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Liquid Metal Cooled Reactors: Experience in Design and Operation

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FOREWORD

In 2002, within the framework of the Department of Nuclear Energy's Technical Working Group on Fast Reactors (TWG-FR), and according to the expressed needs of the TWG-FR Member States to maintain and increase the present knowledge and expertise in fast reactor science and technology, the IAEA established its initiative seeking to establish a comprehensive, international inventory of fast reactor data and knowledge. More generally, at the IAEA meeting of senior officials convened to address issues of nuclear knowledge management underlying the safe and economic use of nuclear science and technology (Vienna, 17–19 June 2002), there was widespread agreement that, for sustainability reasons for fissile sources and waste management, long-term development of nuclear power as a part of the world's future energy mix will require the fast reactor technology. Furthermore, given the decline in fast reactor development projects, data retrieval and knowledge preservation efforts in this area are of particular importance. This consensus concluded from the recognition of immediate need gave support to the IAEA initiative for fast reactor data and knowledge presevation.

To implement the IAEA initiative, the scope of fast reactor knowledge preservation activities and a road map for implementation have been developed. The IAEA supports and coordinates data retrieval and interpretation efforts in the Member States joining the initiative and ensures the collaboration with other international organizations (mainly OECD/NEA) and eventually establishes and maintains a portal for accessing the fast reactor knowledge base.

The IAEA assists Member State activities by providing an umbrella for information exchange and collaborative R&D to pool resources and expertise within the framework of the TWG-FR and the Agency's International Nuclear Information System (INIS) and Nuclear Knowledge Management Section (NKMS). The IAEA collects and summarizes the scientific and technical information on key fast reactor technology aspects in an integrative sense useful to engineers, scientists, managers, university students and professors.

This publication has been prepared to contribute toward the IAEA activity to preserve the knowledge gained in the liquid metal cooled fast reactor (LMFR) technology development. This technology development and experience include aspects addressing not only experimental and demonstration reactors, but also all activities from reactor construction to decommissioning. This publication provides a survey of worldwide experience gained over the past five decades in LMFR development, design, operation and decommissioning, which has been accumulated through the IAEA programmes carried out within the framework of the TWG-FR and the Agency's INIS and NKMS.

The IAEA appreciate the advice and support of the IAEA's TWG-FR members in the preparation of the publication. The draft of the report, compiled by A. Rineiskii (consultant), in cooperation with W. Mandl, A. Badulescu and Y.I. Kim of the IAEA, has been reviewed by the TWG-FR Members of China, France, India, Japan, Republic of Korea and the Russian Federation. The work was guided by the IAEA officers A. Stanculescu and Y. Yanev and they are responsible for this publication.

EDITORIAL NOTE

CONTENTS

INTRODUCTION

Fast reactors have been under development for about 50 years, and several countries have important fast breeder reactor development programs. Ten test fast reactors with thermal power ranging from 8 to 400 MW(th) and six commercial size prototypes with electrical output ranging from 250 to 1 200 MW(e) have been constructed and operated. Two reactors are currently being operated with a thermal efficiency of 43–45%, that is the highest value in the nuclear power practice. Great strides have been made in fast reactor technology in the past, which encourage future development. The closed fuel cycle has been demonstrated and an effective breeding ratio was experimentally confirmed. In total, fast reactors have gained approximately 300 reactor-years of operation. Fuel burnup in excess of 130 000 MWd/t has been reached at several reactors, and major steps towards commercial fast reactor designs have been made.

The worldwide investment already made in the development and demonstration of the unique liquid metal fast breeder reactor (LMFR) technology exceeds US$ 50 billion [4, 5]. Research on LMFRs during the last decades has significantly improved our understanding of LMFR safety. The achievement of the past safety research has been effectively used to develop a system of safety analysis methods which were used to evaluate the safety characteristics of the existing and advanced fast reactors. It is predicted that liquid metal cooled fast breeder reactors currently being designed can achieve a very high degree of safety. A well-validated way forward to commercial utilization of fast reactors has been established. This way is generally consistent with other studies, and indicates that the goal of competitive fast rectors may be within reach.

From the 1980s onward, and mostly for economical and political reasons, fast reactor development in general began to decline, particularly in countries characterised by advanced market economies, relatively slow growth in primary energy consumption and good availability of oil and gas. This has resulted in the premature shutdown of some test and prototype fast reactors and the slowing down or closing of reactor development programs, and the retirement of many of the key developers.

The report documents the knowledge in LMFR design and technology to preserve and disseminate it until the large economic need for fast reactors becomes apparent. The information presented will serve for saving funding required for R&D work in the future owing to preserved data, and will help countries, which plan to embark on their own fast reactor development programs (e.g. some South and East Asian countries). The report intends to provide the public with unbiased information on fast reactor development.

Section 1 presents the state of the art of LMFRs development and IAEA activities. Sections 2–4 provide design features and operating experience of demonstration and commercial size nuclear power plants with sodium-cooled fast breeder reactor. Section 5 provides information on activation characteristics of the primary coolant, reactor and components; treatment and disposal of the spent sodium; removal of the residual sodium deposits and decontamination after shut down of the BN-350 reactor — a typical loop type LMFR. Section 6 provides an overview of lead-bismuth cooled ship reactor operating experience and heavy metal technology development. Sections 7 and 8 provide information on passive principles of fast reactor emergency shutdown and heat removal, and on demonstration of safety characteristics through tests performed in experimental fast reactors during their final stages of operation. Section 9 provides an analysis and assessment of the advantages and disadvantages of sodium as coolant, giving due consideration to the advances in the technology and design of sodium components.

1. LIQUID METAL COOLED FAST REACTOR DEVELOPMENT AND IAEA ACTIVITIES

1.1. ACTIVITIES ON FAST REACTORS DESIGN AND TECHNOLOGY

Studies and design work on fast reactors have been carried out for about 50 years. Nuclear experimental electricity was first generated on 20 December 1951 by 0.2 MW(e) EBR-I LMFR at Argonne National Laboratory (ANL) in the USA. The former USSR fast reactor BR-5 was the first reactor in the world using sodium as a coolant and plutonium oxide as a fuel [1].

Sodium cooled fast reactors have been operated successfully in several countries to produce energy and the sea water desalination, and to demonstrate the fuel breeding. The first in the world demonstration nuclear power plant with fast reactor, BN-350 of 750 MW(th) power was commissioned on 16 July 1973 and has been operated for over 25 years (its design life time), providing 100 000 tons/day fresh water and 150 MW electricity for a large city Aktau (Kazakhstan) and the adjoining industrial region in the desert. The first fuel assemblies containing breeding plutonium recycled into new mixed oxide (MOX) fuel were loaded in the French reactor Phénix and in the UK prototype fast reactor (PFR) cores in 1980 and in 1982, respectively, thereby closing the reactor's fuel cycle. The French fast reactor Super-Phénix of 1 200 MW(e) power, commissioned in May 1986) demonstrated the first large size LMFR [2].

Energy production with fuel breeding is the main goal of fast reactor (FR) development to ensure long-term fuel supply. Fast reactors are also being investigated to reduce the actinide content of nuclear waste, and to take advantage of their high thermal efficiency.

The 65 MW(th)/25 MW(e) China experimental fast reactor (CEFR) is under construction. This is the first step in the Chinese fast reactor engineering development. Ninety percent of the concrete constructions, including the main building, have been completed: hundreds components have been installed in the building. First criticality is foreseen in 2008. As a second step in the Chinese fast reactor technology development effort, a 600 MW(e) China prototype fast reactor (CPFR) is presently under consideration. The role of minor actinide transmutation is also being evaluated taking as reference for the CPFR [3].

The French liquid metal reactor technology has demonstrated a number of positive examples of designs, project realizations and experience in LMFRs construction and operation: experimental reactor Rapsodie (40 MW(th) power, 1967-1983), prototype reactor Phénix (255 MW(e) power, commissioned in December 1973), the large size LMFR Super-Phénix (1986-1998). The plutonium produced in the Phénix was used as the fuel for its core. The cumulative amounts of FR fuel reprocessed in France are about 30 tons. A breeding ratio 1.16 was experimentally confirmed in the Phénix. The Phénix reactor has been operated for ~ 100 000 hours at a temperature of 560°C of the reactor hot structures with thermal efficiency of 45.3% (gross), that is the highest value in the nuclear power practice; an average burnup was increased from 50 000 MWd/t to 100 000 MWd/t, with maximum burnup exceeding 150 000 MWd/t. These levels were reached with 8 cores of fuel which was 166 000 fuel pins.

