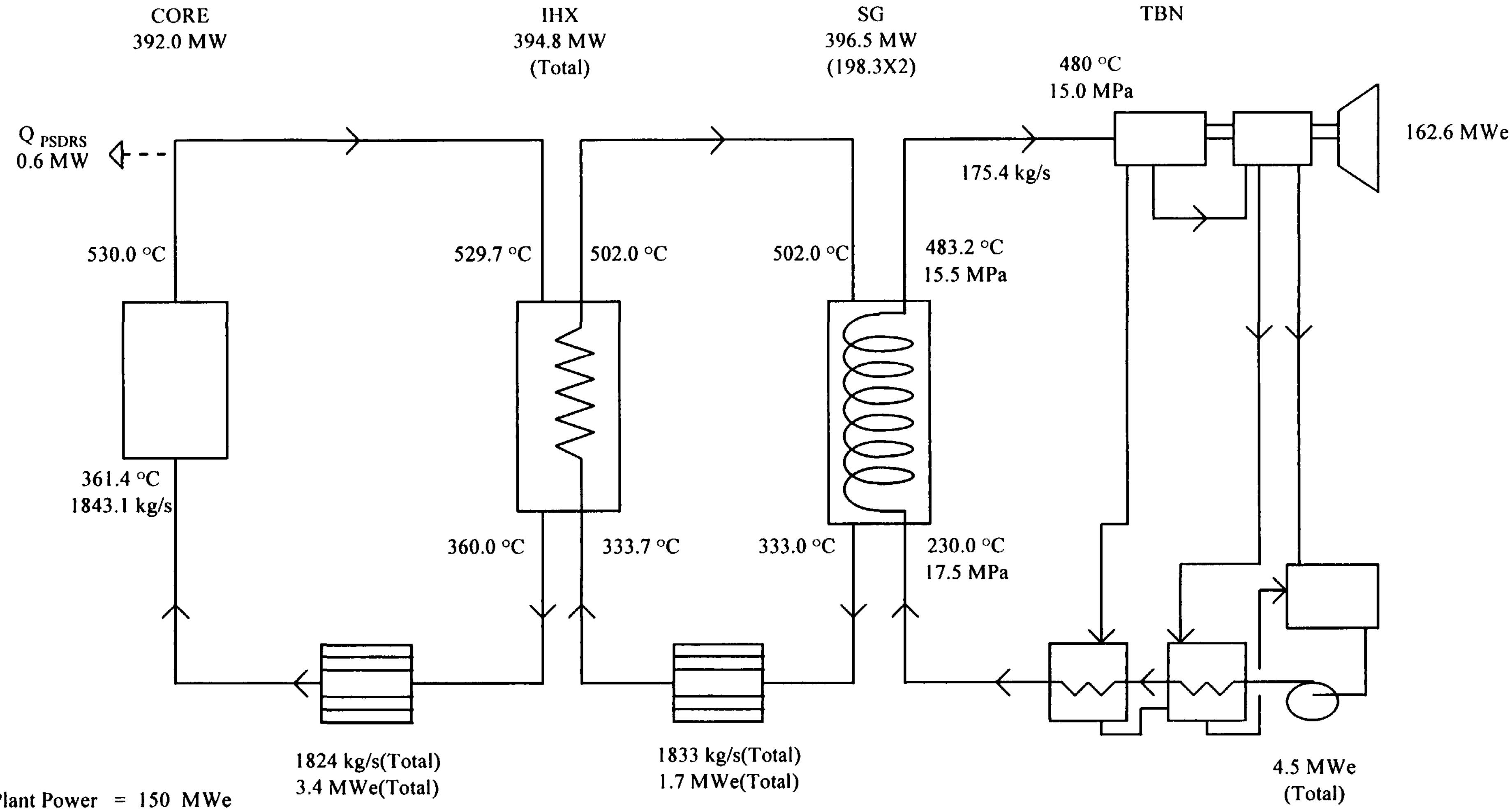


Figure 1. KALIMER Development Schedule



- Net Plant Power = 150 MWe
- Gross Efficiency = 41.5 %
- Net Efficiency = 38.3 %

Figure 2. KALIMER Heat Balance

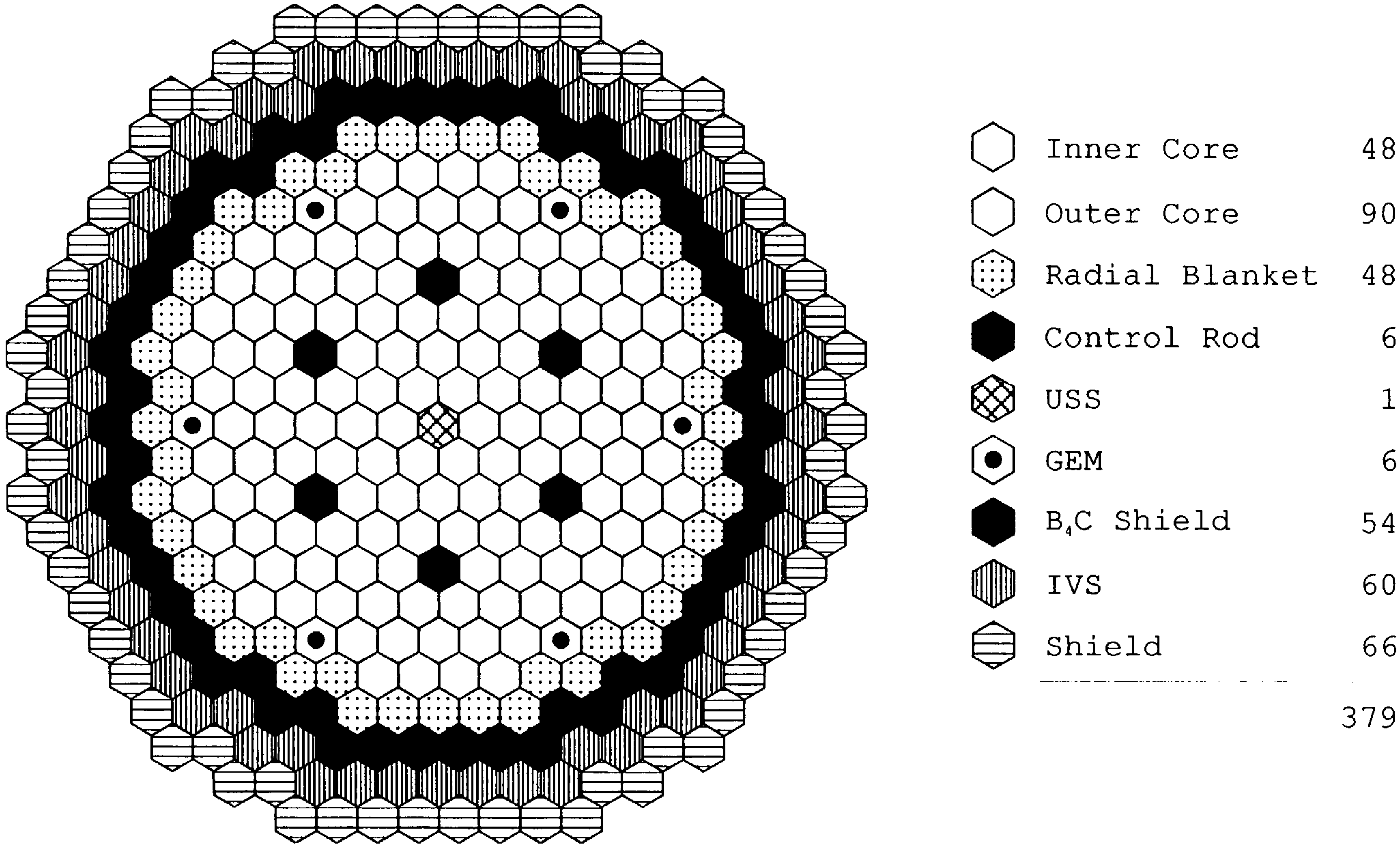


Figure 3. KALIMER Core Layout

Table 1. Key Design Features of KALIMER

OVERALL	
Net plant Power, MWe	150
Core Power, MWth	392
Gross Plant Efficiency, %	41.5
Net Plant Efficiency, %	38.3
Reactor	Pool Type
Number of IHTS Loops	2
Safety Shutdown Heat Removal	PSDRS
Ultimate Shutdown System	SASS
Fuel Handling	Variable Arm Pantograph Type IVTM
Seismic Design	Seismic Isolation Bearing
CORE	
Core Configuration	Homogeneous
Core Height, mm	1000
Axial Blanket Thickness, mm	0
Maximum Core Diameter, mm	3443
Fuel Form	U-10% Zr Alloy
Enrichments (IC/OC), % for the First Core	15.0 / 20.0
Assembly Pitch, mm	161.0
Fuel/Blanket Pins per Assembly	271 / 127
Cladding Material	HT9
Refueling Interval, months	12
PHTS	
Primary Sodium Core I/O Temp., ^{0}C	361.4 / 530.0
Core Total Flow Rate, kg/s	1824
Primary Sodium IHX I/O Temp., ^{0}C	529.7 / 360.0
Primary Pump Type	Electromagnetic
Number of Primary Pumps	4
PHTS Pressure Loss, M Pa	0.80
IHTS	
IHTS Total Flow Rate, kg/s	1833
IHTS Pump Type	Electromagnetic
IHTS Pressure Loss, M Pa	0.4
Number of IHXs	4
Number of SGs	2

of 138 driver fuel assemblies, 48 radial blanket assemblies, 6 control rods, 1 self-actuated shutdown system (SASS) assembly, 6 gas expansion modules, 120 shield assemblies, and 60 inner vessel storages in an annular configuration. The inner vessel storages are located in the stainless steel shielding zone. There are no upper or lower axial blankets surrounding the core. The reference core has an active core height of 100 cm and a radial equivalent diameter (including control rods) of 208 cm. The core outer diameter of all assemblies is 344 cm. The core structural material is HT9, low irradiation swelling characteristics of which permits adequate nuclear and breeding performance in a physically small core.

The core nuclear design will be largely governed by passive and inherent safety, and reactivity controls to have a safe shutdown capability. The ultimate mission for the KALIMER nuclear core design will be to design a minor actinide burning and fissile self-sufficient reactor in a closed fuel cycle through flexible adjustment of radial blanket annulus number. All the design efforts will be made to define the core design to be selected by the year 1999 according to the KALIMER long-term development schedule.

Preliminary analyses have been performed for the ATWS events in order to evaluate the inherent passive safety performance and to assess the safety margin of uranium metal cores. Results show that the temperature limits are met with margins for the core, which has inherent passive means of negative reactivity insertion, sufficient to place the reactor system in a safe stable state for these ATWS events without significant damage to the core or reactor system structure.

FUEL AND ASSEMBLIES

The base alloy, binary (U-10%Zr) metal fuel is a potential start-up fuel for the KALIMER as the driver fuel. Fuel pin is made of sealed HT-9 tubing containing metal fuel slug in columns. The fuel is immersed in sodium for thermal bonding with the cladding. A fission gas plenum is located above the fuel slug and sodium bond. The bottom of each fuel pin is a solid rod end plug for axial shielding.

The driver fuel, blanket fuel, reflector, and shield assemblies use identical structural components with only the bundle and its mounting grid changing from one assembly type to the other. The control assemblies use outer hardware (nosepiece, duct and handling socket) that is identical to that in the other assemblies. Reflector assemblies contain solid HT9 rods. The absorber assemblies use a sliding bundle and a dashpot assembly within the same outer assembly structure as the other assembly types. In all assemblies, the pins are in a triangular pitch array. The bottom end of each assembly is formed by the nosepiece which provides the lower restraint function and the coolant inlet. In fuel, reflector, and shield assemblies, the pin bundle attaches to the nosepiece with mounting rails. Surrounding the pin bundle and welded to the nosepiece is a hexagonal cross-section duct. The duct functions to control the coolant flow and isolate each pin bundle from its neighbors. It is also the structural tie between the top and bottom end hardware of the assembly.

According to the results of comprehensive analysis on in-reactor performance characteristics of metallic fuel, it is apparent that metallic fuel has not only a reliable performance characteristics up to 20 % burnup, but also a superior in-reactor performance compared with oxide fuel. Furthermore, metallic fuel has a capability of higher linear power rating, and provides better economic and diversion resistant characteristics than those of oxide fuel.

HEAT TRANSPORT SYSTEM

The heat transport system of KALIMER is designed with the emphasis on economy, safety and reliability. A super heat steam cycle is implemented to have a high plant efficiency noting that high thermal efficiency reduces the heat discharge from the plant, resulting in less impact to the environment. IHTS consists of two loops, and each loop is equipped with one steam generator unit to simplify the system design. For safety, a large system thermal inertia is achieved by using a pool based primary system. Strong emphasis has been given to the prevention and mitigation of possible sodium-water reaction events for the IHTS piping routing. The system reliability is improved by using electro-magnetic (EM) pumps which do not have moving parts for both of the primary and intermediate coolant pumping. The low momentum inertia of the EM pump is compensated for by using an auxiliary device which keeps a certain amount of rotating kinetic energy when EM pump runs normally but supplies electricity from the rotating kinetic energy to the EM pumps when the electricity supply to the pumps is interrupted. The operating temperature and component size were determined to achieve the gross plant thermal efficiency of 42%. To maximize the efficiency of development efforts, the BOP is designed under the strategy that basically conventional components would be used and the design condition is set for a turbine currently available in the market.

REACTOR STRUCTURE

KALIMER reactor vessel, which is made of type 316 stainless steel, has overall dimensions of 15 m height, 8.4m diameter, and 5 cm thickness in preliminary concept design and is composed with a cylindrical shell with integral hemiellipsoidal shell (bottom head). The improvement of safety of the vessel has been achieved with the consequence that the vessel has no attachments and no penetration other than the core support connections and shipping restraints. And the thermal liner inside the reactor vessel has been introduced to avoid the direct contact of hot pool sodium and the reactor vessel wall during the normal operation. The low operating temperature of the closure head is attained by including the horizontal layers of stainless steel plates under the closure head. For the structural simplicity, lower part of the support barrel enclosing core and the upper part of the support barrel serving as the boundary between the hot plenum and cold plenum are made as integral cylindrical structure.

The length and diameter of the containment vessel are slightly larger than those of reactor vessel. The containment vessel is made of 2.25Cr-1Mo steel. The gap between the containment vessel and the reactor vessel contains argon gas, and the instrument to detect sodium leakage from the reactor vessel will be installed in this gap region. The containment vessel transfers heat from reactor vessel to the air flow of PSDRS continuously.

RESIDUAL HEAT REMOVAL SYSTEM

In KALIMER, the shutdown heat removal system is designed with the emphasis on system reliability to achieve a higher level of plant safety. Safety grade heat removal is achieved by the PSDRS, which consists of the air path around the containment vessel and takes the decay heat from the reactor pool and discharges the heat to the atmosphere. Normally the decay heat is removed by a steam generator and condenser. During the maintenance of any IHTS, the heat is removed by a remaining IHTS loop. Also there is SGACS (Steam Generator Auxiliary Cooling System) to aid the decay heat removal. The SGACS induces natural or forced circulation of atmospheric air past the shell side of steam generator.

SEISMIC ISOLATION
The seismic base isolation for the whole reactor building which includes reactor structures, S/G and other systems and components important for safety will be introduced to maintain the sufficient structural integrity of KALIMER when subjected to the design basis earthquake such as horizontal Safe Shutdown Earthquake, 0.3g. Horizontal base isolation system has been currently being developed by performing the design and the fabrication of high damping rubber bearings, and the tests to understand the characteristics of the isolation system with international cooperations with IAEA IWGFR, GE and EERC of USA and CRIEPI of Japan, and the design procedures of seismic base isolation are also under development. In order to increase the seismic safety margins and to achieve economic design of the KALIMER, development of 3 dimensional seismic base isolation design concept is also considered.

As the establishment of the design guidelines for seismic base isolation system is in progress or developed in Japan, Italy, and USA, the seismic isolation design guideline for the KALIMER would be selected and modified from those developed by these countries.

IN-VESSEL TRANSFER MACHINE
A variable arm pantograph type in-vessel transfer machine (IVTM) will be used in KALIMER to move core assemblies within the reactor vessel. The IVTM is designed to have 6 degrees of freedom, that is, the main body rotation, main body vertical movement, extension/retraction of arm, vertical movement of gripper, gripping fuel assembly and rotation of the gripper. The IVTM with a single rotation plug covers all core locations, in-vessel fuel storage locations and the fuel transfer station. IVTM may be used in common in the neighboring reactors if required and designed to be plug-in/plug-out type. At present, a feasibility study on the plug-in/plug-out type IVTM with plug-in/plug-out type upper internal structure is being carried out.

ULTIMATE SHUTDOWN SYSTEM
KALIMER design adopts a self-actuated shutdown system (SASS) as an ultimate shutdown system located at the center of the core, which drops shut-off rod by gravity. SASS is a passive reactor shutdown system self-actuated by the natural physical phenomenon without any external control signals and any actuating power in the emergency of the reactor. Curie point electromagnet (CPEM) is to be used as a key component in SASS, whose saturated magnetic flux density is remarkably reduced at the curie point of the temperature sensitive material used in CPEM. When the temperature of the primary sodium goes up to the curie point, CPEM loses its electromagnetic force which was holding the shut-off rod. Then the shut-off rod with CPEM drops into the core due to its deadweight. The shut-off rods are designed to be an articulated type for the easy insertion into the core even when the guide tubes are deformed due to the earthquake.

4. CURRENT STATUS OF KALIMER PROGRAM

FEASIBILITY STUDY
KALIMER Program is now completing its first phase of development by producing a KALIMER Design Concept Report which summarizes the result of design concept study and describes the KALIMER design. Basis for the selection of design concept was established by the feasibility studies on power level, fuel types, configuration of primary and intermediate heat transfer systems, steam generator types, circulation pump types, ultimate shutdown

systems, in-vessel and ex-vessel transfer machines, seismic isolations, and residual heat removal systems. These feasibility studies were carried out by comparing various concepts which were already adopted by existing LMRs and were in a concept development stage. First principle and more sophisticated computer codes have been developed and basic experimental facilities including a small scale sodium loop were built at KAERI in order to support these studies.

DESIGN CONCEPT STUDY

KALIMER design concept has been being established by integrating the result of design concept feasibility studies. KALIMER Design Concept Report will provide a basis for the conceptual and basic design activities which will start from August 1997. In order to provide a diversity of perspective that aids to broaden the view of design problems and solutions, the KALIMER design concept will be reviewed by a number of qualified international experts with diverse backgrounds with the objective of identifying areas of high technical risk and possible measures to control and manage the risk. Design review team is now being organized and detailed procedures are being defined in order to perform the task.

PROGRAM PLAN DEVELOPMENT

Development of KALIMER requires the coordinated efforts of many areas of specialty, including engineering, licensing, testing, manufacture and construction. Since the integration of all facets of the process is critical to the success of the project, KALIMER Program Plan is being reviewed and revised for the basic design, detailed design and prototype plant project. For the Basic Design phase, detailed plant design tasks, design and analysis methods development tasks, R&D tasks supporting plant design and methods development will be defined. Key plant design activities and design methods development, validation and verification activities will be outlined for the Detailed Design phase.