At Phénix, after completion of the plant renovation program, power operation was resumed on 15 June 2003. The role of Phénix as an irradiation facility was further strengthened to compensate for the premature shutdown of Super-Phénix in 1998, particularly in support of the CEA R&D program in the context of line 1 of the December 30 1991 law on long-lived radioactive waste management (CAPRA-CADRA programme). The first experiment, called SUPERFACT, led to the incineration of minor actinides (neptunium and americium) was carried out. Since 2003, the reactor Phénix power has been limited to 350 MW(th), 145 MW(e) on two primary/secondary loop operations. The reactor will be operated for 6 irradiation cycles of 120 EFPD each. As regards R&D, CEA has launched a comprehensive R&D program to study alternative technologies for future nuclear energy systems including based on a gas cooled FR with on-site closed fuel cycle. Many of the long-term options investigated are believed to be of generic interest, and to offer the chance of developing high performance fuels and materials. Decommissioning work that started in 1999 is underway at the Super-Phénix. The last fuel subassembly has been unloaded on 19 March 2003.

With France in the leading role, the European fast reactor (EFR) design has been completed. This is synthesis of the extensive experiences from France, Germany and the United Kingdom with large pool-type oxide-fuelled reactors. One of the outstanding achievements of the EFR program has been to make firm and reliable cost estimates. The construction of a reactor to the EFR design may not be possible in the near future, but a well-validated way forward to commercial utilization of fast reactors has been established. This way is generally consistent with other studies, and indicates that the goal of competitive fast rectors may be within reach.

The development of LMFR in Europe has been delayed. However, alternative fast reactor applications are being developed in France, Germany, the UK and other countries, namely the transmutation of long-lived nuclear waste and the utilization of surplus plutonium [1-5].

The 500 MW(e) power prototype fast breeder reactor (PFBR) is under construction in India. Manufacturing technology development of the key plant components was completed. Development of technology of low doubling time fuels and structural materials capable of sustaining high neutron fluence has already been initiated and work is going on satisfactorily. The Indian fast reactor development program is built based on the experience accumulated with the small-size [40 MW(th)/13 MW(e)] fast breeder test reactor (FBTR) located at Kalpakkam, which is operational since 1985. Important works including PFBR shielding experiments, testing of transfer arm in air, boron enrichment, post-irradiation examination of FBTR fuel after 125 GWd/t burnup, structural integrity testing, and reprocessing of carbide fuel are being carried out [3].

The potential of natural uranium resources in India, estimated to be around 50 000 tons, is negligible [about 1 billion tons of coal equivalent (btce)] if utilized in an once-through cycle and the capacity will also be limited to about 10 GWe. If the same uranium along with the plutonium generation in PHWR is invested in FBR, the resource potential can be enhanced to 180 btce and the capacity also can be increased to 250 GWe. FBRs thus forming the second stage of the nuclear power programme of the country. FBRs in India will be deployed on U-Pu cycle for rapid growth of nuclear power capacity and to generate enough nuclear fuel simultaneously for deployment of Th-U cycle in the third stage of the program. Thorium is abundantly available in India and the resource level of 320 000 tons is estimated to be equivalent of 1000 btce. Therefore one of the major alternatives for India is nuclear energy.

The fast reactor development program is a key part of the Japanese policy for greater energy independence. The feasibility studies on commercialised fast reactors cycle systems are in progress. The Japanese R&D are being focused on the design of the candidate concepts and on fundamental tests of key technologies [3]. The prototype reactor MONJU of 280 MW(e) power was stopped temporarily due to a leak in the non-radioactive secondary heat transport system, that occurred in December 1995 during the 40% power pre-operational testing phase. In the Japanese program, it was clarified that MONJU is at the core of the fast reactor research activities, and steps are taken to resume its operation as soon as possible. Considerable effort has been put into activities aiming at regaining public understanding and acceptance. The local governor of Fukui made pre-consent for plant modification work of Monju on 7 February 2005. And the Japan Nuclear Cycle Development Institute (JNC) started the preparatory work and the plant modification work on 3 March and 1 September 2005, respectively. In Japan the experimental fast reactor Joyo has shown excellent performance for more than 26 years. The pre-service inspection for the JOYO MK-III upgrade by the relevant Ministry was completed on 27 November 2003. The JOYO reactor has successfully completed and tested the plant and core modifications for the MK-III upgrade program, and rated power operation was started in 2004. The upgraded MK-III core provides a significantly enhanced irradiation testing capability compared to the MK-II core. Initial critically of the MK-III core was achieved on 2 July 2003, which was followed by the successful operational demonstration up to the rated thermal power of 140 MW. Functional and performance testing verified the design parameters. The utilization plan for future fuels and materials developments and safety testing in the JOYO MK-III core has been developed [2, 3].

In the Republic of Kazakhstan, the fast breeder reactor BN-350 at Aktau commissioned in November 1972, was finally shut down in April 1999. The general plan for the BN-350 decommissioning was developed in close cooperation with Russian, US, and UK specialists within the framework of a relevant project. Operations on primary sodium drainage (510 tons, ~ 19 hours) into special tanks have been completed in December 2003. The project EAGLE is under way since 2000 under a contract between the National Nuclear Centre of Kazakhstan and JNC. The project comprises the preparation and conduct of out-of-pile and in-pile experiments designed to address the key safety issues relevant to eliminating or mitigating the re-criticality potential during a postulated core-disruptive accident in future commercial sodium cooled FRs [3].

Republic of Korea's LMFR program consists in the development of basic design technologies. During Phases 1&2 (1997-2001) of the program, basic technologies and the conceptual design of KALIMER-150 of 150 MW(e) capacity has been developed. Basic key technologies and the advanced concept KALIMER-600 with a capacity of 600 MW(e) is being developed during Phase 3 from the year 2002 to 2004. During this phase, efforts were concentrated on the development of basic key technologies and on the establishment of advanced concepts with emphasis on proliferation-resistant core design, and on the enhancement of economics and safety. In 2003, the preliminary KALIMER-600 design concept was established and several experiments were performed for the validation of computer codes and models. A core without blankets, but maintaining the capability of transuranics (TRU) self-recycling is being developed. From the evaluation of the reactor vessel size and decay heat removal capacity (based on the natural circulation of sodium and air) was retained as the most favourable passive design concept. Also in 2003, a conceptual study on ultra-long life cores with power densities higher than the conventional ones has been performed [3, 6].

Russian experience gained on operation of test, prototype and semi-commercial LMFRs (BR-5/10, BOR-60, BN-350 and BN-600) is rather good. The first fast reactor with sodium coolant and plutonium fuel (BR-5/10) was created in 1958 and was in operation for over 44 years. The BN-600 reactor was connected to the grid in April 1980 and full power was reached in October 1981. Reactor operation is stable: its load factor is 75-77%, turbine efficiency being ~ 43%. Up to end of 2004, the total on-power operation of BN-600 reactor plant time amounted to ~ 170 000 h, and ~ 91 billion kWh of electricity was generated. Current efforts with regard to the fast reactor development in Russian Federation are directed towards increasing safety margins and improving economics. The detailed design of commercial size fast reactor BN-800 [800 MW(e)] was completed, and the license was issued for its construction. According to the revised "Programme for nuclear power development in the Russian Federation for the period 1998-2005, and for the period until 2010", the start-up of the 4th power unit equipped with a BN-800 reactor at the Beloyarskaya NPP (BN-600) site is scheduled for 2010. The next important step in the area of fast reactors in Russian Federation is the development of design proposals for a power plant with a sodium cooled fast reactor BN-1800 [1800 MW(e)]. Further activities in the FR area in Russian Federation include:

(1) Justification of life extensions for BN-600;
(2) Justification of a hybrid, as well as full MOX core design for BN-600 to incinerate weapons-grade plutonium;
(3) Design of the BOR-60 experimental reactor modification, including its replacement by the sodium cooled BOR-60M plant;
(4) Development of advanced and innovative FR designs with enhanced safety with mixed oxide fuel, two circuit reactor concepts with sodium reactor's coolant and gas-turbine cycle [both large size reactors (1200-1600 MW(e), and small/medium size modular and transportable co-generation reactor ATES-BN-300/100, 300 MW(e) plus 100 MW(th) for district heating]; and
(5) Design studies of FR with alternative coolants: supercritical water, lead (BREST-OD-300 and BREST-1200), and lead-bismuth eutectic (SVBR-75/100) [2, 3, 7].

Until 1967, the major problem of damage to cladding materials was embrittlement. However, in 1967 the evidence of considerable void swelling taking place in austenitic stainless steels irradiated to high fast neutron fluences, was firstly detected in the UK Dounreay experimental fast reactor DFR (1962-1977). This phenomenon has since then tended to dominate the attention in the development of cladding and duct alloys. A high nickel alloy was developed in the UK as reference cladding material. In the 250 MW(e) prototype fast reactor (PFR), commissioned in January 1975, large numbers of MOX fuel pins reached high burnup with an irradiation dose in excess of 130 displacements per atom (dpa). These results have been confirmed and surpassed by irradiation in Phénix to more than 160 dpa [3]. The first fuel assemblies containing own plutonium recycled into new MOX fuel were loaded in the PFR reactor core in June 1982, thereby closing the reactor's fuel cycle. In March 1994, the Dounrey reprocessing plant had treated a total of over 23 tons of MOX spent fuel with the highest burnup in the fuel about 18%, providing the technical feasibility of MOX fuel reprocessing via a Purex-cycle, with recovery of over 99.5% of the plutonium. This high recovery was also reflected in the low amounts of plutonium in the liquid and solid waste streams from the plant. The amount of radioactivity discharged to the environment was always about an order of magnitude less than the licensed limits. The plant is subject to IAEA and Euratom safeguards.

In the UK, at present there is no Government sponsored FR R&D programme except for the UKAEA PFR decommissioning work at Dounreay. However, a BNFL-funded FR R&D programme involving BNFL/NNC/SERCO Assurance (formerly AEA Technology) is pursued in the CAPRA-CADRA collaboration with France, Germany, and Belgium, covering reactor systems for plutonium and minor actinides burning sodium cooled FRs. The UK contribution is focused on key skills areas: nuclear methodology and core design, fuel performance and fuel cycle modelling. Recently, emphasis is shifting to gas cooled FRs and to accelerator driven system (ADS) analyses. The UK is contributing to core design, thermal hydraulics design, and fuel design and performance of various FR and ADS concepts [2, 3, 8].

In the USA, the sodium-cooled integral fast reactor (IFR) project has been developed by the Argonne National Laboratory (ANL) based on the use of a ternary U-Pu-Zr fuel alloy for its core loading. The IFR concept was integrated by General Electric into a full plant design of a 300 MW(e) advanced liquid metal cooled reactor (ALMR). The plutonium is not separated from higher actinides; these are recycled together in the reactor and never leave the reactor site. All IFR designs are based on full actinide recycling using a pyrochemical processing and fuel fabrication plant co-located with the reactor complex. In fact, by using a U-Pu-10%Zr alloy and ferritic-martensitic HT9 cladding as duct, a burnup of about 20% has been achieved in EBR-II. All irradiation results achieved in the EBR-II and the Fast Flux Test Facility (FFTF) reactor have demonstrated reliable performance of metallic fuel and the potential to achieve high burnup in prototypical fuel elements cooled by liquid sodium. These ideas were borrowed for some other reactor concepts. The ALMR was designed to provide high reliability for the key safety functions, including reactor shutdown, heat removal, and containment. These functions can be achieved by passive means; thermal expansion, temperature effects on neutron absorption, natural circulation of sodium coolant, and natural air circulation [8]. However, in the U.S. from the 1980s onward, and mostly for economical and political reasons the fast reactor development began to decline. By 1994, the Clinch River breeder reactor (CRBR) project had been cancelled, and the two fast reactor test facilities, the FFTF and EBR-II had been shutdown.

In the U.S, there have been substantial activities in continuing the development of technologies related to advanced nuclear energy systems, including advanced fuel cycles, fast reactors and transmutation. The two main programs that cover the relevant activities in advanced fuel cycles and fast reactor and transmutation are the advanced fuel cycle initiative (AFCI), and the Generation IV nuclear energy systems program [2, 3, 9].

1.2. THE JOINT RESEARCH ACTIVITIES ON LMFR

The IAEA has a long standing program to foster international information exchange and cooperative research and development in the field of LMFRs. This program has been carried out during the past 35 years under the expert guidance of a Technical Working Group on Fast Reactors (TWG-FR), formerly International Working Group on Fast Reactors (IWG-FR). The TWG-FR is the only global forum for the review and discussion of LMFR programmes, and is composed of leading specialists in national programs for LMFR development. The TWG-FR was established in order to suggest to the Agency R&D activities that meet Member States' needs and to serve as an international forum for exchanging information and performing collaborative research on fast reactor development. The TWG-FR deals with all aspects of fast reactor technology, including fuel, coolant and components. Some important results achieved through activities carried out within the framework of the TWG-FR are summarized below.

1.2.1. Sodium void effect

In the event of coolant loss (by boiling or gas intrusion) traditional large LMFR cores show a significant reactivity increase. Some experts consider that in the LMFR, an increase of the coolant temperature should initiate an expansion of the absorber rod guide structure, of the fuel in the axial direction and of the core grid plate in the radial direction, resulting in negative reactivity coefficients counteracting the positive sodium void coefficient in large LMFR. However, taking into account that the cooling disruptions and sodium boiling might be on a time scale much shorter than the time scale of the passive negative feedbacks, there is a strong incentive to reduce the positive sodium void coefficient in large LMFR cores. The idea of the IAEA/CEC benchmark calculation 1990/94 [10] was to investigate the capability of reducing the sodium voiding feedback reactivity of an axially heterogeneous core where a sodium plenum is introduced above the core instead of the upper axial blanket.

The benchmark has shown that the overall void reactivity effect of the reference 2 100 MW(th) MOX-fuelled reactor core with postulated voiding configuration might be close to zero. Future investigations were required to determine the differences in severe accident response, in order to estimate the improvement in overall safety that could be achieved from a reduction in the sodium void worth for reactor cores. Knowledge of in-core sodium boiling and voiding physical phenomena was necessary for the determination of the reactivity insertion rate.

There is a feature of contemporary fast reactors which have attracted particular attention on the grounds of safety: positive reactivity transients which may be exacerbated by the fact that the coolant void reactivity coefficient is under some circumstances positive. An IAEA/EC 1994/98 project assessed the capability of reducing the sodium void feedback reactivity of the homogeneous core by introducing a volume of sodium or plenum immediately above the core, as was proposed by Russian specialists in advanced reactors. It has been established by analyses done in various countries in the framework of the Coordinated Research Project (CRP) that, in the event of an accident in which the sodium is overheated, boiling in the core generates vapour which expands rapidly and voids not only the core where it causes a positive reactivity change, but also the plenum where it causes a negative change so that the overall change is less positive or even negative. This approach has been adopted in the BN-800 fuel assembly design (Fig. 1) [11].

Main advantages of the innovative BN-800 type core design are to be seen in providing an additional inherently activated safety margin of preventing fuel pin failure or local boiling in the domain of operational and severe transients to be considered in the design basis. These features complement well the large margin to fuel pin failure achieved already with the hollow pellet fuel pin design and a clad material providing ductility even under high dose loads. In the beyond design basis accident, domain some clear advantages of the innovative core design have been identified:

– Unprotected reactivity initiated accidents most probably lead to an early reactor shutdown either due to pre-failure in-pin fuel relocation and/or due to a rapid fuel dispersal after fuel pin failure in a few subassembly groups. Linear ratings at failure conditions are most probably high, i.e. at about 1000 W/cm and more. Evaluations of the long term coolability of the established core configuration after reactivity initiated accidents were not part of this comparative exercise. They need careful consideration to evaluate potential consequences of a thermally induced subassembly to subassembly propagation.
– In case of unprotected loss-of-flow (ULOF) accidents, the main advantage of the as specified innovative core design is that it is hardly possible to approach or exceed prompt criticality in the initiating phase of the transient. At the end of most of the calculated event sequences, core configurations that needed transition phase analyses were established. Release of thermal and/or mechanical energy cannot be predicted without performing appropriate analyses taking representative results of the initiation phase as initial conditions.

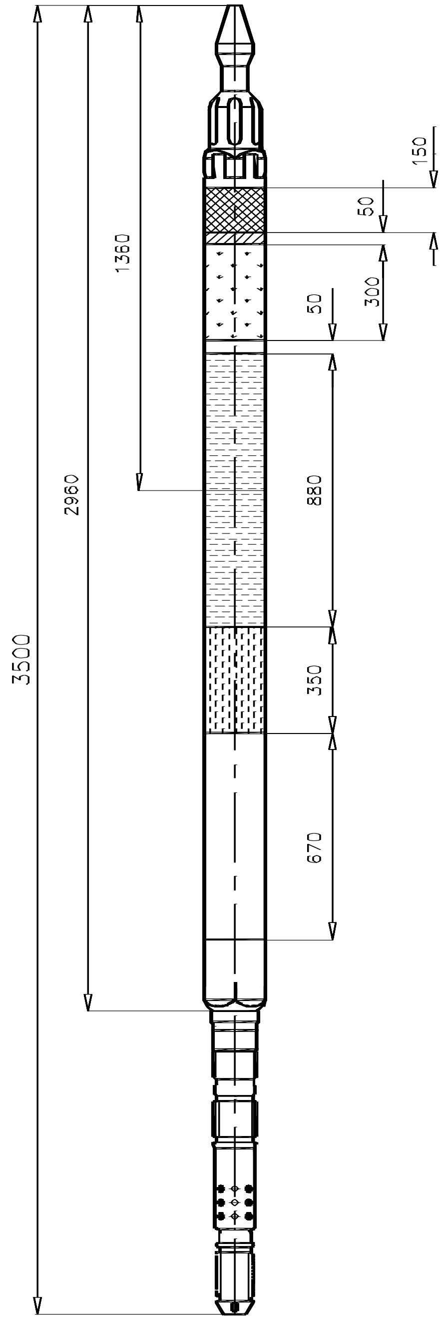


FIG. 1. BN-800 core fuel assembly with MOX-fuel (dimensions in mm, from the top: boron screen (50), sodium plenum (300), core (880), blanket (350), gas (670) [11].