INTERNATIONAL COLLABORATION

There are a number of equipments and technologies which require R&D work for implementation to KALIMER. To name a few, the submerged EM pump, a seismic isolator, high temperature structures, metal fuel, passive Curie point shutdown system, and a steam generator. R&D work for all of these items require massive investment to the facility and it is desirable to carry out the selected R&Ds through an international collaboration if possible. The international collaboration in the systems engineering and methodology development is also needed to support the domestic efforts of Korea for KALIMER development.

KALIMER design team members visited several international organizations in March 1997 in order to investigate the possibilities of collaboration for the next five years of Basic Design phase. International partners will be selected based upon the discussions with these organizations, exchanges of informations with other organizations not visited at that time, and the identification of tasks to be performed through the KALIMER Program Plan development.

5. CONCLUSIONS

The KALIMER Program initiated by the KAEC decision in 1992 is now completing its Concept Study phase during which the KALIMER Concept has been established and the KALIMER Program Plan has been finalized. Basic Design phase of the Program, which will start from August 1997, will require the LMR technology development through the international collaboration with various organizations of LMR development experiences.

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STATUS OF SODIUM COOLED FAST REACTOR DEVELOPMENT IN THE RUSSIAN FEDERATION

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Abstract

This report describes the recent development and activities concerning fast reactors in Russia. The status of nuclear power in 1996 and operational experience of the BOR-60 experimental rector and of the BN-600 nuclear power plant are presented. Main results of R&D programme in fast reactor area are discussed.

1. THE STATUS OF NUCLEAR POWER IN 1996

As of January 1997, 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 NPPs generated $108.8 \cdot 10^6$ MW-h ($99.3 \cdot 10^6$ MW-h in 1995).

The mean load factor was 58.32 % (53.4 % in 1995).

During 1996 there were only 2 incidents which can be classified by the INES as level 1.

The mean value of unplanned outages involving the triggering of emergency protection systems is 0.38 per reactor unit. As a whole, in 1996 safety indices of NPPs in operation were higher than in 1995.

2. FAST REACTOR OPERATIONAL EXPERIENCE.

2.1 THE BN-600 NUCLEAR POWER PLANT

Operating histogram of the BN-600 reactor for 1996 is shown in Fig.1. During this year there have been two shutdowns of the reactor for planned maintenance work and core refueling and three unplanned loops outages. During this period the load factor was 76.32 % (70.31 % in 1995).

In Table 1 main BN-600 reactor characteristics for the last five years and from the start of operation are presented.

Since first start-up the power unit have operated as electricity generating plant during 76.6 % of time, 21.1 % of time heaving been spent for refueling and planned repair works, while 2.3 % of time been taken by the unscheduled shut-downs (Fig.2).

The reactor refueling is carried out twice a year, and scheduled equipment repair works are fulfilled intentionally during refueling periods, taking in average 20 % of time (in April-June and October).

By January 1, 1997, the reactor equipment and systems have operated for about 110 000 hours, the reactor have been on power during 4500 eff. days, total time of the steam generators and steam-water circuit equipment being about 105 000 hours. Average load factors for the periods since full power reactor start-up and since entail start-up are respectively 73 % and 70 %.

Statistical distribution of the main performance indicators of the BN-600 reactor power unit operation and the range of the variation are represented in Fig.3 - 11.

In 1996 there were no water-into-sodium leaks in steam generators.

The operating experience has confirmed the correctness of the adopted steam generator concept - 12 water-into-sodium leaks have accounted for only 0.3 % of loss of power generation.

Inter-circuit leaks were mainly in the superheater modules (six events) and in the reheaters (five events), while in the evaporators there were only one leak event (Table 2).

The important result of the unit operation has been validation of the evaporator lifetime equal to 105. 000 hours (instead of design 50. 000 hours) and this allowed changing them only once instead of three times during the entire lifetime of the unit. This has been an outcome of the long-term program on studying evaporator condition and has been ensured by more strict water quality, the decreased rate of transient and emergency conditions against the design value, the periodical re-agent cleaning and washing by water to remove porous deposits.

By now the evaporators have worked out more than 100.000 hours each. Therefore they are replaced so that their lifetime wouldn't have been expired by the moment of the last old evaporator replacement.

The specified design lifetime of the superheater and re-heater modules corresponds to the reactor plant operation life - 200. 103 h.

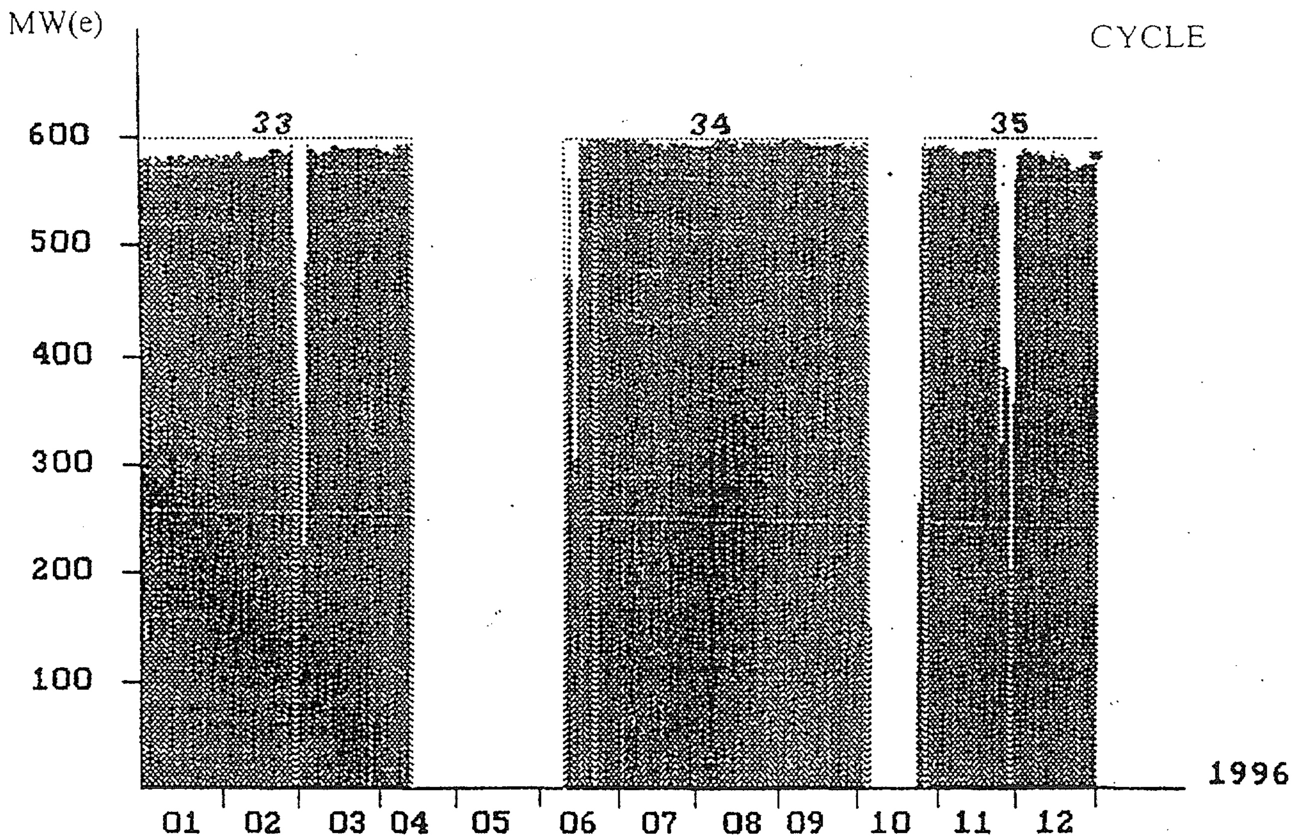

Fig. 1. BN-600 Operating Histogram.

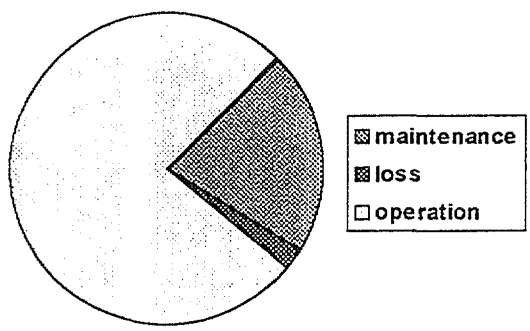

Fig.2. The BN-600 life time distribution.

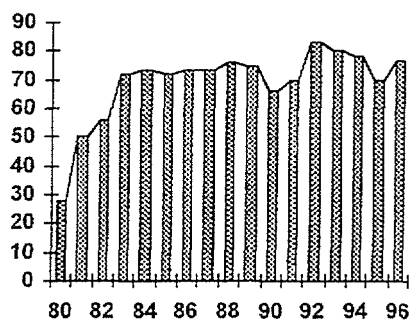

Fig.3. Load factor BN-600 (%)

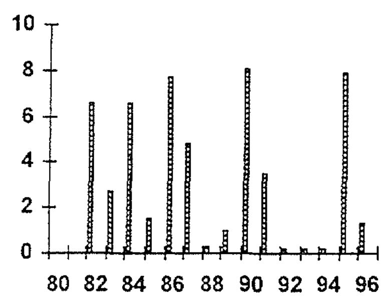

Fig.4. Unplanned load loss factor BN-600 (%)

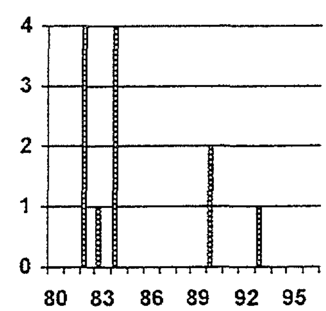

Fig.5. Unplanned scrams.

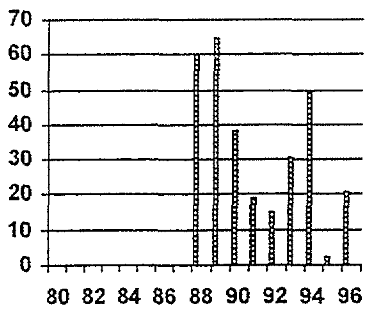

Fig.6. Low-active waste (m^3).

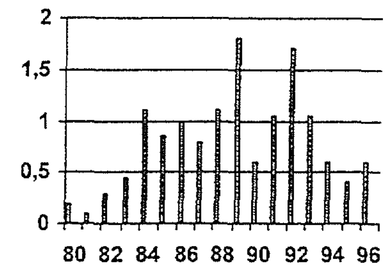

Fig.7. Collective radiation exposure BN-600 (man-Sv).

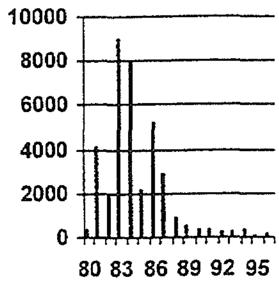

Fig.8. Noble gaseous discharge (Ci).

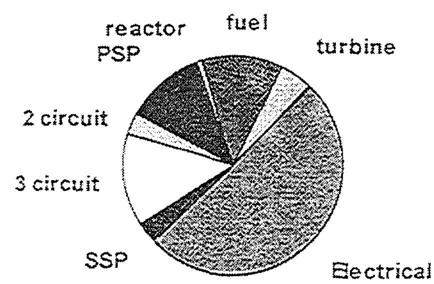

Fig.9. Equipment type events.

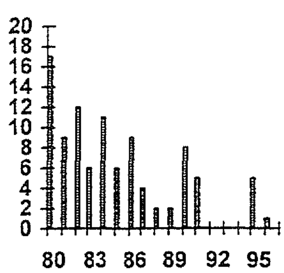

Fig.10. Number of power reduction events.

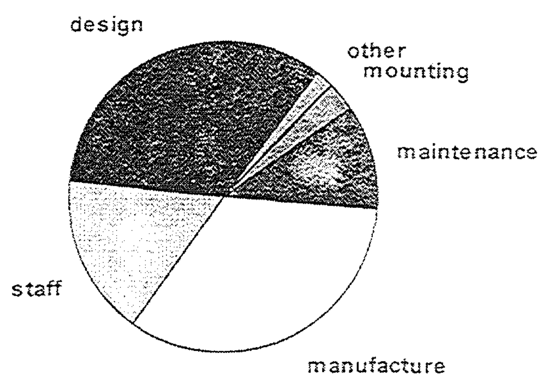

Fig.11. Causes of events.

Table 1. Main BN-600 reactor characteristics for the last 5 years and from the start of operation

Characteristic	Units	1992	1993	1994	1995	1996	From the start of operation up to Jan.1,1996
Electric power output	10^6 kWh	4402	3915	3810	3695	4022	60836
Load factor	%	84	80.3	78.2	70.31	76.32	70
The number of unit shut-downs		2	2	2	3	2	77
The number of loop outages		2	0	0	6	3	66

Table 2. BN-600 SG loss-of-tubes integrity events

Date	24.06 1980	04.07 1980	24.08 1980	08.09 1980	20.10 1980	09.06 1981	19.01 1982	22.08 1983	16.11 1984	10.11 1984	24.02 1985	24.01 1991
Leak place	RH	SH	RH	SH	SH	RH	SH	SH	E	RH	SH	RH
Leak size	L	L	S	S	S	S	L	S	S	S	S	S

Abbreviators: RH - re-heater, SH - main superheater, E - evaporator, L - large leak, S - small leak

Note: Large leak is characterized by changes in integral parameters of a secondary loop (sodium and gas pressure in expansion tank).

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).

In 1995 a decision was made to increase the time duration of the runs 32, 33 and 34 in order to confirm the performance of the core and the radial blanket up to burn-up of 11.3.% and 2 % h.a., respectively.

As a result the total lifetime of fuel subassemblies will be 540 eff. days.

In standard fuel subassemblies a maximum burn-up by the end of run 32 was 11.1.% h.a., a maximum dose was 80 dpa.

Up to now 12 SAs with pellet MOX fuel were irradiated in the BN-600 core. First eight SAs were irradiated during 3 cycles up to MOX burn-up of 10 ÷10.5 % h.a. with maximum dose of 70 ÷ 72 dpa. Last four SAs were irradiated during 4 cycles up to MOX burn-up of 11 ÷ 11.8 % h.a. with maximum dose 80.7 dpa. Three of them were examined in hot cell of BN-600. Initial PIE showed good performance of experimental SAs.

At present more 12 SAs with pellet MOX fuel are under irradiation in the BN-600 core.

2.2. THE BOR-60 EXPERIMENTAL NUCLEAR POWER PLANT

In 1996the BOR-60 plant was operated at 50 - 55 MW. The first loop of the reactor was provided with the reverse-type steam generator RSG1 and air exchanger, another one - with the reverse - type steam generator RSG - 2.

The results of plant operation in 1996 are presented in Table 3. In this table also are given the main results of BOR-60 operation from the commissioning up to 1997.

A large number of irradiation tests of fuel and structural materials of nuclear and thermonuclear reactors are carried out in the core and radial blanket and irradiation of electroceramics and electric insulation materials - in vertical channels.

Standard BOR-60 fuel is MOX vibropacked fuel with uranium gutter, maximum burn-up in some SAs - 21 h % h.a., maximum dose - 100 dpa. The total number of fuel pins with burn-up value more than 20 % h.a. is ~ 410 pins. Three fuel pins are under irradiation now in the dismountable test assembly; at the end of 1996 maximum burn-up was 30.5 at %. Dismountable test assemblies are used for pellet MOX fuel irradiation also, the maximum burn-up for pellet MOX fuel is ~ 17 % h.a.

Table 3 . Results of BOR-60 plant operation

Characteristic	1	2	3	4	1996	From commissioning up to 1997
Operation time, h	2184	1419	1550	1071	6224	146952
Availability factor	1.0	0.65	0. 7	0. 48	0. 71	
Maximum power level, MW (th)	50	50	50	55	-	
Power generation: thermal, Mwh	102002	53429	54165	54300	263896	6049024
electricity, Mwh	15324	9616.8	10891.2	9254.4	45086.4	991987
Steam generators operation time, h RSG1	2184	1399	1532	1062	6177	79609
RSG2	2184	1399	1532	1062	6177	32604
Heat output for consumers, GKal	33435	5006	2865	17036	58342	248807

One SA with U-Pu-10 % Zr fuel is under irradiation now, maximum burn-up - 7 % h.a.(end of 1996).

3. R&D PROGRAM

3.1. PHYSICAL RESEARCH AT CRITICAL FACILITIES

3.1.1. BFS CRITICAL FACILITIES IN 1996

In the first half of 1996, tests were completed on substantiation of the neutronics of the BN-800 reactor core with the sodium plenum at the top, allowing to minimize positive component of the sodium void reactivity effect in case of completely dried mixed uranium-plutonium fuel core.

The final stage was devoted to the measurements of the control rod worth and sodium void reactivity effect in the core sector including all three different enrichment zones (BFS-58-4 critical assembly was simulating the reactor at the beginning of the run after refueling).

In the previous years, more comprehensive tests were carried out modeling two other BN-800 reactor core conditions:

- end of the run, when the reactivity compensating rods are withdrawn from the core completely (BFS-58-1);

- middle of the run, when the reactivity compensators are half withdrawn from the core (BFS-58-3).

Preliminary analysis made in 1996, showed, that the whole experimental program carried out, gave the possibility to determine the sodium void reactivity effect with sufficient accuracy ($-0.2 \pm 0.3\%$ $\Delta k/k$).

In the second half of the year, the preparation work was under way for the experimental studies on the simplified model of the BN-800 reactor core with increased plutonium content (up to 39%) in the mixed U-Pu fuel and non-fertile blankets. These studies are going to be conducted in 1997.

BFS-1 critical facility was used to continue studies on the characteristics of fast reactor cores designed for the weapons grade plutonium utilization and minor actinides burning, for instance, the effect of neptunium introduction into fuel. The first stage of these studies made on the insert of the BN-800-Superphenix reactor fuel with up to 14% of depleted uranium dioxide replaced by neptunium dioxide was accomplished in 1995 (BFS-67 critical assemblies set).

In the first half of 1996, the experimental studies were completed on the insert with MOX fuel containing about 39% of weapons grade plutonium (BFS-69-1). In order to estimate accuracy of the central functional calculation, the following parameters were measured:

- fission and radiation capture spectral indices;
- reactivity coefficients of reactor material samples;
- sodium void reactivity effect;
- worth of boron rod mock-ups;
- distribution of fission rates; etc.

On the further stage, about 13% of depleted uranium dioxide in the central area of the insert were replaced by neptunium dioxide (BFS-69-2), and the same set of measurements was conducted.

The introduction of this considerable neptunium amount into the fuel makes the neutron spectrum more rigid, and the increase of spectral indices following high energy band of the neutron spectrum, is 5% to 11%, that is approximately two times less than that for the MOX fuel with traditional plutonium content.