Conclusions drawn from this comparative exercise hold for the specified case set-up. They need to be reviewed when some of the design features are changed or when more detailed evaluations lead to different input data like:

- Magnitude and/or spatial distributions of reactivity feedback coefficients of core materials;
- Reactivity feedback effects due to radial core expansion;
- Fuel pin mechanical properties; and
- If more rapidly developing consequences of control rod drive line expansion could be demonstrated.

It is recognized that there are possibilities for improvement of the analyses and/or for optimization, especially when a more realistic core design would be considered. However, the comparative exercise has shown as well that consequences of these type of modifications need to be analyzed carefully and in detail on a case to case basis. The use of more sophisticated and experimentally validated theoretical models would be helpful to improve the reliability of results. Evaluation of the impact of the as specified core design features on the core behaviour during operational transients was not part of this exercise as well as stability analyses. This would have needed other theoretical approaches to evaluate the potentially involved problems.

1.2.1.1. Methods and codes for transient analysis

Results of this comparative exercise have shown as well that theoretical approaches chosen by India with their PREDIS code package provides comprehensive results for single phase analyses but they use simplified approaches for two-phase flow. Fuel pin mechanics is not yet modelled in transient.

The Russian GRIF-SM code package with the complementary CANDLE-code package provides results for ULOF-type transients up to molten clad relocation. However, it is strongly recommended to couple a transient fuel pin mechanics code package to the system, to develop fuel pin failure criteria considering special features of the BN-800 fuel pin design and to extend the capabilities of the code system to describe core material relocation phenomena after fuel pin failure or break-up. The different code versions of the SAS4A-code family available in Japan, France and Germany allow evaluation consequences of accident initiators leading to core destruction along all stages of the initiation phase up to hexcan melting on the basis of experimentally qualified models. Even these code systems undergo continuous improvement.

In France, the pre-failure in-pin fuel relocation model EJECT is approaching completion with qualification and in Japan coupling to space time kinetics methods is far advanced. Thus, more improved analysis tools will become available in future which provide the possibility of re-evaluating the current results and to follow continuously the impact of new and/or optimized design features of innovative core designs on the results of accident analyses to be considered in the beyond design basis accident domain [12]. The IAEA/EC program on evaluation of benchmark calculations on a fast reactor core with near zero sodium void effect was one very good example of international cooperation. The participation included Germany, India, Italy, Japan 1 (PNC, Mitsubishi, Hitachi, Toshiba), Japan 2 (Osaka University), Russian Federation, U.K., and the USA.

1.2.2. Intercomparison of LMFR seismic analysis codes: comparison of experimental results with computer prediction

One of the primary requirements for nuclear power plants and facilities is to ensure safety and the absence of damage under strong external dynamic loading from, for example, earthquakes. The use of seismic isolation for structures has been gaining worldwide acceptance as an approach to aseismic design. Seismic isolation of important buildings such as nuclear power plants would result in reducing in seismic induced load and, hence lead to economical structural design. Fast reactors operate at high temperature which induces high thermal stresses during transients. Hence the thickness of the structures needs to be minimized to limit the thermal stresses, which contradicts the conventional requirements. This design approach was pursued by adopting seismic isolation as is studied for the ALMR (Fig. 2) and other LMFRs.

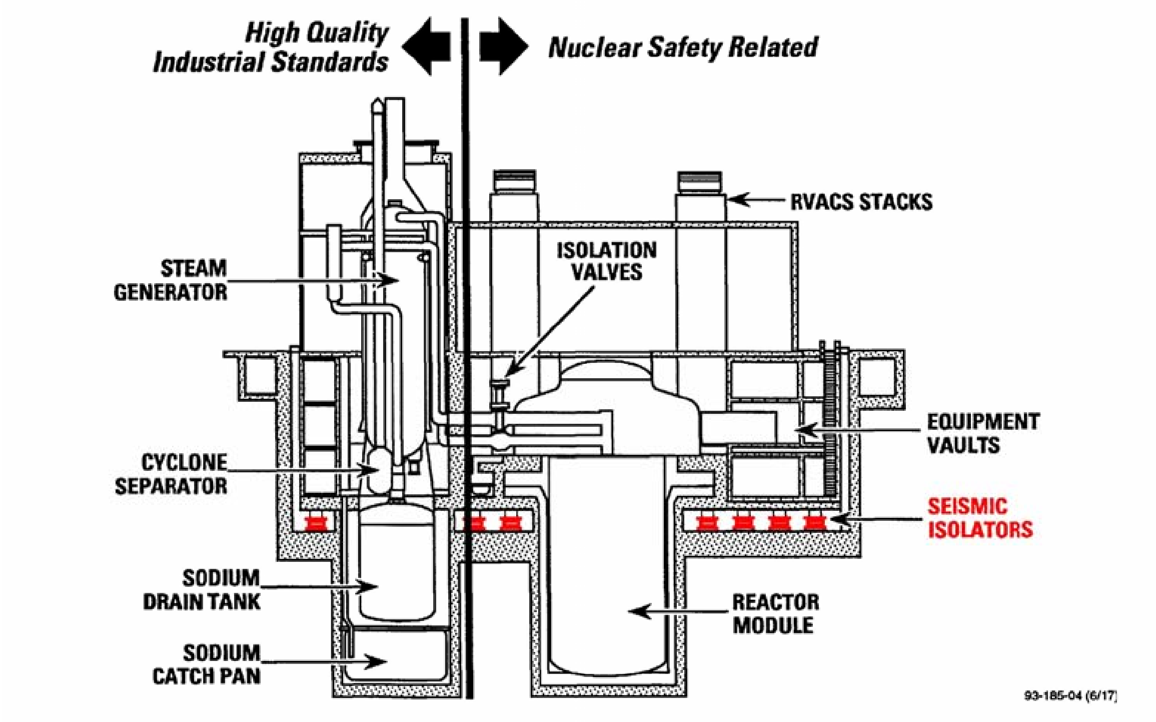


FIG. 2. ALMR: nuclear steam supply system [9].

Therefore, the IAEA through its advanced reactor technology development programme supports the activities of Member States to apply seismic isolation technology to LMFRs. The application of this technology to LMFRs, and other nuclear plants and related facilities would offer the advantage that standard designs may be safely used in areas with a seismic risk. The technology may also provide a means of seismically upgrading nuclear facilities. Design analyses applied to such critical structures need to be firmly established, and the CRP provided a valuable tool in assessing their reliability.

The IAEA has sponsored two Coordinated Research Projects (CRPs) aimed at establishing the reliability of analytical methods and computer codes applied to predicting the behaviour of the reactor core and base-isolated reactor block structures to earthquakes. The studies under the first CRP were useful for the verification and improvement of the reactor core seismic analysis methodologies on the basis of comparison of experimental and numerical results [13]. The second CRP was set up following the good performance of base-isolated buildings [14].

High damping rubber bearings (HDRBs) provide a simple and economical isolation system. They possess the low horizontal stiffness needed and are capable of safely withstanding the large horizontal displacements imposed on them during an earthquake. The need for additional dampers is avoided. In the HDRBs damping is incorporated into the rubber compound. Figure 3 shows a diagrammatic section of an HDRB.

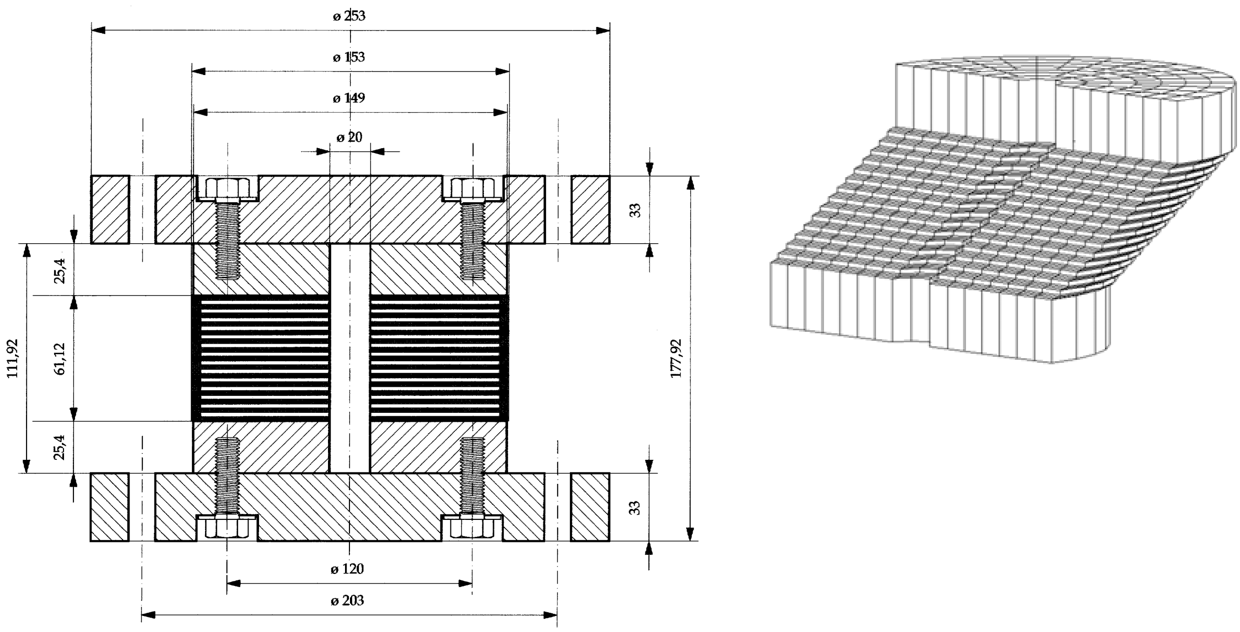


FIG. 3. Sketch of the 1:8 scale prototype of the ALMR high damping isolation bearing; HDRB deformed under a compression and shear strain [14].