Owing to neptunium introduction, the SVRE value caused by the removal of 15 kg of sodium from the central area of the insert becomes less negative by about 0.3 β_{ef}, while the removal of 0.15 kg of sodium causes its change from – 0.16%β_{ef} to + 0.33% β_{ef}, the efficiency of boron mock-ups being decreased by 6-8%. The changes in the central reactivity coefficients are also observed. Fission rate pattern deformation is less significant.

Currently, the experimental results are under analysis.

In the second half of 1996, BFS-71-1 critical assembly was mounted with MOX fuel containing about 55% of plutonium. Standard measurements were carried out, and neptunium was added to the fuel in the late 1996 in order to continue measurements. The results obtained are under analysis.

3.1.2. THE COBRA CRITICAL FACILITY

In 1996, all scheduled studies on the uranium-thorium fuel cycle were completed on the CBR-22 critical assembly with the core containing metal enriched uranium and metallic thorium. The high enrichment of the core fuel (~20%) caused quite rigid neutron spectrum (neutron fraction having energy lower than 10 KeV did not exceed 1%).

Based on the results of the initial stage of the experimental data analysis, the following conclusions can be made:

1. The calculation made using KRAB, MMK and FFCP codes with the BNAB-90 database overestimate K_{eff} value by about 1% as compared to the measurement results.

2. The calculated and measured values of the average capture cross section by U-238 are in a good agreement, while the calculated value of capture cross section for thorium is about 12% higher than that obtained in the experiment.

3. About 8% underestimation of calculated value of Np-237 average capture cross section is observed.

In order to explain the above results, as well as for analysis of considerable amount of experimental data on the transuranium isotope fission cross sections, it is necessary to perform more specified analysis using various nuclear data base versions.

Measurements on time distribution of prompt fission neutrons using Rossi-alpha method (applicable for β_{eff} measurements) were made on this assembly. Prompt neutron decay constants were measured for various arrangements of polyethylene moderator discs in the assembly reflector, including the case when there were no moderator discs in the reflector. It became possible to explain the presence of short decay periods owing to the corrected coefficient of connection between the core and the polyethylene discs in the framework of the two point model.

A series of tests were also carried out on the CBR-22 critical assembly on the substantiation of changing over from the fertile blanket to the steel blanket, the radioactive isotopes to be yielded in the latter. The replacement of uranium blanket by the steel one has resulted in significant reactivity gain (up to 0.9% $\Delta k_{eff}/k_{eff}$).

New designs of irradiation devices for the production of radioactive isotopes, in particular cobalt-60, were tested in the steel blanket. The key innovation in these designs was the absence of the absorber elements. In the course of the experiments, power bursts up to ~50% were revealed on the core boundary. Since the considerable decrease of power is observed in the core periphery, these bursts are unlikely to cause the parameters of power reactor standard fuel subassemblies to go over permissible operating limits. However, the final conclusion on this issue can be drawn only after the comprehensive analysis of S/A performance taking into account the obtained experimental data.

3.2. MINOR ACTINIDES UTILIZATION

The concept of fast reactor core for effective minor actinides (Np and Am) burning was analyzed with safety assurance of such cores taken into account. In results of optimized studies it is suggested three core variants allowed to utilize minor actinides effectively.

The preliminary investigation showed that introduction of minor actinides in fuel results in great increasing of sodium void reactivity effect (SVRE) and therefore the using of traditional fast reactors is impossible because it isn't possible to assure zero value of SVRE which is dictated by Russia Safety Rules. Hence it is necessary to find the way which allows to fulfill this requirement and to burn the minor actinides effectively.

Proceeding from this we suggested some ways to solve these problems:

- the location of minor actinides into radial blanket sub-assemblies;
- the location of minor actinides homogeneously in fuel in which the fuel enrichment was preliminary increased;
- the location of minor actinides in fuel based on inert matrix where uranium-238 is replaced (partially or fully) by diluent (for example, cerium oxide or magnesium oxide).

In first case the introduction of minor actinides doesn't influence on core physical parameters and efficiency of its burning is ~ 30 kg/TW*hour. When we use the core with increased plutonium enrichment the efficiency can be increased up to 32 kg/TW*hour.

In case of homogeneous introduction of minor actinides into the fuel the allowed amount of minor actinides is limited by 10 %. Table 4 shows efficiencies of minor actinides burning for different variants of their isotope content.

The efficiency of minor actinides burning is 14-23 kg/TW*hour. This value is two time less than in case of traditional fast reactor core.

The largest efficiency of minor actinides burning can be reached by using core with fuel without uranium-238. SVRE value as a function of ma fraction for different fuel compositions is shown in Table 5.

The low value of SVRE allows to introduce from 15 % (fuel PuO_2+MAO_2) to 35 % (fuel $U^{235}O_2$- MAO_2) of minor actinides.

Major physical characteristics of the variants are given in Table 6.

Thus, optimal core variants permit the utilization of:

125 kg/year of Np and 100 kg/year of Am for plutonium fuel loading;

260 kg/year of Np and 245 kg of Am/year for uranium fuel loading.

Thus the investigations carried out allowed to show clearly that the problem of minor actinides utilization can be effectively solved by using cores which were developed for plutonium burning.

However the problem of curium isotopes utilization stand over because the fuel fabrication (even we add a small amount of curium) process is very awkward due to high activity of these isotopes (mainly Cm^{244} and Cm^{245}).

Table 4. Efficiency of minor actinides burning (kg/yr.)

Minor actinide fraction, %	2.5	5	10	25
Minor actinides isotope content				
Np-237 - 100 %	93.21$^{*)}$	186.02	357.62	810.01
	73.36$^{**)}$	166.15	338.59	794.1
Am-241 - 100 %	97.04	197.54	386.17	885.01
	64 25	151.34	300.11	727.53
Isotope mixture	-	-	-	-
	57.57	134.96	275.14	671.41

$^{*)}$ the figure corresponds to burning of the isotope considered;

$^{**)}$ the figure corresponds to total minor actinides burning (accounting for their production).

Table 5. SVRE dependence on MA fraction, % Δk/k

Fuel	$PuO_2+Np)_2(AmO_2)$			$U^{235}O_2+NpO_2(AmO_2)$		
MA fraction	5	15	25	10	25	50
SVRE	-2.10	+0.06	+1.52	-6.54	-2.11	+2.31
	(-2.18)	(-0.31)	(+0.84)	(-6.23)	(-2.06)	(+2.11)

Table 6. Major physical characteristics of optimal variants

	PuO_2+15%+MAO_2		$U^{235}O_2$+35%MAO_2	
	Np	Am	Np	Am
Critical plutonium loading in a fuel pin, g	33.3/40.1	33.2/39.5	35.4/42.4	36.0/43.3
MA loading in a fuel pin, g	5.9/7.0	5.9/7.0	19.1/22.8	36.0/43.3
MA burned quantity, kg/year	125.8	100.3	263.4	245.2
Doppler reactivity coefficient, $T\partial k/\partial T$	-0.00289	-0.00277	-0.00135	-0.00130
Rate of reactivity change, % Δk/k	3.6	3.4	1.6	1.5

3.3. WEAPONS-GRADE PLUTONIUM UTILIZATION

To date Russia has certain achievements in the field of design, construction and operation of fast reactor installations BR-10, BOR-60, BN-600 operate successfully in the country, besides the BN-350 reactor in Kazakhstan operates under technical assistance of Russia. An example of fast reactor successful is NPP with BN-600 reactor, which is the best in operation results per 1995-1996 in the "Rosenergoatom" system.

Despite the fact that the BN-350 and BN-600 reactors use presently uranium fuel, Russia has gained significant experience in plutonium utilization in fast reactors (Table 7).

For example, as early as in 1957 for the pulse reactor IBR-30, a core from metal fuel-plutonium alloy was manufactured. In 1957-1965 a fuel was produced in the form of plutonium dioxide for the BR-5 and IBR-2 reactors.

Thus far in the BR-10 reactor two cores have been tested with fuel containing a weapons grade plutonium.

In the BOR-60 reactor a number of fuel pins has been tested, manufactured by different technologies with the use of plutonium having different isotope composition. Reactor operates, re-circulating its own plutonium.

Extended tests of MOX-fuel were carried out under conditions of the BN-350 and BN-600 reactors with the use of installation "PAKET" (I.E. "MAJAK") designed for SA production (up to 10 SAs per year). In the BN-350 reactor, in-reactor tests and subsequent investigations were performed of test SAs containing up to 350 kg of weapons grade plutonium. To date more than 3000 fuel pins on the basis of such fuel have been manufactured and tested in the BN-350 and BN-600 reactors.

Test works with NOX-fuel for fast reactors (FR) performed permitted the transition to designing and construction (using MOX-fuel) of a small series of power fast reactors BN-800 (South-Ural and Belojarsk), which corresponds to the power strategy of Russia and the problem of fuel cycle closing in nuclear power.

Far and away the development of nuclear power in Russia with fast neutron reactors is connected with the use in a closed fuel cycle primarily of power plutonium, produced in thermal reactors. However, availability of successfully operated BN-600 reactor and the construction in sight of BN-800 reactors permits to consider the

Table 7. Utilization of mixed uranium-plutonium fuel in fast reactors

Reactor	BR-10 (PuO_2)	BOR-60	BN-350	BN-600
Pellet fuel				
Fuel pins number	3287	~370	1778	1524
Maximum burn-up, % h.a.	14.1	12.6	10.8	10.5
Maximum linear power, kW/m	17	50	48	48
Vibrocompacted fuel				
Fuel pins number	-	~12800	254	762
Maximum burn-up, % h.a.	-	26	7.2	9.8
Maximum linear power, kW/m	-	54	48	48

Table 8. Changing of plutonium isotope composition*

	Pu-239	Pu-240	The rest of the isotopes
Initial weapon plutonium[1]	93.5 %	6.0 %	0.5 %
Plutonium from BN-800 reactor (100% MOX)	84.4 %	14.2 %	1.4 %
Plutonium from WWER-1000 reactor (1/3 MOX)	52.1 %	25.2 %	22.7 %

* The data are average over the total mass of unloaded spent fuel.

variant of ex-weapon plutonium utilization as a well actual and acceptable in near and distant prospects.

3.3.1.SOME PECULIARITIES OF WEAPONS GRADE PLUTONIUM UTILIZATION IN FAST REACTORS

Reactor physics

Physical aspects of a fast reactor core with the use of MOX-fuel are well studied and justified by the practice of NPP in operation. It has been shown that a weak dependence exists of major reactivity coefficients, influencing safety (except SYRE) on a fuel type - uranium or plutonium, or on plutonium isotope composition.

Oscillations of critical parameters as a function of plutonium content in fresh fuel are compensated by correcting initial enrichment in values in accordance with special method worked out for the BN-800 reactor design. However, it is typical for the civil plutonium. No such problems arise if weapons-grade plutonium is used.

The SVRE problem is solved by means of sodium plenum introduction over the core, providing zero or slightly negative SVRE value. This realized in the BN-800 reactor design.

When using FR-burner with a fuel, containing increased plutonium content, both Doppler - effect and SVRE value decrease. However, the Doppler - effect value decrease is in allowable limits, and besides reactor self-protection in some beyond design basis accidents improves.

Thus, we can state with assurance that the utilization of weapons grade plutonium in fast reactor, even at transition to a FR-burner, doesn't lead to serious problems, connected with the reactor physics and the provision of its safety.

Core design modification

Weapons grade plutonium utilization in the BN-600 and BN-800 reactors will of course result to the necessity of solving of some technical problems. First of all, the new core design will be required for the BN-600 reactor. However, this development will be based upon already checked technological approach adopted for the BN-800 reactor design, and this will simplify the development.

Hybrid core design for the BN-600 reactor is based on approaches tested in the BN-600 reactor. The new element, common for all cores of BN-600 and BN-800 reactors intended for weapons grade plutonium utilization is the non-fertile (steel) blanket. This is necessary requirement from the standpoint of weapons grade plutonium utilization in fast reactors.

Weapons grade plutonium denaturation

With the use of weapons grade plutonium in fast reactor MOX-fuel, the unloaded spent fuel will contain plutonium, which is of greater interest for the subsequent power use, as compared with plutonium in spent fuel of thermal reactors. Both in fast and thermal reactors plutonium is reliably removed from weapon standards.

As an example, Table 8 presents data characterizing a denaturation process (isotope composition changing) of weapons grade plutonium in BN-800 (reactor-burner) and WWER-1000 reactors.

In the case, when a more deep plutonium denaturation is needed, the following methods can be in principle used for these purposes in a fast reactor:

- previous mixing of power and weapons grade plutonium prior to its loading into the reactor;
- the increase in fuel burn-up level;
- the introduction into fresh fuel of minor actinides (Np, Am) for enhanced production of Pu^{238}, yield of spontaneous fission neutrons of which is 2.6 times as higher, and specific heat release is 80 times higher, as compared with Pu^{240}.

It is necessary to note, that whereas the first two variants lead to some decrease of weapons grade plutonium consumption rate, the third variant, in addition to effective weapons grade plutonium denaturation, permits "burning" of minor actinides (MA).

<u>Reasoning from the above, we can state that weapons grade plutonium utilization in a fast reactor permits "to regulate" over a wide range its denaturation extent, and thus to solve optimally tasks, connected both with subsequent power utilization and with prevention of plutonium use in military purposes. In either case, denaturated plutonium is reliably removed from the weapons grade plutonium grade.</u>

3.3.2. MAJOR CHARACTERISTICS OF REACTOR TECHNOLOGIES FOR WEAPONS GRADE PLUTONIUM UTILIZATION IN RUSSIAN FAST REACTORS

Solution of a strategic task of plutonium utilization under closed fuel cycle conditions is connected with BN-800 type reactor construction and corresponding fuel infrastructure.

The modern BN-800 reactor design has a breeding ratio equal to unit, uses MOX-fuel on the basis of power plutonium and meets all safety requirements.

The construction of South-Ural NPP first unit with BN-800 reactor at present time is discontinued for economical reasons, but in the Russian power program the construction of four similar blocks is provided (sites of South-Ural and Belojarsk NPP).

No problems exist with substitution of power plutonium with weapons grade plutonium in the BN-800 core conditions. In this case, the traditional core (design version) may transform 1640 kg of weapons grade plutonium per year to a spent fuel standard, burning and producing 140 kg Pu/year.

More efficient plutonium burning in BN-800 reactor can be carried out by transition from a reactor-converter to a reactor-burner. By now, two cores-burners types have been developed, as applied to structure dimensions of the reactor design variant.

First of them considers the replacement of breeding blankets to steel ones. In this variant, with annual consumption of approximately 1700 kg weapons grade plutonium, physically 140 kg/year is destructed.

More intensive burning (with approximately the same annual consumption can be achieved by simultaneous replacement of breeding blankets and some increase in the fuel enrichment by plutonium. In this case up to 270 kg Pu/yr can be burned.

Major characteristics of the above-mentioned cores of the BN-800 reactor, developed to date at a level of technical proposals, are presented in Table 9.

Under conditions of delay in BN-800 units construction, as a first stage of realization of the weapons grade plutonium utilization program in Russia, the transition to MOX-fuel of the operating BN-600 reactor can be considered.

In this case, at the initial stage it is appropriate to use a variant without changing the standard SA design with introduction of MOX-fuel only into limited core volume (hybrid core) and replacement of breeding radial blanket to a steel one.

Replacement of uranium SAs to subassemblies with MOX-fuel is performed in the high enrichment zone (HEZ). Allowable number of SAs with plutonium is determined reasoning from the requirement of retaining SVRE value in zero or negative regions.

On the basis of safety conditions, a limit number of SAs with MOX-fuel has been determined; in this case an annual consumption of weapons grade plutonium is equal to 240 kg.

Table 9. Major core parameters of BN-800 reactor at weapons grade plutonium utilization

	Traditional core (design variant)	Core with steel blankets	Core with increased fuel enrichment and steel blankets
SA number LEZ/MEZ/HEZ	585 211/156/198	585 211/156/218	585 211/156/218
Fuel enrichment in zones, %	.5/18.7/20.816	16.4/18.6/20.8	19.2/23.5/27.5
Annual needs in SAs, ($\varphi = 0.8$)	393	407	407
Annual needs in Pu, kg ($\varphi = 0.8$)	1640	1700	1650
Maximum burn-up, % h.a.	10	10	11.5
Annual quantity of burned plutonium, kg ($\varphi = 0.8$)	0*	140	270
Isotope composition of unloaded plutonium	LEZ MEZ HEZ Core	LEZ MEZ HEZ Core	LEZ MEZ HEZ Core
Pu-239	83.0 84.9 87.1 85.1	82.5 84.0 86.2 84.4	81.1 83.0 85.2 83.1
Pu-240	15.5 13.8 11.8 13.6	15.9 14.5 12.6 14.2	17.4 15.6 13.6 15.5
Pu-241	1.4 1.2 1.0 1.2	1.5 1.3 1.1 1.3	1.4 1.3 1.1 1.3
Pu-242	0.1 0.1 0.1 0.1	0.1 0.1 0.1 0.1	0.1 0.1 0.1 0.1
Minor actinide fraction in unloaded plutonium, %	0.09 0.08 0.06 0.08	0.09 0.08 0.06 0.08	0.08 0.08 0.06 0.07
Isotope composition of minor actinides			
Np-237	70.7 71.5 66.4 69.4	70.7 71.5 66.4 69.4	66.5 65.3 62.9 64.8
Am-241	22.7 22.9 29.3 25.1	22.7 22.9 29.3 25.1	25.8 27.8 30.9 28.2
Am-242m	0.5 0.5 0.3 0.5	0.5 0.5 0.3 0.5	0.6 0.6 0.5 0.5
Am-243	4.1 3.4 2.6 3.4	4.1 3.4 2.6 3.4	4.8 4.2 3.8 4.4
Cm-242	1.8 1.6 1.4 1.5	1.8 1.6 1.4 1.5	2.1 1.9 1.7 1.9
Cm-244	0.2 0.1 - 0.1	0.2 0.1 - 0..1	0.2 0.2 0.2 0.2

*) ~ 140 kg/year of weapon plutonium are bushed in the core, and approximately the same quantity is accumulated

Table 10. SVRE dependence

Number of SAs with Pu	SVRE, % ΔK/K
0	-0.254
32	-0.144
98	0.021
159	0.124

Table 11. Major cores parameters of the BN-600 reactor at weapons grade plutonium utilization

	Hybrid core with steel radial blanket (80 SAs with MOX-fuel)	Core with full loading by MOX-fuel without breeding zones
SA number	80[1]	404
LEZ	-	136
MEZ	-	94
HEZ	80	174
Fuel enrichment by zones		
LEZ	-	16.1
MEZ	-	20
HEZ	20.5	24.7
Annual needs in SA (φ=0.8), items	40	273
Annual needs in Pu (φ = 0.8), kg	243	1310
maximum burn-up, % h.à.	11.3	10
Annual quantity of burned plutonium, kg	22[2]	115
Isotope composition of unloaded plutonium[3]		
Pu-238	0.02	0.03
Pu-239	87.39	86.02
Pu-240	11.57	12.75
Pu-241	0.94	1.1
Pu-242	0.08	0.1

[1] - with MOX-fuel only

[2] - for active part of SA with Pu, but in the whole reactor 216 kg of Pu are accumulated

[3] - from active SA parts with MOX fuei.