Numerical simulation of rubber bearings by code ABAQUS gives satisfactory results as far as material properties are evaluated properly and a suitable strain energy density function is selected. Rubber material can be characterized by using ABAQUS by two important forms of strain energy density functions: polynomial and Ogden. For the cyclic loading tests, numerical simulation using a strain function of polynomial formulation gives better agreement with the HDRB test. However, for the ruptures test, where the displacement is larger than the cyclic loading test, simulation using Ogden's formulation gives better results.

The achieved results confirmed that finite element (FE) methods are useful tools for both the detailed analyses of elastomeric bearings and their design for the better, they allow for a considerable reduction of the number of tests to be performed. The FE analysis of the whole of the rubber bearing carried by participant shoved that the analysis of a single layer of the bearing can be used to predict the horizontal deformation of the bearing by scaling up the results. This is significant in the sense that it reduces the computational time greatly. Further this model can be used effectively to validate the material behaviour. However, more detailed three-dimensional FEM is necessary to analyze the stress distribution of the isolator or to evaluate the behaviour of the bearing at very large deformation.

When the same input data were used, all codes provided predictions of the horizontal force-deformation characteristics of all isolators consistent with the test data. All predictions of vertical behaviour in the absence of horizontal displacement were consistent with the test data. A significant deviation of the numerical prediction of vertical displacement when compression loading is combined with shear was found for some types of rubber bearings.

This deviation is attributable to the modelling of rubber, as almost incompressible, in the constitutive equations.

The modelling of lead proved to be a problem for all teams as the material is deformed in shear within the isolators, while the codes require as input data in tension. Continuing research is needed on the accurate prediction of isolator hysteresis.

The main recommendation derived from the results of the CRP is that the study of isolated nuclear structures should be continued and extended to non-seismic extreme load conditions. The refinement of the characterization of hyper-elastic behavior of the elastomer is needed to predict multi-directional response under combined loading. The modelling of the flexibility of the reinforced plates and connecting plates should be improved. Investigation of the impact on material characteristics of the special environmental conditions of nuclear facilities is needed. Investigation of the finite element prediction of isolator failure mechanisms is needed. Simple, accurate, reliable models for the isolator response over a wide range of multi-directional deformation is essential for accurately predicting floor response spectra and other dynamic design quantities [14].

Future research work should also look at the development of alternative seismic protective technologies such as passive, semi-active and active control for the seismic protection of nuclear facilities and components. The dynamic method to test large scale structures has been validated for base-isolated civil structures and should be extended to isolated nuclear facilities. The influence of vertical ground input on the response of all internal components of an isolated nuclear structure should be investigated.

Fifteen Institutes from India, Italy, Japan, the Republic of Korea, Russian Federation, U.K., and the USA cooperated in this CRP.

1.2.3. Sodium mixing problems

During reactor operation, temperature fluctuations in the coolant close to a structure may occur in many areas such as core outlet zone, lower part of hot pool, free surface of pool, secondary circuit, and water/steam interface in steam generators. In certain conditions, these temperature fluctuations can lead to thermomechanical damage to structures. In 1992 extensive cracking was found in a control rod guide tube that had been removed from the core of the UK Prototype Fast Reactor (PFR).

High-cycle thermal fatigue was found to be the cause of the cracks in the connecting pipe and the middle-stage heat exchange shell at the Tsuruga-2 PWR (Japan) in 1999: two coolant flows – lower temperature main flow inside the inner cylinder of the HE and higher temperature bypass flow outside the inner cylinder were mixed. Repair of the damage interrupted the reactor operation programme. Consequently, knowledge of temperature fluctuations and induced thermomechanical damage to structures is essential to properly support operation maintenance of a nuclear reactor during the plant lifetime. In an LMFR, several areas of the reactor are also subject to this problem.

During normal operation, sodium at low temperature flows into the main pipe of the secondary circuit. A small pipe, connected with a tee-junction to the main pipe, discharges sodium into the main pipe at a temperature which is higher than in the main pipe. In certain conditions, temperature fluctuations in the coolant close to a structure caused by thermal striping can lead to thermomechanical damage to structures. This issue has been encountered in the Phénix reactor in the secondary loop, where initial crack in a tee-junction zone was detected during a campaign of inspections. In 1993, at the BN-600 reactor a sodium leak on the purification loop of primary circuit was observed. Metallurgical expertise showed that it was due to thermal fatigue caused by a fluctuating mixing of hot and cold sodium. There are often difficulties in calculations, because of the complexity of the phenomena involved.

The Technical Working Group on Fast Reactors (TWG-FR) recommended the IAEA to organize a Specialists' meeting on "Correlations between material properties and thermohydraulics conditions, fluid temperature fluctuations and induced thermomechanical damage in LMFR", to identify common trends in the interpretation of experimental and analytical work, and the influence on design features.

The IAEA's specialists meeting held in Aix-en-Provence, France in November 1994, noted that various computer codes for thermomechanical analyses had been and were being developed in some countries. Significant progress has been made in substantiating the phenomenological basis of the material design codes available to reactor designers, but some problems still remain to be harmonized.

Great advances have been made in the last few years to improve calculation methods for predicting fluid temperature fluctuation amplitudes and frequencies, involving large-eddy simulation and direct solution of the Navier-Stokes equation. Within a short time, when improved computing capacity which at present is being implemented, has become available, it should be possible to provide information on fluid temperature at all times and at all important locations. When this level of detail in thermohydraulic calculations has been reached, all the requirements for calculating stresses in structural materials will have been met.

However two problems need subsequent development and harmonization. Firstly, the question of appropriate boundary conditions between more approximate fluid mechanic codes used to predict overall flow properties and codes to predict the fine detail of thermal fluctuations in critical regions has to be harmonized. Secondly, adequate experimental validation including measurements of suitable detail and precision has to be provided.

The above mentioned specialists meeting also stressed that the best way to improve and validate calculation codes and methods would be to organize benchmark calculation comparison in the frame work of the TWG-FR. The TWG-FR following the recommendation of Member States, at its 28th Annual Meeting in May 1995, proposed to perform benchmark analyses in order to verify and validate the thermohydraulic and thermomechanical codes and analytical methods using experimental data.

Institutes from a number of Member States have a great motivation to improve engineering tools and predictions techniques concerning the characterization of the thermal striping effects, in which numerical models have a major role.

Two possible benchmark problems were discussed at the Specialists Meeting in Aix-en-Provence to verify and validate the thermomechanical codes using experimental and analytical data: the square channel with transverse jet, and the t-junction of the LMFR secondary circuit. The former would test thermohydraulic calculations and experiments while the latter would test thermohydraulic and mechanical calculations. Design analyses applied to thermal striping phenomena need to be firmly established, and the benchmark exercise on T-junction of LMFR secondary circuit was approved by the IAEA in 1996 in the scope of the CRP on "Harmonization and validation of fast reactor thermomechanical and thermo-hydraulic codes and relations using experimental data" [20]. The physical phenomenon chosen here deals with the mixture of two flows of different temperatures which may induce temperature fluctuations that results in fatigue damage of the pipe wall.

Eleven institutes from France, India, Italy, Japan, the Republic of Korea, Russian Federation and the UK participated in this CRP.

Work has been done at Framatome-Novatome (France) to provide the participants with experimental data to harmonize and validate codes and methods by comparison of predictions with test results. A set of French experimental data was made available to all the countries participating in the CRP. The participating countries applied their codes and methods to analyze these data. The input data provided come consequently from the actual operation of the reactor. Necessarily, because of the complexity of these data, some have been simplified where possible (i.e. when it was certain that this simplification would not influence the results). Also, the comparison of thermohydraulic results with the actual phenomenon was possible, owing to an instrumentation installed on site during the campaign of inspection. It was agreed that each participant may concentrate his efforts to one or several benchmark problem areas. Geometrical characteristics are:

main pipe	D = 494 mm,	t = 7 mm
small pipe	D = 68 mm,	t = 8.5 mm
circumferential weld	located 160 mm downstream from the tee-junction.	

Operating conditions: no transient to be considered (only nominal steady state); total duration: 90 000 hours.

Through cracks of about 100 mm length were observed on the circumferential weld (160 mm from the centre line of the branch) at almost symmetrical locations on either side of the meridional line. These cracks were noticed after grinding of the external weld bead. The pipe was cut in air to observe the nature of cracks on the inner surface. An immediate appearance of a white spot around the tee (1st plume) was also noticed. While the white spot was due to the constant wetting of hot sodium, the black spot may be due to oxidation due to contact of air in the zone where hot and cold sodium mixing takes place. The cracks appeared on the black spot in the weld adjacent to heat the affected zone originated from the inner surface. Information was provided on the mass flow rates, pressures and membrane stresses. Under the operating conditions temperatures of the sodium in main pipe were 320°C with flow rate of 2 850 m^3/h and the corresponding values for sodium in branch pipe are 410°C and 25 m^3/h. Additionally, 15 thermocouples, mounted upon the external diameter of the main pipe, have given both mean and fluctuating temperatures for relevant positions. In addition to these measurements, a lot of visual inspections and metallurgical examinations have been performed and were available. They made possible to make comparisons with thermomechanical evaluations. After the completion of thermalhydraulic and thermal calculations, each participant to the benchmark was required to:

- Characterize the mixing area, which means locate, define the dimensions of the mixing area, defining the parts of the pipe subjected to the mixing phenomenon;
- Provide isothermal lines (mean temperatures) on the parts of the pipe as defined previously;
- Provide the temperature signals (temperature variations as a function of time) on inner and outer skins of the pipe at points significant, as well as the points noted a in Fig. 1 on inner and outer skins; (the points noted in Fig. 1 locate thermocouples, which are fixed on outer skin of pipes);

— Provide the results of a spectral analysis of the metal temperatures at the points defined above.

The CRP results and recommendations are given in Ref. [15].

1.2.3.1. Thermohydraulics modelling

— The models used in this benchmark show that the pseudo direct Navier-Stokes simulation is an alternative way in case of no subgrid scale model available, and is possible with standard codes.
— The steady-state approaches need a priori assumptions on the frequencies which are therefore not a result of the calculations. Moreover, the finite volume (or finite difference) implementation with precise discretization schemes is the most used.
— The boundary conditions prescribed are mainly steady-state, with spatial distributions in some cases and particular care is taken for the treatment of the wall attenuation.
— The domain size depends on the kind of grid. Non-structured meshes allow to model larger domains and a part of the hot pipe. Local refining is also useful for large domains. The number of computational cells varies significantly, but one can retain the order of 100 000 cells. The physical duration of the simulation remains short and insufficient as regard to the damaging low frequencies.

1.2.3.2. Comparison with Phénix measurements and between thermohydraulics computations

— Different types of flow behaviours are obtained: computation giving flattened jets provides temperatures close to the Phénix's ones for location just behind the tee-junction. For further locations, mean temperature results are sparse. The closest results to experiments correspond to computational domains covering the whole pipe and made of fine meshes.
— An important dispersion is observed for the fluid peak to peak values. The low values are due to too diffusive calculations (k-ε run or finite element run with 1st order in space).
— The longer runs (as regard to the physical time) provide the lower frequencies (this means that the results depends strongly on this time). In all the cases, low frequencies measured on the Phénix are not found, which implies a too high attenuation by the metal wall. The maximum of fluctuation is found further downstream from the tee for the cases where steady boundary conditions are the closest to the tee junction.
— A reasonably good matching with measurements but on the meridian line is obtained for some cases, but the maximum measured peak to peak value of 19 K is not found by any calculation.

1.2.3.3. Thermomechanical assessments

— In spite of the different codes used, the maximum stress range values obtained are consistent. The prediction of cracking is strongly dependent on the fatigue strength reduction factor applied in the method. It appears that if this coefficient is too severe, the parent metal is found cracked which is not observed in reality.
— The geometrical discontinuity formed by the weld bead has to be considered. However, the actual value in a sharp discontinuity is not easily assessable by classical FEM (finite elements method). In that case a σ_d approach could be a good alternative. The crack propagation results are diverging due to the different formulations used for the stress intensity factor variation and due to the presence or not of a mean stress effect.

1.2.3.4. Range of frequencies, sampling

— The range of the damaging frequencies from the wall thickness should be determined firstly:
  - (i) Frequencies lower than this band do not produce sufficient ΔT across the wall,
  - (ii) Frequencies higher than this band cannot penetrate the wall,

— The physical time of computation is deduced from the lower bound of the range, considering that this physical time must cover at least 10 periods of this low frequency. The time step of the computation must also be able to catch the higher bound of frequency. [There can be other and more severe constraints (numeric) for the time step)].

1.2.3.5. Domain and boundary conditions

The main objective is to provide realistic boundary conditions. Some phenomena cannot be modelled by a local thermalhydraulic domain. Hence, the boundary conditions should include:

- Secondary flows (e.g. swirl flow);
- Low frequency variations of temperature and/or velocity.

A parametric study can be a good way to state on the necessity of modelling those phenomena.

The following recommendations are proposed for elaborating the thermohydraulic model:

— Use of an extended domain (mainly upstream) plus prescription of low frequency phenomena at boundary conditions. Mesh must remain very fine in the region of temperature fluctuations: non structured grids and local refinement are therefore very useful;

— An alternative way is the use of realistic boundary conditions from experiments or other computations which can allow the use of a smaller domain;

— If the physical time required (by the range of frequencies of interest) is too long and not compatible with the CPU time, catch high frequencies on the small domain and during a relatively short physical time by assuming to disconnect high and low frequencies, add low variations of temperature to the computed signal of temperature for the thermomechanical analysis;

— The low frequencies (if in the range of interest) may appear in the computation as a combination of the two neighbouring frequencies in case of fluctuating flow near the wall. But, if they depend on a global behaviour (loop scale or plant itself), should be got from measurements (e.g. correlation with flow rate signals). In case of a pipe, the domain must include the full pipe (not half or part of it).

1.2.3.6. Thermohydraulic simulation

— Transient simulation using a Large Eddy Simulation model (or as a pseudo direct Navier-Stokes simulation if no subgrid model is available) is recommended. The discretization schemes must be at least of order 2 in space. Order 2 in time is also better but not completely needed.

— Care must be given to the transient behaviour of the first computational mesh (mesh adjacent to the wall) in association with transient heat transfer coefficient with induced filtering of high frequencies. The effect of small geometrical singularities may not be negligible, but this problem is of secondary priority and cannot be solved before the problem of attenuation on a flat plate.

1.2.3.7. Thermomechanical calculations

— Mean stresses shall be considered in the fatigue damage assessment. The mean stresses influence also the crack propagation. A parametric study is recommended to quantify this influence and the manufacturing residual stresses have to be included in the analysis.
— A weld can be characterized by 3 parameters:
  - (i) The weld bead which is a geometrical discontinuity;
  - (ii) A different material from the parent metal; and
  - (iii) The presence of residual stresses.

These parameters influence the time to initiate and propagate a crack and must be defined precisely. In particular, the factor 1.25 applied on the fatigue rupture curve to represent a material effect is too low to cover all the consequences induced by the 'as-welded' weld. It is therefore recommended to take care of the representation of the weld in thermomechanical assessments:

— Combination of major cycles with thermal striping cycles is necessary to get the total fatigue damage.
— The time to propagate a crack is sensitive to formulation chosen for the stress intensity variation. The strain controlled formulation is preferred to the stress controlled one for this problem.
— A 3D crack could be assessed in relation to the size of the mixing zone involved (here the size of the hot spot).
— The sensitivity of the defect depth on the threshold of the stress intensity factor should be taken into account. The σ_d approach using the random signal if possible could be a good alternative.
— Beyond 70% of the thickness, an analysis of the instability of the remaining ligament shall be performed.

1.2.4. Core structural and fuel materials assuring high fuel burnup

At an IAEA technical meeting on the "Influence of high dose irradiation on advanced reactor core structural and fuel material" [16], it was stressed that high fissile materials burnup is an aim of all current designs because it reduces reprocessing costs and losses of radioactive materials in reprocessing. During more than 40 years of intensive multinational development, a significant experience has been accumulated on fast reactor MOX fuel pins as follows:

— In Europe, more than 7 000 pins have reached burnup values of 15 at%. In addition, some experimental pins (with solid or annular pellets) have attained burnup levels of 23.5 at% in PFR and 17 at% in Phénix;
— In the U.S., more than 63 500 pins with solid pellets have been irradiated in FFTF under prototypical conditions with more than 3 000 pins at 15 at% burnup and with a maximum burnup level of around 24.5 at%;
— In Japan, 64 000 pins with solid pellets have been irradiated in JOYO and foreign fast reactors, with a maximum burnup level of around 15 at% in JOYO and 15 at% in FFTF;
— In the former USSR, presently in Russian Federation, a great experience was gained with vibrocompacted MOX fuel. A record high burnup level of about 35 at% has been

successfully reached with an experimental subassembly in BOR-60 (6 fuel pins) while about 260 standard fuel pins have attained burnt levels of 25-30 at%. More than 4 000 fuel pins with pellet MOX fuel were irradiated in BN-350 and BN-600; maximum burnup in BN-600 was 11.8 at%.