Major hybrid core characteristics of the BN-600 reactor are given in Table 11.

Next step seems to be realistic in the program of weapons grade plutonium utilization in the operating fast reactor BN-600 - transition to 100 % MOX-fuel.

Major core characteristics of the BN-600 reactor with 100 % loading with MOX-fuel are presented in Table 11.

Annual consumption of weapons grade plutonium in this variant is 1310 kg, and in this case 115 kg Pu/yr are burned.

3.4. RADIOLOGY

Besides highly radioactive wastes of the core and blankets (spent core and blanket fuel subassemblies), medium and low level radioactive wastes are produced in fast reactors, including primary sodium coolant, shielding materials and the primary system components.

In spite of their low radioactivity level, all the primary circuit wastes are to be disposed, their volume significantly exceeding that of the spent core fuel subassemblies.

It is theoretically possible, that the problem of the radioactive wastes of the primary circuit, except for the spent fuel and fission products (S/A), can be solved without their disposal, if the wastes are stored in the intermediate storage during 50 or 100 years. In order to do this, several interconnected problems should be solved.

For example, in order to decrease the amount of solid radioactive wastes, accumulated as a result of irradiation of fast reactor core and shielding materials, it is necessary to solve the problems of introduction of the low activated materials and development of technology of the highly activated impurities removal, as well as the problem of decreasing superficial pollution of structures with the radioactive deposits from the coolant.

The superficial pollution rate can be decreased by means of limiting the release of the fuel and fission products from the failed fuel elements into the primary sodium coolant and the yield of the radioactive corrosion products, and by removing highly activated impurities from the coolant.

The fulfillment of these requirements is even more important, since it would allow avoiding disposal of the primary coolant.

On this stage, studies have been aimed at the evaluation of permissible content of highly activated impurities in the design structural materials of the core, blanket and shielding, as well as in the advanced low activated materials.

Analysis of the results of activity studies on the low activated materials and estimates of permissible limits on the impurities content in these materials have shown theoretical possibility of creation of structural materials for the core, blanket and shielding not requiring disposal because of their low induced activity after 50 or 100 years of storage (on condition, that the impurities content in the materials has been decreased down to permissible limit, and the problem of radioactive deposits on the component surfaces has been solved).

Studies have shown that the activity of 19 elements (Ta, W, As, Si, Ge, H, F, Na, Pr, Ga, Au, Ti, Mn, Cr, V, Rh, Yb, Hg, Mg), irradiated in the blanket, after their storage during 50 or 100 years, is such that there would be no need in their disposal.

Studies on the impurities activity have shown, that there are about 20 impurity elements (Co, Ba, U, Cd, Th, Sm, Nd, Li, Dy, N, Cs, Nb, K, Hf, Ni, Mo, Er, Gd, Ho, Cl), whose content in the low activated materials should be monitored.

For the most promising low activated materials, such as Ti and TiH_2, irradiated in the blanket of LMFR, the content of the natural uranium and niobium impurities should not exceed several thousandths of ppm, while for thorium and dysprosium this limit is several hundredths of ppm.

As regards the requirements to the low activated steel (which, in addition to Fe, should only contain the elements with the activity lower than that of Fe) and boron carbide, these are less rigid (10 ppm and over), however, the blanket and some part of shielding made of these materials would have to be buried.

In the BN-600, BN-800 and BN-1600 reactors, all structural steel of the blanket and shielding located between the core and the intermediate heat exchangers, should be buried because of presence of the activation products of niobium, molybdenum and nickel in the stainless steel.

3.5. THERMOHYDRAULICS

3.5.1. LIQUID METAL BOILING IN THE CIRCUIT UNDER NATURAL FLOW CONDITIONS

In the SSC RF IPPE, boiling of the liquid metal coolant (sodium-potassium eutectic alloy) under the natural flow conditions has been studied. The objective of these studies is to investigate the possibility of the stable decay heat removal from the fast reactor core.

The experimental studies are carried out on the fuel subassembly model containing seven mock-up fuel elements installed in the natural flow liquid metal circuit (Fig. 12). The length of the heater zone is 420 mm. There is a 165 mm length section of hydrodynamic stabilization upstream of the heater zone, the latter being followed by the mobile bundle of 200 mm length (see Table 12). All necessary instrumentation is

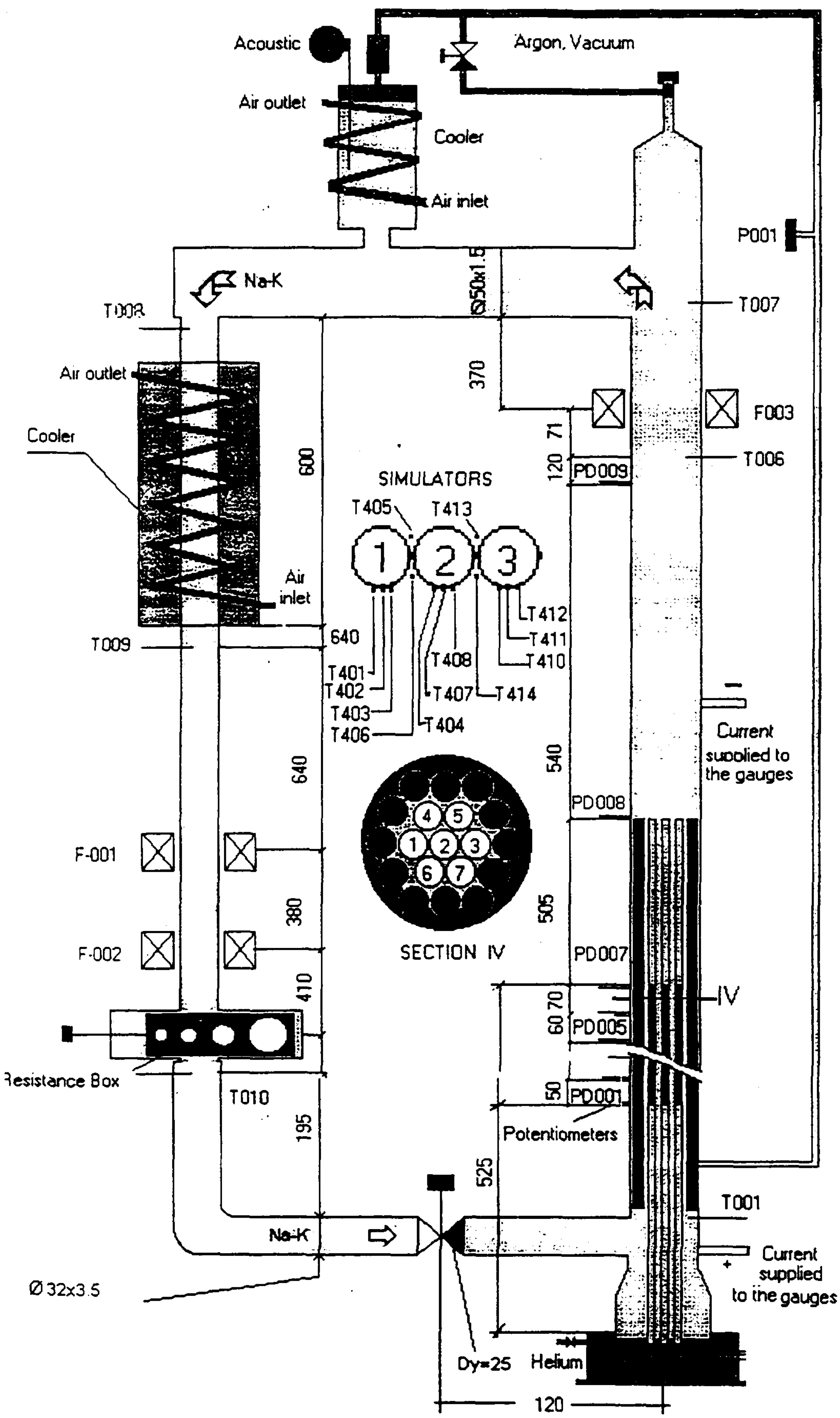

Fig. 12. Schematic diagram of 7-pin test section in AR-1 facility

Table 12. The main geometrical parameter of 7-pin subassemly.

N	Name	Value
1	Outer diameter of pin simulator, d, mm	8
2	Inner diameter of pin simulator, d_1, mm	6
3	Length of heating region, l_0, mm	420
4	Pitch-to-diameter ratio	1.185
5	Number of simulators, n	7
6	Number of displacers	12
7	Pitch of wire-wrapping, h, mm	100
8	Diameter of wire-wrapping, d_n, mm	1.5
9	Outer diameter of subassembly wrapper tube, D, mm	50
10	Thickness of wrapper tube, δ, mm	1.5
11	Length of wrapper tube, L, mm	2878
12	Length of mobile 7-pin bundle, l, mm	200
13	Maximal displacement of mobile 7-pin bundle, h_n, mm	300

provided for both model and the circuit. Temperature distribution over the fuel element surface, temperature of the coolant, its flow rate, pressure and vapor content pattern, as well as acoustic signals are measured. The circuit design makes it possible to vary the coolant flow rate in the model by controlling the pressure drop value of the circuit.

Tubes with different surface roughness were tested: smooth surface (roughness of 0.15 μm), and natural (roughness in 0.63-1.25 μm range).

The method of studies was based on the increasing of power supplied to the working section on condition that the circuit hydraulic resistance characteristic is preliminarily set. The coolant temperature in the working section increased until the saturation point was reached and the boiling process started.

Three boiling modes were observed in the course of studies: stable boiling on the initial stage, and pulsation boiling mode, which further was transformed into the stable mode again, when the power supply was increased.

In order to study the effect of heat transfer surface condition on the fuel element cladding temperature, one fuel element mock-up with polished cladding surface and the others with the rough surface were used.

Data presented in Figs. 13 and 14, demonstrate significant effect of the surface condition on the fuel cladding temperature. The abrupt temperature rise of the cladding under the appeared vapor bubble was observed on the smooth cladding surface (corresponding to the standard fuel elements), while there was no increase of the rough surface cladding temperature which was kept at the saturation point.

Liquid metal boiling tests carried out during 5 hours with over 20 motion cycles of the mobile bundle, have demonstrated, that the coolant evaporation and boiling processes do not depend on presence (or absence) of the liquid metal plenum over the working section.

Values of the heat transfer coefficient obtained experimentally depend on the heat flux. The increase of the heat flux value causes the increase of the heat transfer coefficient. Heat transfer coefficient values are similar for the stable and unstable boiling modes. The comparison of the above data with those for the liquid metal boiling in tubes has shown good agreement. The following relationship can be used:

$$\alpha = 3\, q^{0.7}\, p^{0.15} \quad ,$$

where $[q] = W/m^2$; $[p] = bar$.

The only relationship obtained for the critical heat flux value under boiling conditions in the smooth tube bundle is in a good agreement with the data on the crisis of liquid metal boiling in the tubes and annular gaps, and the integrating relationship proposed by Kottowski:

$$q_{av} = 0.216\, r\, (1 - 2x_1)(\rho\omega)^{0.807}(d/l)^{0.8} \,.$$

The results obtained are not sufficient for giving recommendations on the stable sodium boiling modes in the core of fast reactor. In order to specify the area of stable boiling depending on various parameters and factors and to generalize and justify data for their application in fast reactors, fine experimental studies are to be performed.

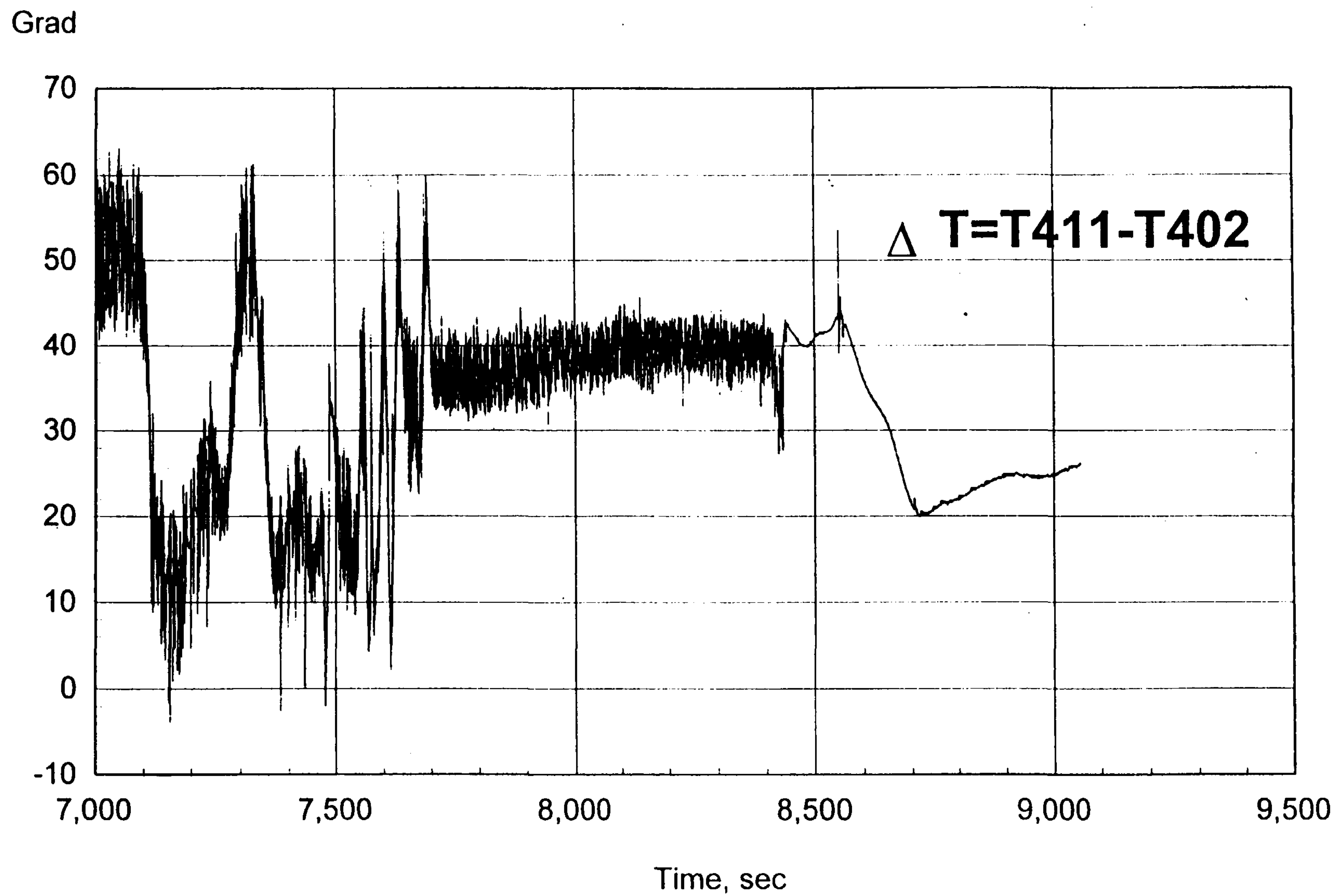

Fig. 13. Difference in readings of thermocouples embedded in smooth and rough pins in top section of region heated

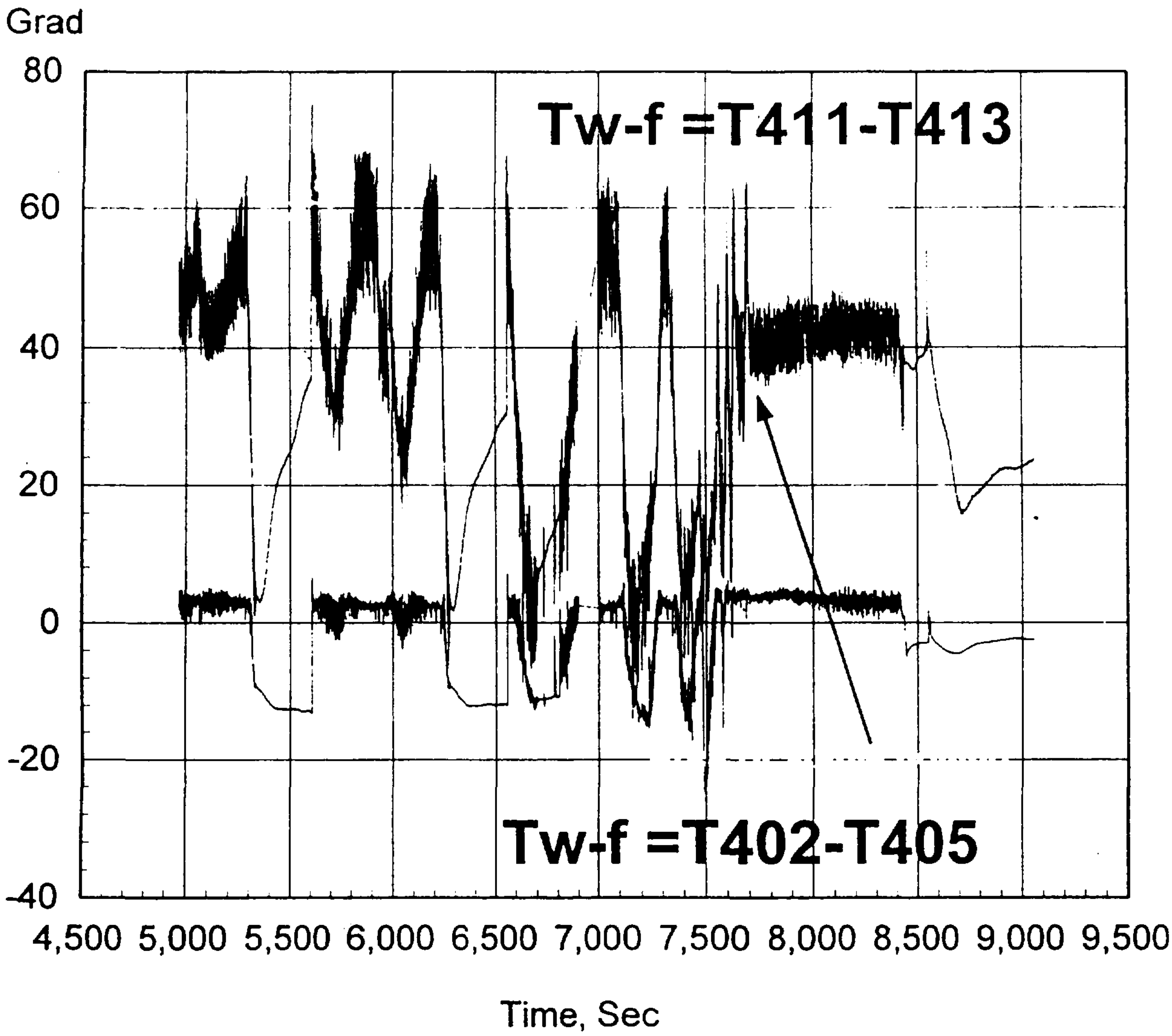

Fig. 14. Difference between smooth pin temperature and coolant temperature (T402,T405). and rough pin temperature and coolant temperature in top section of heated region (T402-T405)

3.5.2. SOME ISSUES OF APPROXIMATE MODELING OF FAST REACTOR DECAY HEAT REMOVAL SYSTEMS

Considering the progress in studies on fast reactor decay heat removal systems, the following points may be noted:

- variety of foreign studies made in water (Ramona, Neptune and other models), allowed getting an idea of the process physics and verifying computer codes, used for the further reactor calculations;

- recently, studies have been carried out in Japan on applicability of experimental results obtained in water tests, to the sodium cooled reactor conditions.

However, the theory of water based modeling of the reactor cooling by the decay heat removal system lately developed in Japan, leads to certain problems related to the effect of some physical phenomena on the process of transition to the natural flow. The comprehensive analysis made at the SSC RF IPPE on the results gained in many countries on the modeling theory has shown as follows:

- existing opinion on the minor role of the Reynolds number in natural flow development is not correct. Its direct effect upon the thermal hydraulics within the plenum is insignificant, but its influence on the hydraulic resistance coefficients determining the entire process of natural flow (Euler criteria), is appreciable;

- in order to specify some points of theory and practice of modeling, additional experimental studies are required using fragmentary models in water and liquid metal. Such experiments also give grounds for the improvement of calculation codes.

In particular, it is necessary to check self-similarity of the processes with regard to Peclet number, to study the effect of initial conditions, system pressure drop values, etc. For this purpose, simple flat water model has been constructed at the SSC RF IPPE. This model will be used for studying the effect of various factors and similarity criteria on the accuracy of modeling of the natural sodium flow (LMFR DHRS) on the water experimental rig.

Currently, adjustment works are under way. On the first stage of studies, experimental justification of the self-similarity of transients for different Peclet numbers.

3.6. SODIUM LEAKS AND FIRES

3.6.1. SODIUM SPRAY FIRES

Results of analysis and operating experience gained have shown that the large sodium spray leaks are not real as far as the reactor facilities are concerned. Nevertheless, studies on the sodium spray burning are still going on. Leak tight vertical cylindrical chambers with elliptical bottoms and covers are used for the tests. The main objective of the experimental studies is to determine the portion of sodium burnt in drops. The sodium jet is directed straight upwards. The injector is located near the chamber bottom. There is a reflector provided on the jet route in order to break the jet into drops. The distance between the injector and reflector can be varied.

Originally, tests were carried out in $2m^3$ volume, ~1m diameter chamber. The injecting device operated from the burst of the membrane that occurred at 0.45 MPa to 0.65 MPa (the value of pressure being dependent on the membrane design). The temperature of injected sodium was in the range of 400^0C-500^0C. The amount of sodium injected in 0.07s to 0.7s varied from 400g to 2000g.

Maximum gas pressure in the chamber varied from 0.16 MPa (injection of 400g of sodium) to 0.41 MPa (injection of 2000g of sodium). Maximum local gas temperature (in the area of the jet striking against the reflector) was 1250^0C.

The experimental data obtained were not sufficient for answering the question concerning the portion of the sodium burnt in drops, since the large amount of sodium fell on the chamber walls after the jet had been broken. That was why in 1996 the experiments were continued in the $8m^3$ volume, ~2m diameter chamber. The sodium was supplied by opening the valve instead of bursting the membrane. Maximum amount of the sodium delivered at 500^0C was 12 kg. Maximum pressure was 0.16 MPa and maximum temperature in the upper part of the chamber was ~ 700^0C. The results obtained so far show that the maximum portion of the injected sodium burnt in drops is 10%. However, the tests are going on.

3.6.2. SODIUM LEAKS UNDER THERMAL INSULATION

The effect of thermal insulation on the mechanism of sodium leaking through the crack in the pipe wall is of great interest. A decision was made to make the experimental studies on the phenomenon. For this purpose, a series of tests were carried at the IPPE in 1996. The experimental rig included working section (horizontal 145x5 mm diameter glass fiber insulated pipe with 3 mm diameter orifice on the top), sodium tank, and gas vessel. In the first three tests, there was one 50 mm layer of thermal insulation over the pipe. In the fourth test, the diameter of the orifice was 8 mm, and there were three 40 mm insulation layers over the pipe: two fiber glass layers and one layer of kaolin wool. In contrast to the design applied in the reactor facilities, there was no jacket over the insulation. Sodium was forced to the working section by increasing the gas pressure. Besides the regular observation in the course of tests, measurements of temperatures of the sodium in several points of the working section and that of the air in the cell, as well as of the sodium leak rate and time were conducted.

The input data and maximum temperature values measured in the area under the insulation in the course of experiments completed by now presented in Table 13.

Table 13. Input data and results of experimental studies on the effect of thermal insulation on the process of sodium burning

Test No.	T_{Na}, ^0C	P_{Na}, MPa	τ_{Na}, min	G_{Na}, kg/s	V_{Na}, l	T_{max} under insulation, °C	Notes
1	520	4	7.5	0.08	40	800	*)
2	515	5.5	8.0	0.09	50	850	*)
3	400	5.5	10	0.1	70	600	*)
4	450	5.5	2	1.0	120	600	**)

*) 3 mm diameter orifice. Thermal insulation: one 50 mm layer of glass wool;

**) 8 mm diameter orifice. Thermal insulation: two 40 mm glass wool layers plus one 40 mm kaolin wool layer.

In these tests, neither burst of thermal insulation, nor flowing out of sodium jet, nor its spraying occurred. The sodium was leaking through the insulation, their interaction being very active. It was most significant in the area of the working section, located below the defect.

3.7. WORKS AND INVESTIGATIONS ON SODIUM TECHNOLOGY

- The effects of oxygen behavior in sodium loops with carbon simultaneously present in sodium at high concentrations (as compared to its saturation level at a given temperature) while providing the contact of sodium flow with a developed surface of mild steel (carbon content 0.2 % wt). The experiments have been performed with oxygen supplied into the rig in the form of sodium carbonate, sodium oxide and water. The initial levels of oxygen in sodium (determined by means or chemical analysis and electrochemical cells measurements) were equal to several tens ppm with the volume of sodium in the rig of ~ 100 kg during 3 - 4 hours the oxygen level has decreased down to the threshold values of oxygen determination by routine methods. Solid phase products found in the rig contained high levels of carbon and iron. An explanation of the phenomenon, being now in progress, presupposes the formation of the sodium-iron compound oxides by means of the intermediate interactions with carbon participation.

- A set of the experimental runs performed in study of the depositions formation mechanism and their subsequent behavior in contact with sodium. Depositions were obtained on the gas plenum surfaces of the experimental parts, by means of pumping of argon, polluted by water and oil vapor through the gas plenum during various time intervals and at various pollution's concentrations. Reactions rates and volatile products composition have been measured with account for various sodium temperatures, times of the depositions aging etc. For the depositions, formed due to wetted argon it was shown, that theirs hydrogen content is essentially dependent on sodium mirror temperature and temperature distribution on the walls of the gas plenum.

- Experiments have been carried out to study sodium saturation temperature by the products of its reaction with water. It was shown that sodium hydride was the first to crystallize from the saturated solution. For the "water" concentrations of 10 -150 ppm there were obtained terminal solubility expressions for sodium hydride and sodium oxide in sodium.

- A physical model has been developed to describe the processes in the system "structural material surface-evaporated drop of sodium" (or some other liquid metal coolants). During experimental verification of the model it was shown that interaction of sodium with structural material took place at the total contact surface.

4. STATE OF ART OF THE BN-800 REACTOR CONSTRUCTION

Currently, the BN-800 reactor design undergoes examination according to the regulations now in force in Russia, in order to obtain the license to continue its construction on both Beloyarsk and South-Urals sites.

In order to fulfill the construction of the 4th power unit of the Beloyarsk NPP, the public joint-stock company has been established incorporating several large enterprises and institutions whose financial potential is sufficient for accomplishing the construction during 7 years. The economical analysis of the BN-800 reactor design, including the revision of some expense items is carried out aimed at the cost reduction.

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FAST REACTOR RESEARCH IN SWTZERLAND

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Abstract

The small Swiss research program on fast reactors serves to further understanding of the role of LMFR for energy production and to convert radioactive waste to more environmentally benign forms. These activities are on the one hand the contribution to the comparison of advanced nuclear systems and bring on the other to our physical and engineers understanding.

1. Comparison of Advanced Nuclear Systems

In the 90s, industry and research have invested efforts in order to further develop nuclear technology in the direction of a "new safety quality" while preserving or even regaining competitiveness against today's most economic energy carriers (e.g. natural gas). Different plant concepts rely on different approaches (evolutionary - innovative), focus on diverging realisation periods and use passive systems or inherent safety features to a different extent. There are also considerable differences between fuel cycles either in the basic approach or in the details.

This study aims at a comparison of future reactor concepts, paying particular attention to aspects of safety, of the fuel cycle, the economics, the experience-base and the state of development. Representative examples of typical development lines, that could possibly be "of interest" within a time horizon of 50 years were selected for comparison. This can be divided into three phases:

- Phase I includes the next 10 years and will be characterised mainly by evolutionary developments of light water reactors (LWR) of large unit size; representative: EPR.
- Phase II, i.e. the time between 2005 and 2020 approximately, encompasses the forecasted doubling of today's world-wide installed nuclear capacity; along with evolutionary reactors, innovative Systems like AP600, PIUS, MHTGR, EFR will emerge.
- Phase III covers the time between 2020 and 2050 and is characterised by the issue of sufficient fissile material resources; novel fast reactor systems including hybrid systems can, thus, become available; representatives: IFR, EA, ITER‡ (the latter being).

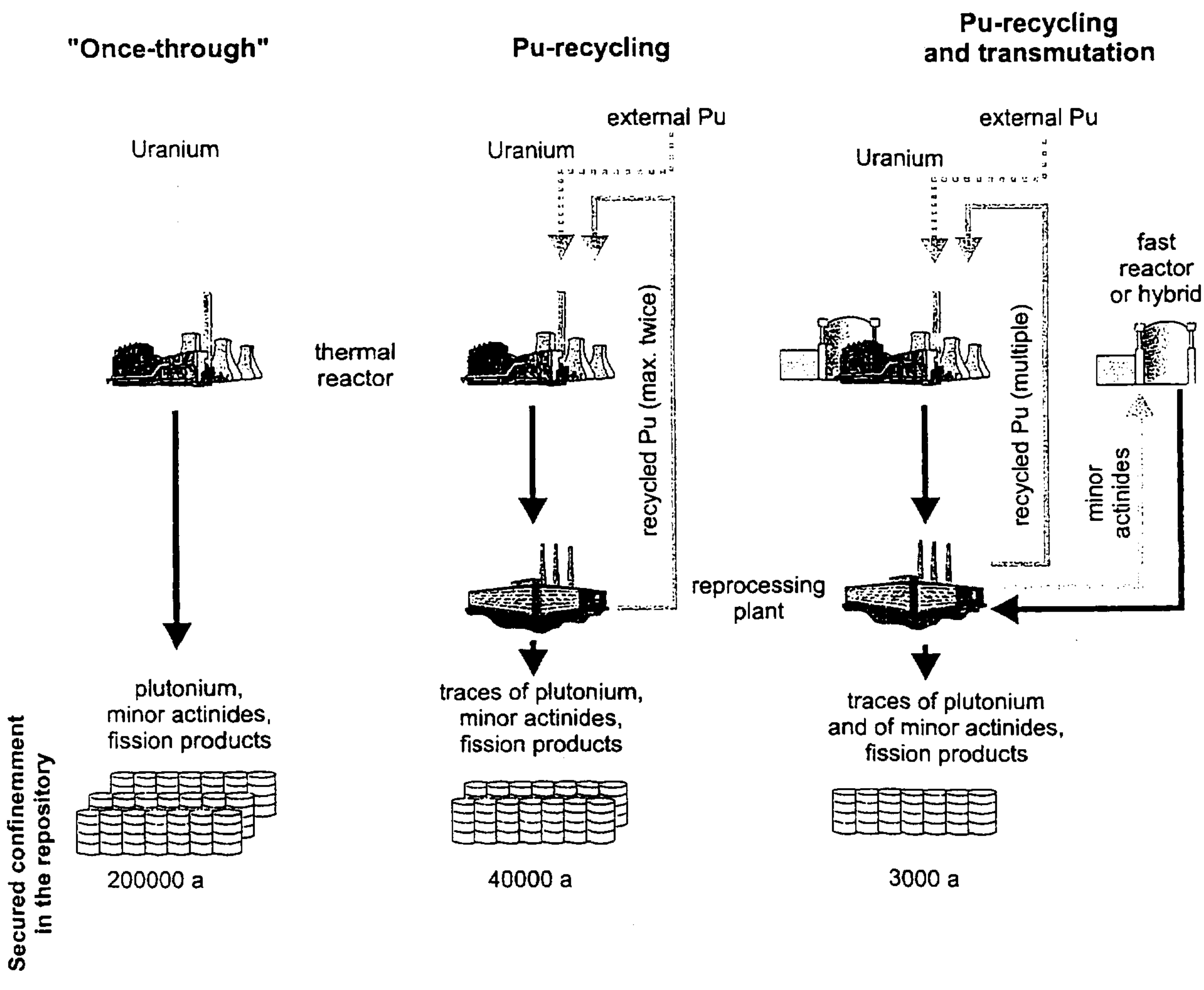
The evaluated concepts foresee partly different fuel cycles. Fission reactors can be operated in principle on the basis of either a Uranium-Plutonium-cycle or a Thorium-Uranium-cycle, while combinations of these cycles among them or with other reactor concepts than proposed are possible. With today's nuclear park (comprising mainly LWRs), the world-wide plutonium excess increases annually by about 100 t. Besides strategies based on reprocessing like

- recycling in thermal and fast reactors with mixed oxide fuels ($(U, Pu)O_2$),
- plutonium "burning" in reactors with novel fuels without uranium or in "hybrid" systems

allowing a reduction of this excess, direct disposal spent fuel elements including their plutonium content ("once-through") is being considered.

‡ Used as a comparison object that does not represent a true alternative in the time period considered.

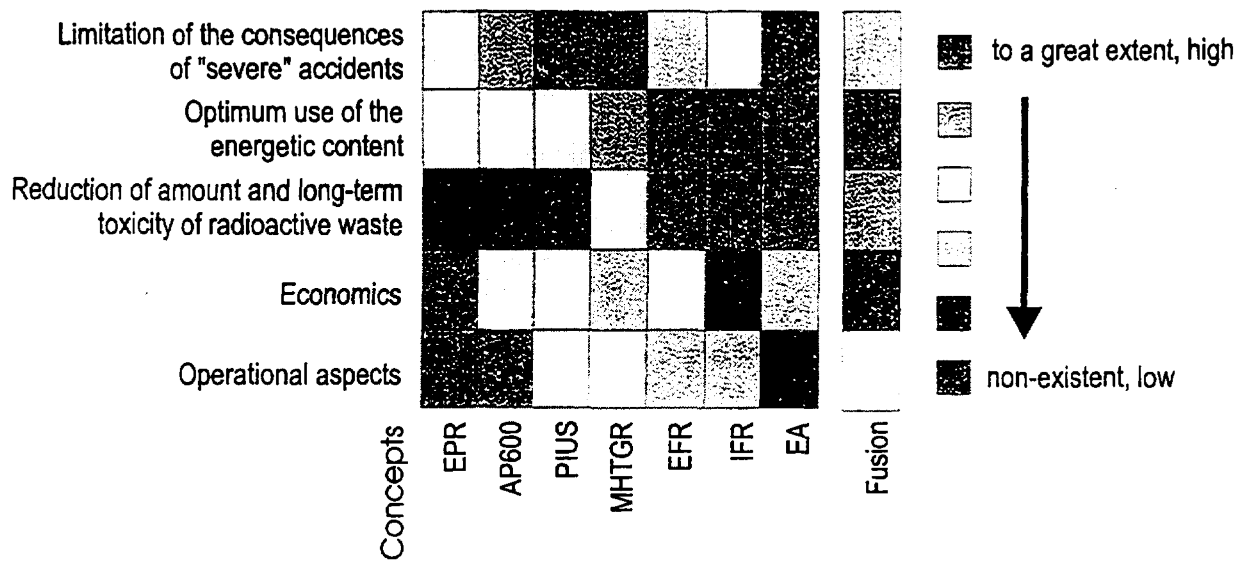
Problems relevant in the long-term are the radiotoxicity of the fuel and the long-term risk related to a final repository, which can be steered by an adequate choice of fuel cycle and reactor type. Moreover, it is possible to "transmute" very long-lived actinides and fission products into less toxic or stable nuclei by means of specific nuclear reactions. Following figure summarises these options for the back-end in the case of the uranium-plutonium fuel cycle.



The most important criteria for the assessment of the evaluated concepts are:

- **safety**, focusing on the limitation of the consequences of "severe" accidents beyond to-day's design basis,
- **use of the energetic content** of the uranium/thorium reserves,
- amount and radiotoxicity of **radioactive waste**, final repository risk,
- **economics** (investment and operating costs), and,
- **operational aspects** (experience, technical development status, robustness).

The results of the study with regard to these criteria can be represented in a matrix as follows, where present western technology is taken as departure and reference point:



In summary, the study leads to the following main findings:

- LWRs (with a thermal neutron spectrum) have a safety potential that should not be underestimated and which can be further exploited. This could allow them to fulfil all expected safety requirements in the foreseeable future. Exhausting this potential by innovative means (e.g. in PIUS) is, however, coupled to considerable economic penalties, as it implies smaller plants, eventually with lower power densities. In order to further improve the use of resources and to reduce the amount of radioactive waste, the fuel cycle should be further closed, which is only partially possible with a pure LWR-strategy.

- Reactors with a fast neutron spectum allow to perform a "quantum leap" in the use of resources (factor of ~100) and can further defuse the waste problem, as they allow the fuel cycle to be fully closed. They fulfil, therefore, an important postulate for a sustainable development. The quantum leap implies, however, a more expensive and complex technology (e.g. liquid metal cooling technology). By steadily exploiting advantages like the low coolant pressure, new developments (IFR, EA) could probably reach a safety standard that is at least equivalent, or even better than the standard of the "best" LWRs.

- Newly emerging fuel cycle technologies, will allow to keep the problem of high level waste under control, even with an increased and long-lasting deployment of nuclear energy. The reduction of the radiotoxicity and, thus, also of the long-term risk should take place primarily by minimising the amount of uranium, thorium and actinides in the waste. A reduction by a factor of 10 to 50 can be probably achieved with improved reprocessing and recycling technologies. At a similar level of fuel losses, switching from the uranium-plutonium cycle to the thorium-uranium cycle would reduce the toxicity of actinides for decay times up to 10000 years, but would not bring any advantage for longer decay times. The main issue being the best possible way of closing the fuel cycle, the improved fuel technologies are important mainly within the context of fast reactors (including the EA).