Experts concluded that the cladding rather than the fuel or the wrapper material, provides the greatest limitation in reaching high displacements per atom (dpa) levels and thereby impedes high fuel burnup. Major degradation problems are void swelling for austenitic and, to a lesser extent, embrittlement at low temperatures for martensitic and ferritic-martensitic steels. Maintenance of desirable properties is directly coupled to the maintenance of a stable microstructure against the action of neutron-induced displacements. The most important requirements for such stability are well-defined specifications and well-controlled production methods. There are three paths toward achievement of high fuel burnup:

- Use of low nickel austenitic steel;
- Martensitic and ferritic-martensitic alloys; and
- High-nickel nimonic PE16 alloy.

Of these, the lower-nickel austenite is inherently unstable during irradiation and eventually must swell. If fuel burnup of > 20% is to be achieved in fast reactors, the combination of austenitic cladding together with one of the other two swelling-resistant materials for wrappers may pose a problem.

1.2.5. Concept for fuel resources and waste management

The scope and role of nuclear energy for long term development will be dependent upon its ability to properly utilize uranium resources, to minimize the overall environmental impact, and to contribute to sustainable energy supply. Thermal reactors (e.g. LWR and HWR) operated on once-through uranium fueling extract about 1% of the potential energy of natural uranium, while the bulk of uranium mined ends up in enrichment tails and spent fuel. Advanced thermal reactor cycles can double this extracted energy to about 2%. Because of this relatively low energy yield, nuclear power based only on thermal reactors should be considered as an intermediate step towards sustainable nuclear energy development. It is understood that sustainable use of nuclear power can be achieved only with fast reactors. By using weapons-grade and separated plutonium and plutonium accumulated in spent fuel of thermal reactors, fast reactors could allow the use of the large amount of depleted uranium from enrichment tails. It is expected that hundreds of tons of weapons-grade plutonium will be declared surplus to military needs.

Thus there is a sufficient amount of fissile materials for introduction of commercial fast reactors. Corresponding reprocessing capacity for extraction of plutonium and minor actinides from spent fuel and MOX fuel fabrication capacity is needed. In the long run, the efficiency of uranium energy extraction may be increased up to 60-80%, i.e. the available fissile resources could be stretched with LMFRs by a factor of 60-80. Development and deployment of nuclear reactor and fuel cycle systems for complete extraction of energy from every uranium atom fed into nuclear system would imply that:

- Using only the stock of depleted uranium would, if converted to fissile plutonium, be sufficient to supply the world with energy for hundreds of years; and
- Only shorter-lived radioactive waste fission products would leave the system as an outflow.

It should be noted that none of the advanced reactor systems being developed entirely meets these features, but LMFRs are well suited for still further extracting energy from the ^{238}U and actinides. Thus, in fast reactors a multiple recycle could achieve total destruction of transuranics (plutonium and minor actinides) and selected fission products so that all energy is extracted from uranium atom(s) and only shorter-lived fission products leave the system for returning to the earth's crust. Depending on their core geometrics and compositions, fast reactors at a given power rating and core size can increase, maintain or decrease the inventory of transuranics. Using this flexibility the loading of fast reactors can variously be configurated/composed to produce a transuranic conversion ratio (CR) of less or more than 1.0. If CR > 1 the reactor system would become a breeder and generate fissile materials in response to increased nuclear fuel (power) demand. If CR < 1, the fast reactor would become a burner and could decrease the fissile materials (and actinides) stock.

Long-lived radioactive materials are produced by the operation of reactors of all types and, with the exception of the very few that have commercial applications, have to be treated as waste. By the year 2010, it is estimated that there will be more than 300 000 tons of spent fuel, including 3 000 tons of plutonium and of the order of 100 tons of ^{237}Np and americium [16]. The most intensely radioactive high-level wastes (HLW) from irradiated fuel are either the spent fuel itself, if it is not reprocessed, or the waste streams from the reprocessing plant. The main contributors to the high radioactivity are the fission products and the isotopes of elements beyond uranium in the periodic table (apart from plutonium). The latter are often referred to as minor actinides (MAs).

The long-lived fission products and MAs set severe demands on the arrangements for safe waste disposal, because it is necessary to ensure that they are kept isolated until they have decayed to activity levels at which they pose no danger to the health of people and other living organisms. In some cases this requires secure containment for many millennia. In most countries the policy is to construct waste repositories, which will ensure the adequate protection of the environment for foreseeable future. However, the opinion is gaining influence that it is not right to impose on future generations the obligation to care for the waste products of the present day. If it is eventually decided that it would be better to destroy (or incinerate) the HLW rather than store it, fast reactors will play an important role. MA isotopes can be utilized more efficiently in fast than thermal reactors, because most have a lower ratio of capture to fission for fast neutrons. The IAEA report addresses this approach in more detail [18].

1.2.6. Core disruptive accident

The safety level in an advanced LMFR plant could be further enhanced also by improving reliability of the safety system, by installing passive safety features, and by a plant simplification. However, even given a passive safety regime, one can envisage extremely low probability but plausible scenarios in which the bases envisaging passive safety response are violated (e.g. strong earthquake, and manufacturing flaw stochastic failure). Therefore, core disruptive accident (CDA) will remain subject to investigations and discussions because of recriticality potential and loss of second barriers (vessel) integrity cannot be excluded from consideration. The advanced reactor design, as well as experimental and analytical research on CDAs, should be directed to show achievability of in-vessel retention of the debris generated in the non-energetic initiating phase by demonstrating in-vessel coolability, subcriticality and vessel integrity. As noted at the IAEA/TWG-FR Technical Committee Meeting on Material-Coolant Interaction and Material Movement and Relocation in LMFR [19], the safety analysis of CDAs is supported by three inter-related elements of safety research: identification of key phenomena, improved understanding through in- and out-of-core experiments, and development of computer codes for safety analysis. Several system analysis computer codes have been developed to evaluate CDAs. For example, computer codes SAS4A- and SIMMER series are used as a system of tools for whole core accident analysis.

1.2.7. Acoustic signal processing for the detection of sodium boiling in reactor cores or leak detection and location in steam generators

As all highly-rated heat exchangers, e.g. steam generators, have a risk of failure, the probability of leaks has to be taken into account in the design and operation. The IAEA Coordinated Research Project (CRP) on “Acoustic Signal Processing for the Detection of Sodium Boiling or Sodium/Water Reaction in LMFR” covered acoustic leak detection as well as acoustic boiling noise detection.

It used experimental recordings of leak noise from test rigs and of background noise from operating steam generators, and showed that a 1 g/s leak can be detected within 1 s with high reliability and a false indication rate of the order of one in 25 years. A passive acoustic leak detection system can make use of the same detectors as a plant monitoring system to detect vibration. Active acoustic leak detection uses the changes to the acoustic properties of the steam generator caused by the gas liberated.

The cloud of bubbles formed increases the acoustic attenuation of sodium. This can be detected by means of acoustic pulses transmitted along the length of the steam generator between the tubes. High frequency pulses of around 10 kHz are transmitted from one transducer, along the tube bundle, to a receiving transducer with a pulse repetition rate of around 1 Hz. If there is a significant fall in amplitude of the received pulses, it is concluded that a leak is present. An advantage of the system is that it is fail-safe and self checking [20].

1.3. ACTIVITIES ON ADVANCED FAST LMFR AND TECHNOLOGY

The experience gained with fast reactors has revealed no fundamental problems with reactor physics, operation of various equipments and sodium technology. This experience will be useful in upgrading the design of next generation LMFRs. The objectives of the development of advanced LMFRs are to achieve better economics relative to alternatives, to find optimal solutions for the back-end fuel cycle and to achieve a high degree of safety and reliability.

The final goal relating to the objective of achieving of a very high degree of safety is to design a reactor system which, not only during normal operation but also in case of an accident, could exclude any radiological impact that would require evacuation of the public. The following aspects are considered among the most important [21]:

- Assure stability of the reactor core under all modes of normal and abnormal operating conditions. Minimization of excess reactivity and sodium void effects, the reactor's strong negative power and reactivity feedback with increased temperature, the large margin to reactor coolant boiling at its operating temperatures, low pressure system with large thermal inertia and sufficient safety margins are key aspects;
- Assuring the heat removal from the core and the reactor under all upset conditions, by assuring the ability to limit coolant temperatures below boiling, and fuel element clad temperature below prescribed limits without the need for rapid operator action, i.e. having a reasonably long grace period;

— Take advantage of passive safety systems to provide safety related functions without reliance on operator action or on external mechanical and/or electrical power signals;
— Minimize the burden on the operator of a nuclear power plant. Extensive consideration is being given to man-machine interfaces.