- Once plutonium is fully recycled and the use of uranium "waste tails" (from the enrichment and reprocessing process) becomes economically acceptable, the closing of the fuel cycle can be considered for the less abundant actinides Np, Am and Cm (transmutation) as well, thus further improving the sustainable character of nuclear energy. With a strategy relying mainly on fast reactors, the recycling of the less abundant actinides can take place in these reactors. However, combined strategies with LWRs and fast reactors in symbiosis with hybrid systems can be envisaged. In all these considerations, one should also keep in mind the radiotoxicity and the long-term risk of fission products, which depends only on the produced thermal energy. For the remote future, possibilities to transmute particularly long-lived fission products can be anticipated.

- With regard to proliferation, the INFCE study of the IAEA from 1980 has shown, that there are no technical means that allow to make the nuclear fuel cycle completely "proliferation-proof" and that the thorium cycle cannot resolve this problem either. Of primary importance is the question, whether the proposed fuel cycle foresees reprocessing or not, the latter being considered more favourable from a proliferation point of view. If one considers reprocessing for the aforementioned reasons to be nevertheless desirable, an (expensive) integration in the reactor plant, like in the IFR-concept, would be certainly advantageous. The increased build-up of higher isotopes through multiple recycling makes plutonium less attractive for theft, but does not provide an absolute protection.

This study confirms, that nuclear fission as a physical principle with the corresponding technology has chances also for the future: The potential to fulfil requirements even beyond the present ones is not exhausted. Developments are underway. The ultimately necessary industrial scale is, however, present only for the rather evolutionary concepts (e.g. EPR). A steady continuation of the developments under discussion, implies further important investments and times, which are the larger, the more the concepts are novel and the closer they still are to the "blueprint" stage. Similar considerations hold for their realisation chances: Cost-relevant additional requirements jeopardise competitiveness. The more a concept deviates from today's standards, the more a political will for a break with – presently strengthening – market rules (deregulation) is necessary. However, differenecs in economics between concepts are in general of the same magnitude as between single and multiple unit plants of the same concept.

The EA has, along with PIUS, MHTGR and also IFR, the largest potential to fulfil the criteria used in this study – as far as this can be assessed in the present early stage of its concept development. However, its competitiveness is at present still rated rather low; at least another 20 years are estimated to be necessary before the concept can reach commercial maturity, and this will necessitate considerable (public) funds. With this in mind, one should not overlook the potential of more "traditional" concepts that are closer to realisation to fulfil the criteria used, especially with a well-aimed use of their specific characteristics and in combination with fuel cycles tailored to future needs.

2. LMFBR Physics Research

2.1 Introduction

PSI's efforts in the area of LMFBR physics research and development were centred on three main topics.

- Adjustment of our deterministic calculational capabilities in view of their use in neutron physics analyses of plutonium burning fast reactor cores.
- Validation of aforementioned methods and data through comparisons with Monte Carlo results.
- Contributions to the development and validation of photon heating calculational capabilities within the framework of the European Code System ECCO/ERANOS [2, 3].

All our LMFBR physics R&D activities were pursued in close cooperation with our colleagues at CEA Cadarache.

2 Main Results

2.1 Adjustment of the Deterministic Codes

PSI's deterministic calculational path for fast systems is based on the cell code MICROX-2 [4]. The use of this tool for plutonium burning fast systems – i.e. fertile blanket lacking cores surrounded by important steel reflector regions – has revealed a few shortcomings of the code: both methods and coding approximations, as well as shortcomings in the use of the MICROX-2 results in the overall calculational route introduced unacceptably high uncertainty levels.

Two major methods adjustment topics have been dealt with:

a) Consistent cross sections calculation for discrete ordinate transport theory codes (S_N-codes). With regard to the broad-group cross sections generated by MICROX-2 for use in S_N-codes, two important improvements have been implemented:

   - the Legendre-moment dependence of the total cross section is consistently included through order P_3;

   - in addition to the "diagonal transport" approximation, three more accurate approximations (i.e.: Bell, Hansen and Sandmeier (BHS) or "extended approximation", "inflow transport approximation", and "inconsistent P_N approximation") have been introduced in order to correct the broad-group cross sections for the effect of the first neglected Legendre-moment

b) P_N spatial weighting

   The available MICROX-2 version did not treat properly the spatial dependence of flux moments higher than Po, particularly in the case of very heterogeneous, and optically very thick or thin regions. This, in turn, made the preparation of proper cell-averaged scattering matrices for Legendre-moments higher than Po impossible. Therefore, new spatial homogenization equations were implemented in MICROX-2, yielding accurate cell-averaged transport cross sections, also in the heterogeneous / high leakage cases mentioned above.

2.2 Validation Work

The validation effort was pursued in close cooperation with CEA Cadarache. Its aims were two-fold: comparison between our deterministic route (MICROX-2/TWODANT) and the Monte Carlo (MCNP-4A) one, on the one side, and support for the validation effort on the European code system ECCO/ERANOS, on the other side. The work concentrated on three numerical benchmarks derived from the ZONA-2 series of the CIRANO experimental program (performed in the zero power facility MASURCA at Cadarache). For detailed results, see references [5, 6, 7]; here only a brief summary of the main findings is given.

The influence of the basic data was studied with the stochastic route (MCNP-4A employed continuos energy data based on ENDF/B-V, ENDF/B-VI and JEF-2.2). The deterministic results were obtained for JEF-2.2 data.

Regardless of the methods and data used, there is an increasing trend for the eigenvalues obtained for configurations having higher steel content (i.e. enhanced characteristics). However, this increasing trend is more pronounced in the deterministic route. The reason for this effect is not yet fully understood; it might be due to a different approach in the calculation of the slowing down source in the steel regions.

With regard to the effect of the basic data, it has been found that ENDF/B-V and JEF-2.2 based k_{eff}-results agree within the calculational errors, while the ENDF/B-VI values are slightly higher.

Finally, the comparison of flux traverses from deterministic and stochastic calculations show very good agreement.

2.3 Photon Heating Studies

This activity is also placed within the framework of the CEA/PSI cooperation agreement.

Given the importance of the photon heating contribution in plutonium burning fast reactor cores, a comprehensive validation work is under way for ECCO/ERANOS. Within this framework, PSI has contributed with both an experimental and an analytical effort. On the experimental side, thermoluminescent detector (TLD) measurements were performed in some of the aforementioned CIRANO configurations. With regard to methods/calculational aspects, an important effort was necessary to produce consistent neutron kerma factors and photon spectra data (from photon production due to radiative neutron capture, fission, and both elastic and inelastic neutron scattering).

First analysis work of the CIRANO experiments has started and is now well under way.

3. Conclusions

The review of PSI's activities within the framework of the fast reactor physics R&D has clearly shown that the main thrust for this work was to contribute to the understanding of the neutron physics characteristics of plutonium burning fast cores. The removal of the fertile blanket zones, the increase in plutonium content and the introduction of a considerable amount of steel surrounding the fissile zones alter both the operational and safety characteristics of such cores, making this data and calulational methods validation effort necessary.

3. Liquid Metal Thermohydraulics

3.1 Introduction

The LMFBR-related thermal-hydraulics research programme in Switzerland, referring to stratification phenomena between sodium streams of different temperature in the upper plenum of FBRs under decay-heat-removal conditions, was terminated end of 1996.

The main purpose of the experimental part of this programme was to investigate the thermal-hydraulic mixing phenomena between two horizontal fluid layers of different velocities and temperatures, with particular attention paid to the effects of Richardson number (experiments in a wide range of velocity and temperature differences between the two streams) and Prandtl number (use of two fluids water and liquid sodium).

In the analytical part of the programme, different available computer codes have been developed further and adapted in order to be used for mixing layer calculation. To accurately predict the flow fields and temperature distribution in the pools a satisfactory validation of codes, based on a reliable experimental data base, must be performed.

3.2 Short review of investigations

In the first phase of the programme, experiments with water were performed in a test-section made of acrylic glass (WAMIX I experiment). Visual observations, which primarily lead to a better qualitative understanding of the phenomena in the developing mixing layer, were followed by application of different experimental techniques (particularly the laser sheet technique) to investigate generation, formation and interaction of vortices. The visualisation by a moving laser sheet was used in order to investigate the formation of streamwise vortices. Extensive work of the WAMIX I experiment was also related to the design the sodium experiment.

Since 1995 the experimental work has been concentrated on preparation of the second phase of investigations in water (WAMIX II) and providing a comparable testing arrangement in sodium (NAMIX experiment). For measurements of local velocity (see Fig. 1) and temperature as well as their fluctuations in the WAMIX test-section, laser Doppler anemometry and resistance thermometry are applied in a modified test-section. To make these measuring techniques applicable, some modifications on the water loop were also made.

In parallel to the construction and mounting of other components of the sodium experiment NAMIX (Fig. 2) the test-section (Fig. 3) was designed and manufactured. Because of the delay in test-section delivery, the experiment was put in operation late in 1996.

The main objective of the analytical work was to check the ability of general-purpose fluid dynamics codes (like FLOW3D [8] or ASTEC [9],[10]) to reproduce (with reasonable accuracy) the overall development of stratified mixing layers at various Prandtl numbers and to provide a code for direct numerical simulation (modification of code FLOW-SB [11]). Particularly important was the participation of ASTEC as well as FLOW3D in different benchmark calculation tasks. The adaptation of the pseudo-spectral code FLOW-SB for the direct numerical simulation (DNS) of stratified mixing layers was very successful. Its ability to calculate the temporal development of flow structures and temperature fields is particularly important, and because of the possibility to visualise these structures, direct comparison with experimental observations (video records) is also feasible.

3.3 Achieved results and concluding remarks

In extensive experimental investigations, based on the combination of different visualisation and measuring techniques, important progress in understanding of phenomena (generation, formation and interaction of vortices) was achieved (Fig. 4)

The mixing-layer thickness as a function of distance from the splitter plate, Richardson number and velocity ratio was determined and semi-empirical correlations for their calculation were given.

By analysing the development of streamwise vortices, their wave length and their behaviour in relation to the spanwise vortices were determined.

An important influence of stratification on mixing was found. With increasing Richardson number (e.g. increase of the temperature difference between the layers) the formation of vortices is more and more suppressed and the growth of mixing layers reduced.

Measurements of local velocities and temperatures could not be finished up to the end of 1996. These are still under way and the complete results will be summarised in a separate doctoral dissertation.

The experimental results were used for comparison with codes, particularly with direct numerical simulations of the code FLOW-SB.

Detailed information about experimental investigations together with achieved results can be found in different reports (see references [12] to [15]).

The ability of the general-purpose turbulence codes was successfully shown already by using them as a help for the design of experiments, but particularly due to their good performance in different benchmark exercises (see e.g. [16]).

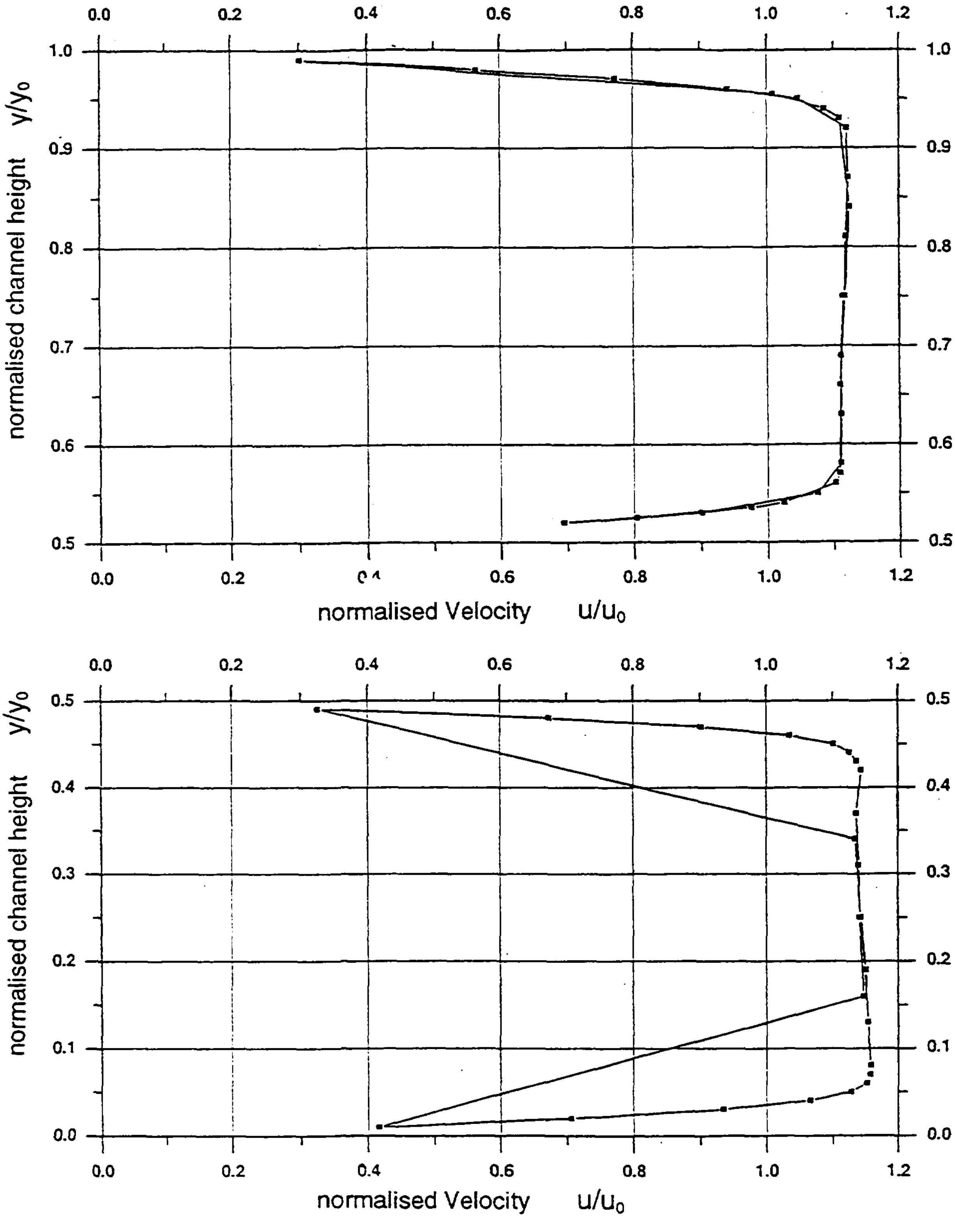
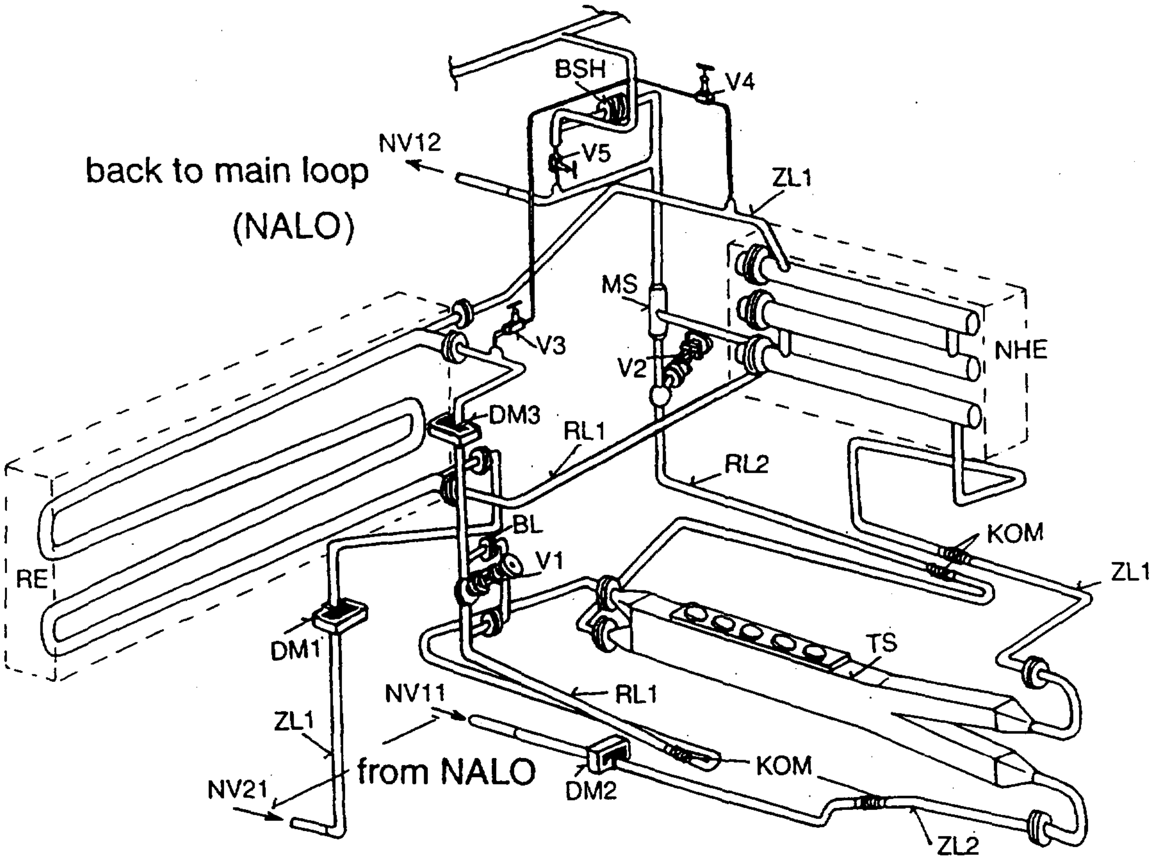


Fig. 1. Velocity (normalised: u/u_0 , where $u_0 = 55$ mm/s) distribution in the subchannels (over the normalised channel height y/y_0 , where $y_0 = 100$ mm), measured in the test section of WAMIX (lines connect subsequent measurement positions).



Legend

BL	- orifice in by-pass	MS	- mixing tee	RLi	- outlet tube i
BSH	- burst disk	NHE	- heating element	TS	- test-section
DMi	- flowmeter i	NVi	- valve i in NALO	Vi	- valve i
KOM	- compensator	RE	- recuperator	ZLi	- inlet tube i

Fig. 2. NAMIX experiment with important components of the experimental setup.

From the direct numerical simulation of mixing layers with the code FLOW-SB, detailed flow and temperature information in space and time can be obtained. A large number of simulations was performed (at relatively low Reynolds number) in a wide Prandtl number (Pr) range (from 0.00535 for liquid sodium, to 6.9 for water), where the Richardson number (Ri) varies from 0.0 (no stratification) to 0.2 (close to critical, stable stratification).

Investigating with particular attention the behaviour of entrainment and mixing in the layer, strong effects on varying Pr and Ri were found. Both quantities decrease with increasing Richardson and decreasing Prandtl numbers.

Some results of these simulations were reported at the EUROMECH 339 conference [17], but more complete information can be found in [18].

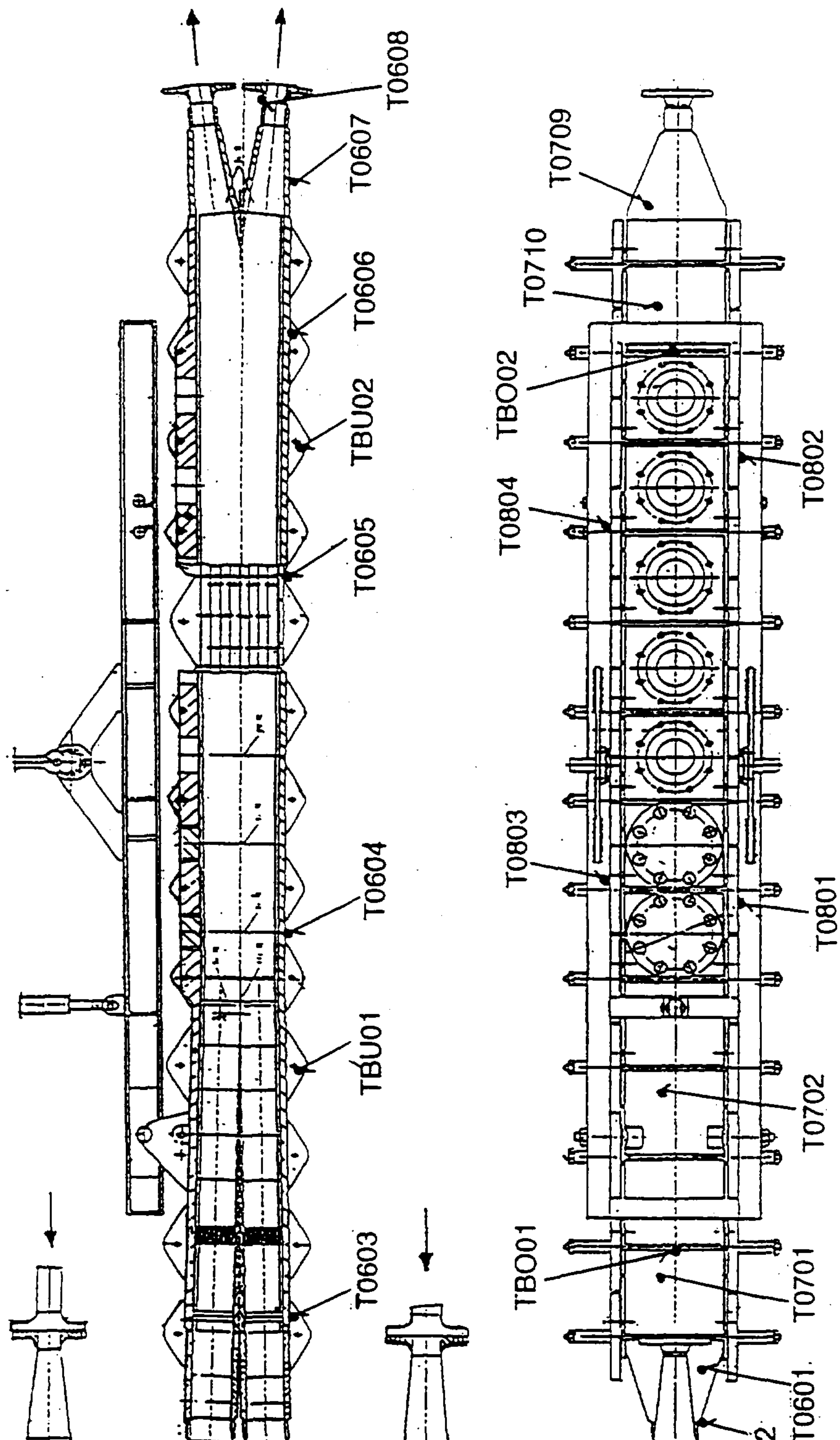
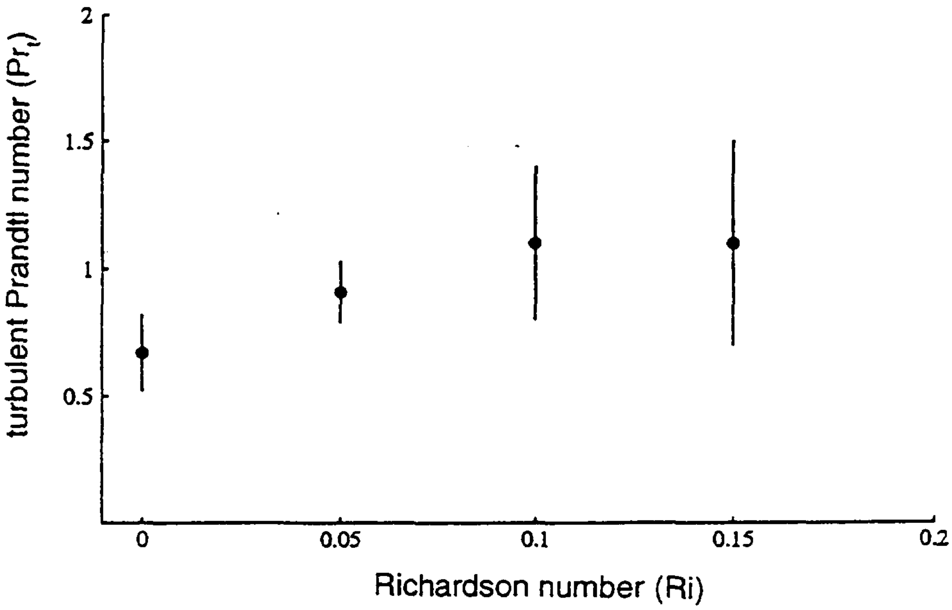


Fig. 3. Test-section of NAMIX experiment with indication of some important thermocouple positions.

a)



b)

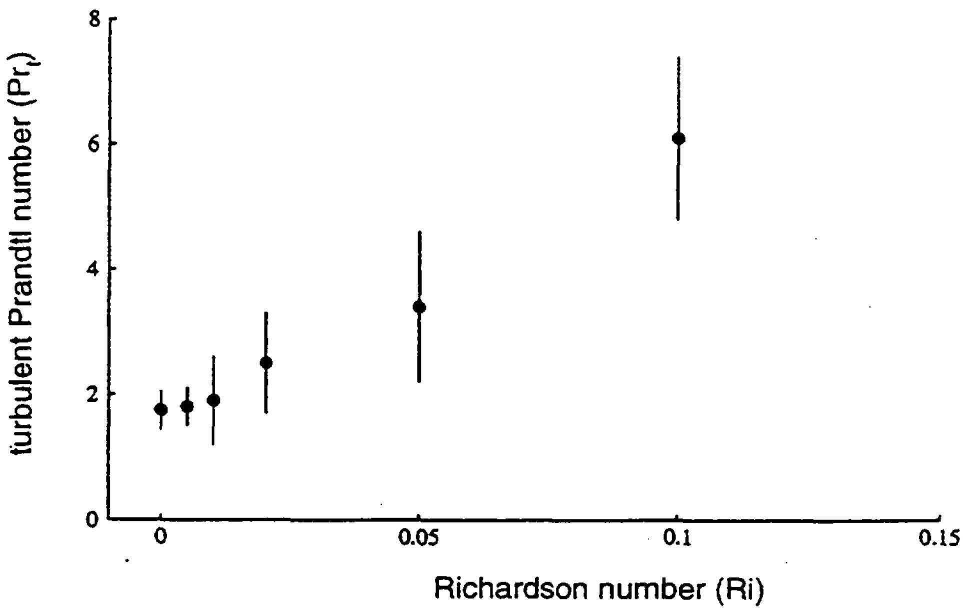


Fig. 4. Turbulent Prandtl numbers for air (Pr = 0.69, a) and for sodium (Pr = 0.00535, b) as a function of stratification (Ri).The values were obtained from the results of direct numerical simulations with the code FLOW-SB.

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A REVIEW OF THE UK FAST REACTOR PROGRAMME

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Abstract

The general position with regard to nuclear power and fast reactors in the UK during 1996 is described. The main UK Government-funded fast reactor research and development programme was concluded in 1993, to be replaced by a smaller programme which is mainly funded and managed by British Nuclear Fuels plc. The main focus of this programme sustains the UK participation in the European Fast Reactor (EFR) collaboration and the broader international links built-up over the previous decades. The status of fast reactor studies made in the UK in 1996 is outlined and, with respect to the Prototype Fast Reactor at Dounreay, a report of progress with the closure studies, fuel reprocessing and decommissioning activities is provided.

1. The UK Nuclear Industry

The latest available statistics on electricity supply relate to 1995. The general trend of recent years continues, ie. increases in generation from combined cycle gas turbine plant and from renewable energy sources whilst the coal and oil contribution is reduced. Nuclear electricity remains an important contributor in Britain, but it is not expected to grow in the near future. Electricity supplied in 1995 comprised:

Coal	47%	
Nuclear	28%	
Gas	16%	
Oil	5%	
Others	5%	(renewables, imports and hydro)

The nuclear component comprised

Total NPP	35
Magnox Units	20
AGRs	14
PWRs	1

Total Net Nuclear Capacity 14168 MW(e)

A total of ten nuclear power reactors have been taken out of service including the experimental and prototype fast reactors (DFR and PFR) at Dounreay and are in various early stages of decommissioning.

The UK nuclear industry has been restructured as a consequence of the Government's Review which was published in May 1995. The main features of the new structure are:

> A new company 'British Energy plc' has been formed but which retains the previous nuclear utilities 'Nuclear Electric' and 'Scottish Nuclear' for licensing reasons. British Energy plc, which was successfully floated on the UK stock exchange in July 1996 owns the fourteen UK Advanced Gas Cooled Reactors and the Sizewell B Pressurised Water Reactor.
>
> The twenty Magnox stations are retained in the newly formed, nationalised 'Magnox Electric' (ME). The intention is that ME will become part of British Nuclear Fuels plc (BNFL) and negotiations are in progress to facilitate this transition.
>
> The former United Kingdom Atomic Energy Authority has been divided into two parts: 'AEA Technology plc' which is a privatised company offering technical services on a commercial basis to other organisations, and the 'UKAEA Government Division' which remains Government owned and which is 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 fast reactor system 'Intellectual Property' previously owned by UKAEA was vested with AEA-Technology on 31.03.97. AEA Technology plc was subsequently successfully floated on the UK stock exchange in September 1996.

BNFL remains 100% Government owned. The company invests a significant amount annually into fuel cycle research and development and under the restructured nuclear industry has sole exploitation rights to FBR fuel cycle technology.

Other key events in the UK during 1996 included:

- The official opening of Sizewell B PWR
- Life extension to 50 years for BNFL and 37 years for ME Magnox reactors

2. <u>UK Activities on Fast Reactors: Progress in 1996</u>

The UK has continued to participate in the European collaboration. Work has been carried out by the appropriate organisations, AEA-T, NNC and BNFL, with funding mainly from BNFL. As in previous years the main activities have been centred on the continued development of the EFR design, and on the CAPRA programme led by the French CEA, which are reported in detail by our French colleagues. The notes below refer to the main UK contributions.

EFR

Further potential improvements to the design of EFR have been examined. As members of EFR-Associates NNC have investigated the advantages of Gas Expansion Modules ("GEMs") as passive safety devices providing negative feedback. They have concluded that GEMs would be useful in unprotected loss-of-flow accidents, but that other devices, for example providing enhanced expansion of the control rod suspension, are effective in a wider range of accident transients. NNC have also participated in work on the effectiveness of a rectangular containment building for EFR.

NNC have also contributed to an investigation of radical alternatives to the EFR design by examining the potentiality of a gas-cooled fast reactor, based on past UK studies and on the extensive UK experience of AGR thermal reactors. There are advantages over liquid metal coolants in terms of in-service inspection and repair, but there may be difficulty in meeting safety requirements.
BNFL is an active member of the European Fast Reactor Utilities Group (EFRUG) and in 1996 hosted the annual exchange meeting between EFRUG and the Japan FEPC. BNFL is currently participating in an EFRUG study of the relative safety of fast and thermal reactors, in which the specification of EFR and the users' requirements for a European PWR are being compared. The work is continuing, but preliminary conclusions are that identical safety criteria could be met by the two systems.

AEA-T and NNC have undertaken several activities to make available operating experience from PFR to the designers of EFR and as a contribution to information exchanges within international agreements involving the European collaboration.

CAPRA

The CAPRA reference core utilises mixed oxide fuel with a high plutonium concentration to maximise the Pu consumption rate. As a result the ratio of cladding damage to fuel burnup is lower than in conventional fast reactor cores, and it may be possible to make use of this by increasing the discharge burnup. NNC have optimised a high-burnup CAPRA core in which the fuel is irradiated until a cladding dose of 180 dpa is reached, and shown that there are potential advantages in terms of improved reactivity control and longer fuel residence time, and that safety criteria can be met. Irradiation testing would of course be needed to confirm that burnups of 25% are possible.

The performance of CAPRA cores has so far been calculated using nuclear data adjusted and validated for use with cores of the Super-Phenix and EFR type, in which the Pu concentration is relatively low. AEA-T have been investigating the use of unadjusted data from the JEF 2.2 set processed via the ECCO cell code. It is important to trace the origin of differences in the calculated reactivity and performance parameters, and to propose routes for validating the calculation procedures.

The highest Pu consumption rates can be achieved only if uranium is eliminated from the core. Nitride appears to be a possible non-uranium fuel material, and the performance of a core fuelled with pure PuN has been studied. AEA-T have studied the vaporisation behaviour of nitride fuels, surveyed the extant data on the physical and chemical properties of PuN and (U,Pu)N, and set up a calculational model of a nitride fuel pin. Preliminary results indicate that acceptable burnups can be achieved provided potential problems of fuel swelling can be solved.

Fuel Cycle Studies

In parallel with the work done in collaboration with the European partners BNFL has conducted studies of the potential role of fast reactors in the UK and elsewhere. It is important to consider the fuel cycle as a whole and to make use of fast reactors in the optimum way to maximise safety and economic advantage while minimising environmental impact and proliferation risks. To this end accelerator-based systems as alternatives to critical reactors, and the thorium cycle as an alternative to the uranium-plutonium cycle, have been examined with particular reference to the implications for fuel fabrication, reprocessing and waste disposal. This work continues but the initial conclusion is that the critical Pu-fuelled fast reactor, properly integrated with reactors of other types, and with optimised arrangements for Pu recycling, has many attractive advantages.

IAEA IWGFR Meetings

The UK continues to support the activities of the IWGFR when appropriate UK expertise is available. In 1996 this included attendance at:

- The Annual IWGFR meeting in Kazakhstan in May
- The TCM on 'Creep Fatigue Damage Rules to be used in Fast Reactor Design', hosted in the UK in June
- The TCM on 'Evaluation of Radioactive Materials Release and Sodium Fires in Fast Reactors'.

3. PFR Closure Experiments

The programme of tests associated with the closure of PFR made use of the opportunity to gain information that could not have been obtained from rig experiments or modelling. The programme was described in some detail in the paper presented at the 1995 Annual Meeting of IWGFR. The studies were in 3 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 studies of steam generator leak detection and tube overheating were completed on 31 March 1995 and were described in the paper presented at the 1995 Annual Meeting of IWGFR. A programme examining materials removed from the reactor was concluded at end March 1996. These examinations were restricted to areas outside the primary circuit, since the funding available precluded the study of active components such as the Above Core Structure or Intermediate Heat Exchangers. The secondary circuit studies were 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.

The programme and the preliminary results were outlined at the 1996 Annual Meeting of IWGFR.

Examination of as-welded joints in the PFR carbon steel sodium storage tanks that were exposed to sodium vapour were carried out to provide some assurance that similar steels, likely to be used for fast reactor roof structures, would not be susceptible to the type of cracking observed in the molybdenum-containing 15Mo3 steel. Carbon steel could then be used in the as-welded state for roof constructions, avoiding the need for costly stress relief heat treatment. The study included ultrasonic inspection of welds on the selected storage tank and removal of three weld samples for dye penetrant and metallurgical examination. No cracking was found. It was concluded that, within the relatively restricted range of temperature (≤150°C) and time (several months) of exposure examined, carbon steels in the as-welded condition can operate successfully in contact with sodium without cracking.

Three welds from a length of secondary sodium circuit (superheater) pipework that had operated at about 550°C and three samples containing known defects in the Superheater 3 vessel shell (all of these being in Type 321 steel welded with Type 347 consumable) were removed for metallurgical examination. This was primarily to study delayed reheat (relaxation) cracking. No cracking attributable to this mechanism was found in the pipework welds. It was concluded that this gave added confidence that this type of cracking is unlikely to occur in the less susceptible Type 316L(N) steel, in conditions of low restraint, during a reactor lifetime.

For the known defects in the Superheater 3 vessel shell, the crack size measured by ultrasonic inspection was, in each case, very similar to the actual crack size found by metallography, although the positioning of the crack was not in good agreement for one location. There was some evidence of crack initiation by the fracture of inclusions and embryonic cracking of the matrix from these. It was not clear if the fracture of inclusions had taken place during operation or prior to operation. Progressive growth of a small defect over 3 to 4 years of operation had been by intergranular cracking. It was not possible to distinguish whether this was by delayed reheat cracking or conventional creep crack growth, as this characteristics is common to both mechanisms. A large crack that had remained stable with little growth over some 4 years of operation was confirmed to be at the weld centreline and has characteristics consistent with a hot tear that occurred during fabrication of the vessel.

A secondary pipework transition weld between Type 321 steel pipework from the superheater and a $2^1/_4$Cr1Mo nozzle at the inlet of the evaporator on secondary sodium circuit 1 was examined metallurgically, including use of transmission electron microscopy. This weld had operated at 490°C. No evidence was found of any mechanism that was likely to have led to failure of this type of weld during longer term operation.

A notable finding of the overall materials study was that there was a significant content of tritium in the secondary circuit components. This was over three orders of magnitude above the 'free issue' level. Very marked partitioning of the tritium content away from the ferritic steel to the austenitic steel took place at the Evaporator 1 nozzle transition weld. The level on the austenitic steel side of the transition weld was three times higher than in the Superheater 3 vessel and Circuit 3 pipework austenitic steel and was at least 5000 times higher than in the adjacent $2^1/_4$Cr1Mo steel.

4. PFR Decommissioning

The PFR core has been removed and replaced by a dummy core as part of the first stage decommissioning. Most of the sub-assemblies with failed fuel cladding have had the pins removed. Although problems were expected with this operation, the pins have so far been pulled cleanly. Decommissioning work is presently concentrating on the preparation of the sodium disposal plant to dispose of the sodium inventory of PFR, and the supporting transfer system, to be ready for operation by end 1998. Construction of the sodium disposal plant began in January 1997. All present major plant engineering work is in support of the sodium disposal phase. A flask is being prepared to transfer components out so that a pump to be put in place to effect the sodium transfer.

5. PFR Fuel Reprocessing

The reprocessing of fuel from PFR has continued. Successful reprocessing of all categories, fuel, axial and radial breeder and residues, has been demonstrated. During 1996, 24 radial breeder and 4 core assemblies were reprocessed. In total, approximately 127 kg Pu was separated from 3654 kg U.

A dissolver leak occurred in September 1996 and an assessment is being made on how this can be dealt with. It is, however, possible to by-pass the head end using the residues recovery plant in order to reprocess non-irradiated fuel.

EUROPEAN UNION: REVIEW OF FAST REACTOR RELATED ACTIVITIES

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Abstract

The European Commission (EC) continued its fast reactor research activities on the same lines as in the past, but with the main emphasis on partitioning and transmutation (P&T) of long-lived radionuclides. The work was carried out by research institutions in the Member States and by the EC Joint Research Centre (JRC) as cost shared actions. The JRC has also been performing its own programme through institutional and competitive research activities. The JRC institutes involved in these studies are the Institute of Systems, Informatics and Safety (ISIS) in Ispra (I), the Institute for Transuranium Elements (ITU) in Karlsruhe (D) and the Institute for Advanced Materials in Petten.
This paper summarises the main activities performed in the field of (i) fast reactor safety and of (ii) partitioning and transmutation.

1. FAST REACTOR SAFETY

1.1. Research Coordinated Activities

The Fast Reactor Coordinating Committee (FRCC) and the Safety Working Group (SWG) pursued, through their subgroup WAC, their task of monitoring benchmark exercises about the Russian BN-800 fast reactor in cooperation with IPPE/Obninsk.

Regarding collaboration with Central Europe and CIS, the EC went on in 1995 with their assistance and cooperation Programmes PHARE and TACIS, which include also Fast Reactor research and development activities.

1.1.1. Safety Working Group (SWG)

After the 42th FRCC meeting in October 1994, where a report was presented about the past activities of the EFR ad-hoc safety club (AHSC), no further meeting of this high level coordination group was held. During 1995 only WAC group meetings were held as it is reported further down.

1.1.2. Whole-core Accident Calculation group (WAC)

Regarding the present situation in the European Union, the following observations can be made. After the high pressure load of the European Fast Reactor (EFR) Project, the interest for LMFBR severe accidents is somewhat reduced. Actually some savings were made in the funding of computer code developments for whole core accident calculations thanks to the coordination activities of the WAC Group which managed to converge all calculational developments in the EU towards one single code, namely the European version of SAS/4A

- which has been adopted also by Japan. As an alternative solution - especially for the modelling of pre-irradiation and fuel pin mechanics phenomena - the computer codes TRAFIC (UK) and GERMINAL (F) have been made available to some of the WAC Group activities in view of specific calculations in parallel with SAS/4A.

Outside the EU, IPPE/Russia and India are developing their own computer tools for severe accident analysis.

As a result of a common interest for developing severe accidents analysis codes, a first common benchmark exercise about BN-800 in its non-zero void reactivity version has been proposed jointly by EU and IPPE at the December 1992 meeting of the WAC Group. In December 1994, a second common follow-up benchmark exercise about BN-800 in its nearly-zero void reactivity version has been proposed jointly by EU and IAEA/IWGFR with the aim of including also India and Japan besides IPPE/Russia.

Unprotected Loss Of Coolant (ULOF) comparative calculations of the Russian BN-800 core with nearly-zero void reactivity.

As far as the WAC group is concerned, the benchmark exercise about BN-800 in its nearly-zero void reactivity version focused on severe transient accident conditions of the ULOF type. At their yearly review meeting of May 94, the IWGFR of IAEA had agreed to support this new BN-800 calculational exercise proposed by IPPE. Participation includes : Germany, France, UK, Italy and Russia 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 calculation of BN-800 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 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 of 1500-2100 MW (thermal). Sodium void reactivity effect is $(0 \text{ to } -0.1)\times10^{-2}$. The number of the zones with different enrichment is 2 or 3. A multi-batch loading scheme is assumed. 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 taken as about 1000 s.

During 1995, the 2nd "Consultancy meeting on IAEA/EC Comparative Calculation for severe accident (ULOF) in a BN-800 reactor" was organized at EC/Brussels (26-28 July 1995) and the 3rd one was held at IAEA, VIC Vienna, on 11-13 December '95, with the participation of FZK, CEA, AEA, IPPE, PNC and HITACHI under the supervision of IAEA/IWGFR.

The first phase of the above mentioned Comparative Calculation, consisting of steps I, II and III, has been successfully executed and is expected to terminate in December 1996, namely:

(1) Case setup
Neutron Physics:
- standard (ie Na, Doppler, stainless steel, fuel)
- structural feedbacks
Thermal Hydraulic:
- S/A + Primary Circuit + Cover Gas
(Specification of End of Equilibrium Core Loading Scheme)

(2) Produce input for codes to be used in the comparison calculations:
- Steady State Calculations for End of Equilibrium Cycle
- Uncertainty assessment and agreement on sensitivities to be propagated

(3) Pre-boiling transient:
- with full circuit dynamics, or
- with pre-specified boundary conditions for core dynamics

This cooperation with IPPE/Russia has proven to be fruitful not only because of the many exchanges of know-how (e.g. physical models) and personnel (e.g. scientific staff delegations to FZK) but also because the western and eastern approaches seem to converge towards one common safety analysis strategy, that is: the accurate validation of numerical models against existing experiments and the development of a single coherent numerical model starting from operational conditions and going from accident initiating event up to late phase boiling phenomena.

As far as the future developments after 1996 are concerned, the second Phase shall consist of the following steps to be executed within 2 years:

(4) Boiling transient:
- with full circuit dynamics, or
- with pre-specified boundary conditions for core dynamics

(5) Post-failure transient (if required):
- with full circuit dynamics, or
- with pre-specified boundary conditions for core dynamics

At the end of the contractual periods of Phases I and II, a final report will be prepared with financial support from IAEA. The report will be approved by all participants prior to publication.

2.2. European Accident Code (EAC-2) at the JRC/ISIS

Safety research is performed at the JRC/ISIS in Ispra.

Accident studies with the European Accident Code (EAC-2) were continued, but at a reduced pace, for the sodium-cooled 800 MWe fast reactor design used in the European WAC benchmark calculation of 1989.

The investigation of safety and more particularly of severe accident conditions is important for accelerator driven systems (ADS). Subcritical ADS could be of particular interest for the actinide transmutation from the safety point of view, because fast reactors with Neptunium, Americium and Curium have a much smaller fraction of delayed neutron emitters (compared to the common fuels ^{238}U and ^{235}U), a small Doppler effect and possibly a 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. In addition, the JRC presented in the past 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.

Computations have been performed for an ADS consisting of an accelerator coupled to a subcritical fast reactor with thorium/^{233}U fuel and natural convection lead cooling. After introducing a source into the point kinetics module of the EAC2 code, Loss-of-Flow accident calculations for an ADS were performed.

These safety calculations for ADSs have since been complemented by a study of reactivity insertion accidents. For an assumed subcriticality of - 3$, reactivity ramp rates of 170, 6, and 0.1 $/s were introduced, leading to a total reactivity insertion of about + 3$. These calculations showed an initially benign behaviour of the ADS (this important safety feature of an ADS had already been found earlier with simpler calculations). However, after tenths of seconds a limited steady state type overpower condition was predicted by the present calculations. In particular the slowest ramp led to a longer-term overpower condition of about 1.5 times nominal. If the accelerator is not switched off or the proton beam interrupted, this overpower will eventually lead to some pin ruptures and fuel sweepout which will stabilise the behaviour of the ADS at a low overpower. This core damage could be avoided by selecting a lower subcriticality of the ADS.

The calculated Loss-of-Flow and reactivity accidents were described and estimates of the behaviour of a lead-cooled fast system and that of thermal ADSs with a circulating fuel/salt mixture were described in the chapter on ADS safety in the IAEA State of the Art (SOAR) report on accelerator-driven systems which will soon be published.

3. PARTITIONING AND TRANSMUTATION

3.1. Research Activities between 1990 and 1994

The study of the potentialities of transmutation of long-lived radionuclides has been included in the fourth five-year shared-cost research and development programme (FWP-4) on "Management and Storage of Radioactive Waste 1990-1994" of the European Commission. Besides, research work on partitioning and transmutation (P&T) was carried out at the JRC/ITU in Karlsruhe.

The implementation of P&T involves research in three areas: (i) partitioning of long-lived radionuclides from the high level waste, (ii) development of fuel and targets containing these long-lived elements in view of their (iii) transmutation in various burners (fission reactors and accelerator driven transmutation devices). The European Commission has partly supported experimental work on partitioning both in the framework of the shared-cost programme and at the JRC/ITU, fuel and target development at the JRC/ITU and an overall strategy study on the potentialities of P&T for nuclear waste management as a shared-cost action.

3.1.1. Chemical Separation of Long-lived Radionuclides

Four European research institutions have investigated experimentally the partitioning of actinides from high level liquid waste (HLLW). In the present studies, this operation is carried out in two steps: (i) removal of actinides and lanthanides from HLLW resulting from reprocessing of spent nuclear fuel in the PUREX process; (ii) partitioning between actinides and lanthanides using soft electron donor extractants. In particular, CEA Fontenay-aux-Roses has been developing for step (i) the DIAMEX process, which uses diamides as extractants.

CEA Cadarache and seven European universities were involved in a research programme to synthesize and test new macrocyclic extractants (crown-ethers and calixarenes). The main aim of this study was to selectively remove caesium, strontium and actinides from medium level liquid waste (MLLW) to decontaminate them to the extent that they can be disposed of in a near surface site. A new calixarene has been synthesized by the University of Parma, which has a cesium/sodium selectivity 100 times higher than that of the best current extractant for cesium. The large selectivity of this molecule has been explained theoretically. The results obtained for cesium extraction from simulated MLLW have been confirmed with real HLLW. Finally, new functionalized calixarenes have been also synthesized, which are more selective to actinides and lanthanides than the best extractant available on the market.

The JRC/ITU has been studying the extraction capabilities and the radiation stability of TRPO, a trialkyl phospine oxide synthesized in China. Experiments were carried out with real HLW from reprocessing of WAK commercial spent fuel. TRPO showed excellent extraction properties for the actinides, the lanthanides and also technecium and high stability in the presence of a, b and g radiation. Similar tests have been carried out for other extractants. In addition, a battery of centrifugal extractors has been installed in one of the chemical hot cells to have a better assessment of the different partitioning processes.

3.1.2. Fuel and Target Development

The activities reported in this section are part of the JRC/ITU own research programme.

The JRC/ITU prepared oxide fuels containing minor actinides for irradiation tests in the fast reactor PHENIX in France (SUPERFACT experiment). The analysis of the irradiated fuels enabled to determine the transmutation rate of minor actinides and the incurred occupational dose during handling of this material. Another experiment, SUPERFACT2, of transmutation of U and Pu oxide with 2% minor actinides is planned in PHENIX.

The irradiation experiment TRABANT (**Tra**nsmutation and **B**urning of **Act**inides in **T**RIOX) was agreed in the framework of the project CAPRA (Consommation Accrue de

Plutonium en Réacteur Rapide). The partners in this experiment are CEA (F), JRC/ITU, FZK (D) and AEA/BNFL (UK). The goal of these irradiation experiments is to demonstrate basic properties, nuclear efficiency and safety behaviour of new fuel types, which are either defined by a very high content of plutonium (> 40 wt%), together with uranium and minor actinides (MA) or by the complete absence of uranium (in order to avoid additional breeding). Irradiation of a mixed $(U, Pu)O_2$ fuel pin with 45 wt% Pu started in July 1995 at HFR Petten and that of mixed $(U,Pu,Np)O_2$ and $(Pu,Ce)O_2$ fuel pins in December 1995.

The JRC/ITU has started a collaboration with CEA (F), ECN (NL), EDF (F) and FZK (D) in September 1992 to set up joint experiments for the study of materials for transmutation, including the fabrication and characterisation of fuels and samples, their irradiation and their in-pile behaviour. The group is called EFTTRA (Experimental Feasibility of Targets for Transmutation). Irradiation tests of technecium as a metal and iodine as 3 inorganic compounds have been carried out in the high flux thermal reactor HFR at Petten (NL). Post irradiation analyses indicate that the material structure of the technecium samples is not significantly affected by the irradiation after a burn-up of about 6%. Irradiation of one technecium sample is continuing in HFR and it is planned to irradiate 3 technecium capsules in the PHENIX fast reactor.

3.1.3. Strategy Studies on the Potentialities of P&T for Nuclear Waste Management

Five European research institutions have carried out strategy studies on Partitioning and Transmutation to assess its benefits for the safety of the management and storage of radioactive waste. The main conclusions obtained are summarised below.

CEA and AEA Technology analysed the potentialities of a strategy for the management of radioactive waste aiming at reducing the inventory of long-lived radionuclides with P&T and assessed its technological requirements and costs. Reference scenarios without and with conventional reprocessing and scenarios using P&T have been compared to assess the potentialities of P&T. The incineration of americium and neptunium was either done in PWRs or fast reactors (FR) in homogeneous or in heterogeneous mode. The fast reactors considered had an electrical power of 1500 MW_e and were either of the EFR (European Fast Reactor) type or of the CAPRA type loaded with MOX containing 45% of Pu.

Compared to uranium and plutonium recycling only, the additional recycling of 95% of americium and neptunium leads to a reduction by a factor of about 6 in the potential radiotoxicity (without barriers) of the waste to be disposed of between 10^2 and 10^3 years (mainly due to Am removal) and between 5.10^5 and 5.10^6 years (mainly due to Np removal). The radiotoxicity of the waste resulting from actinide incineration is smaller for fast reactors than for PWRs. Americium and Neptunium recycling increases the global cost of the overall fuel cycle by 10% to 50%.

Siemens has mainly analysed the potential of fast reactors to transmute actinides and long-lived fission products. The reference fast reactor is EFR with an electrical power of 1500 MW_e, oxide fuel and a 1m core height.

Computations show that it is possible to transmute the amount of minor actinides (MA) produced by about five PWRs with a transmutation half-time of around 10 years in a fast reactor of the EFR type without compromising its safety behaviour. Concerning long lived fission products, the ^{99}Tc production of 5 to 6 1GWe PWRs could be transmuted in a fast reactor with a transmutation half-time of about 25 years.

From safety and burning computations, separation factors between minor actinides and lanthanides around 30-50 are necessary to achieve MA transmutation efficiencies in fast reactors equivalent to those obtained without lanthanides.

ECN Petten has analysed the motivations for transmuting long-lived radionuclides and has established priorities for the nuclides to be recycled. Plutonium should be first on the list because of its very large radiotoxicity and of risks of proliferation. It is at present recycled partly in PWRs as MOX fuel. After removal of plutonium from the waste, americium is the most radiotoxic radionuclide up to 50 000 years. In the framework of this study, it has been shown that the americium radiotoxicity could be reduced by a factor of 20, when irradiating Am during 6 years with a thermal neutron flux of about 10^{14} cm^{-2} s^{-1}. The transmutation of long-lived fission products has the third priority, since the collective dose risk from them is far below the natural dose risk. When comparing the capabilities of heavy water reactors, FRs and PWRs to transmute ^{99}Tc and ^{129}I, the largest transmutation rates and shortest half-times are obtained in FRs; the results of computation are in agreement with those obtained by Siemens, showing that large scale transmutation of fission products in existing fission reactors will be difficult. Finally, the long term (beyond 50 000 years) radiotoxicity of ^{234}U in reprocessed uranium should not be forgotten, because one of its daughters ^{226}Ra is responsible of 60% of the natural radiation dose.

Belgonucléaire has investigated the possibilities and limitations of plutonium and americium recycling as MOX fuel in PWRs. Recycling Am with Pu in MOX fuel leads to an increase in Pu enrichment of the fuel compared to Pu recycling only. The number of recycling steps is limited to one instead of two or three in order to keep the void coefficient negative; more detailed computations are needed to identify the limits for a safe core operation. The dose rates are increased by a factor of 4.5 for the first recycling step; an extra 25 mm steel shielding is required to reduce the dose rates to the values corresponding to Pu recycling only; the additional shielding hinders fabrication and increases the operation costs.

3.2. Research Activities between 1994 and 1998

Nuclear Fission Safety is one of the specific programmes of the Framework Programme for the European Atomic Energy Community (1994-1998) launched by the European Union in 1994. Exploring new fuel cycle concepts is part of this programme with three research tasks: (i) partitioning techniques, (ii) transmutation techniques, (iii) strategy studies. Eight research proposals to be partly funded by the European Commission have been selected in 1995 by the Commission on the basis of an evaluation made by independent experts. An overview of the objectives of these projects and of the research work to be carried out is given below. The JRC/ITU is participating to six of these research projects and the JRC Institute for Advanced Materials (Petten) to one of them.

3.2.1. Partitioning Techniques

Experimental work on new partitioning techniques is performed in the framework of two projects. The first one has the objective to develop processes for the separation of minor actinides from very acidic aqueous solutions containing high level waste without the generation of secondary solid waste. In the second one, extractants selective to strontium, actinides + lanthanides, and actinides only such as calixarene and crown ether derivatives will be synthesised. The extracting properties of these compounds will be determined experimentally and modelled with molecular mechanics and molecular dynamics simulations.

3.2.2. *Transmutation Techniques*

Transmutation techniques are covered by two projects. A transmutation experiment will be performed by the EFTTRA group in the High Flux Reactor at Petten by irradiating a target of ^{241}Am embedded in an inert matrix (spinelle). The project also includes the fabrication of the target and post-irradiation analyses. In the second project, a new method, the adiabatic resonance crossing, which enables to enhance strongly the capture rate of neutrons by the radionuclides to be incinerated, will be developed both theoretically and experimentally. The neutrons are produced by a spallation source using a proton accelerator at CERN.

3.2.3. *Strategy Studies*

Four projects are dealing with the strategy studies. First, a general study will assess different possible P&T scenarios and the technical feasibility of P&T techniques and of advanced fuel and target fabrication. Transmutation of minor actinides and plutonium burning is considered either in over-moderated PWRs or in fast reactors of the CAPRA type. The long term risk and residual dose to man from the waste resulting from these scenarios will be evaluated, when disposed of in an underground repository. The objective of the second project is to provide the previous strategy study with more accurate nuclear data for the scenarios aiming at reducing the waste toxicity in MOX recycling schemes either in PWRs or fast reactors. The relevant nuclear data working libraries will be updated with new information from basic data evaluations and available integral experiments. The third project is an assessment of the thorium fuel cycle to limit nuclear waste production and to burn waste. This study will cover the major aspects of the thorium fuel cycle, i.e. from mining and fuel fabrication up to non proliferation aspects. The last project is related to the assessment of Accelerator Driven Systems (ADS) for nuclear waste transmutation. It is dealing with the Impact of the Accelerator-Based Technologies on Nuclear Fission Safety (IABAT). The objectives of the IABAT project are to perform system studies on accelerator driven hybrid systems, to assess accelerator technology, to obtain basic data on nuclear reaction cross-sections and on radiation damages at the spallation target walls and to study the radiotoxicity of the ADS fuel cycles.

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