In considering the objective to increase safety and reliability margins, advanced LMFRs are being designed using all previous experience of being both good and not so good. Notable examples of the innovative safety characteristics are: passive decay heat removal systems; passive (inherent) reactor shutdown and stabilization by thermal and reactivity response characteristics of the reactor even under extremely unlikely accident conditions.

As to the argument that some LMFRs faced reliability issues, it should be noted that this would be true for any other reactor line at the initial stage of development, as LMFRs are. Full industrial development of fast reactors has not been completed yet. It is simply too early at the prototype stage of development to more general view of LMFR technology and economics, particularly in the present situation where antinuclear environmentalists trend to discourage investors from entering this field with the required resources. Other reactor technologies, including water cooled reactors, achieved high reliability and lower generating costs when their respective large scale introduction had taken place. One cannot say that this will not happen in the case of LMFRs.

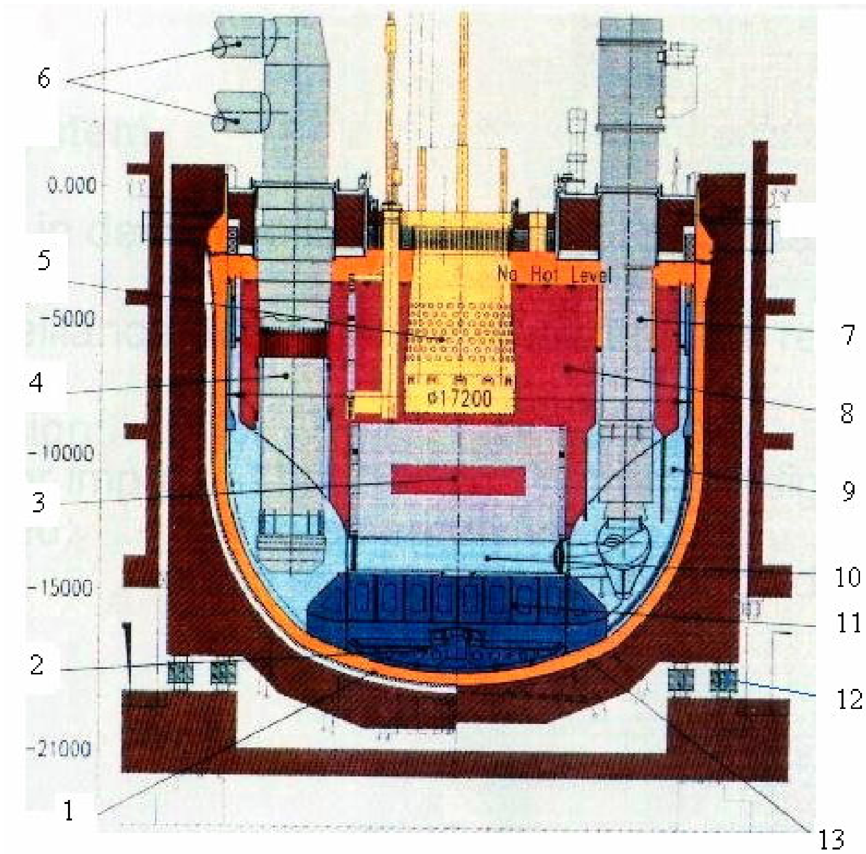
The early development of experimental and prototype liquid metal fast reactors was conducted to a large extent on a national basis. However, for advanced LMFRs, international cooperation begins to play a greater role and the Agency promotes international cooperation — interalia — in its development. Especially for R&D incorporating innovative features, international cooperation can play an important role allowing a pooling of resources and expertise in areas of common interest to help to share the high costs of development. To support the IAEA's functions of encouraging LMFR development, the IAEA promotes technical information exchange and cooperation between Member States with fast reactor development program, offers assistance to Member States with an interest in exploratory or research program, and publishes reports available to all Member States interested in the current status of LMFR development. Experience gained from R&D, operation and construction of fast reactors has been reviewed periodically by the TWG-FR.

The role of the IAEA as a truly international forum for cooperation and exchange of information cannot be overemphasized. About 130 various specialist meetings, seminars and symposia have been organized so far by the IAEA on the advice of the TWG-FR (formerly IWG-FR) and about 3 900 specialists from more than 25 countries have participated in these events. Some examples of topics of Specialists and Technical Committee Meetings held on the advice of the TWG-FR/IWG-FR include:

— Sodium combustion and its extinguishment-techniques and technology;
— Steam generators for LMFBRs, fuel failure mechanisms;
— Fuel and cladding interaction;
— Properties of primary circuit structural materials including environmental effects;
— Leak detection and location in LMFBR steam generators;
— Sodium fires, design and testing;
— LMFBR steam generator integrity and reliability with a particular reference to leak development and detection;
— Repair of FR steam generators: acoustic/ultrasonic detection of in-sodium water leaks,
— Propulsion reactor technology for civilian applications;

— Sodium removal and disposal from LMFRs in normal operation in the framework of decommissioning;
— Operational and decommissioning experience with fast reactors;
— Conceptual design of advanced fast reactor;
— Absorber materials control rods and design of shutdown systems for advanced liquid metal fast reactor;
— Influence of low dose irradiation on the design criteria of fixed internals in fast reactors;
— Use of fast reactors for actinide transmutation, and
— Creep-fatigue damage rules for advanced fast reactor design.

Research on LMFR during the last decades has significantly improved understanding of LMFR designs, technology and safety. Pool type design concept was chosen for all small, medium and large size advanced reactors: CEFR (China), PFBR (India), KALIMER (Republic of Korea), BN-800 and BN-1800 (Russian Federation), and EFR-1500 (European Fast Reactor, Fig. 4), except for DFBR-660 and JSFR (Japan) which uses the loop-type concept.



1-hung safety vessel; 2-debris tray; 3-core; 4-IHX; 5-above core structure; 6-secondary loop; 7-primary pump; 8-hot collector; 9-cold collector; 10-grid plate; 11-core support structure; 12-spring bearing; 13-anchered safety vessel

FIG. 4. EFR: cut of reactor.

The pool reactor placed in a guard vessel has been shown to have very attractive safety characteristics, resulting to a large extent from a liquid metal cooled reactor being a low pressure system with large thermal inertia. This type of design practically excludes unfavourable consequences of failures in the external radioactive systems and loss of reactor coolant [22]. One of the outstanding achievements of the EFR programme is to make firm and reliable cost estimates. Construction of a reactor to the EFR design may not be possible in the near future, but a well validated way forward to commercial utilisation of fast reactors has been established. This way is generally consistent with other studies, and it indicates that the goal of competitive fast rectors may be within reach (Fig. 5).

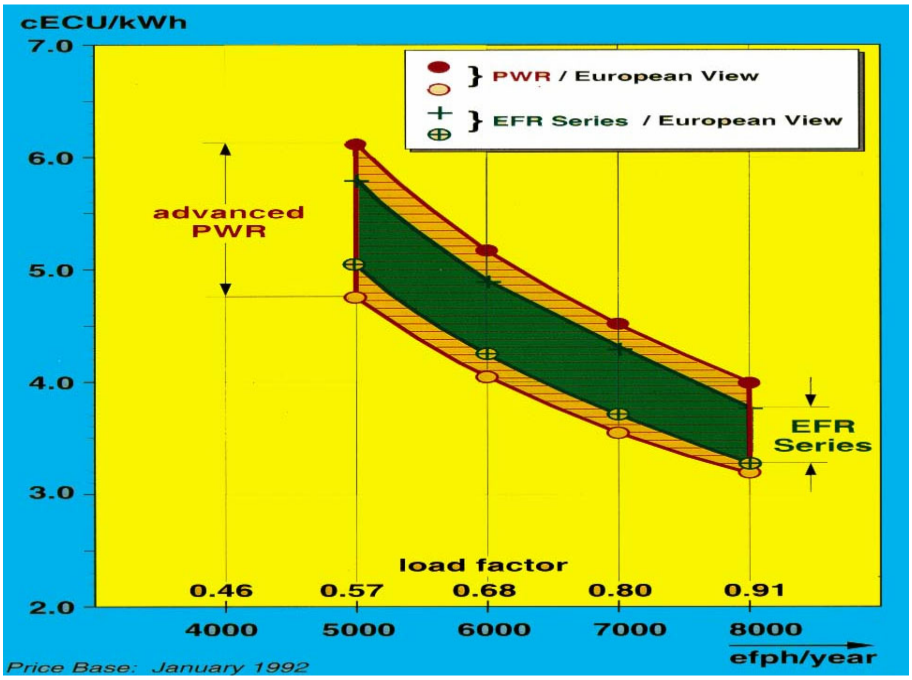


FIG. 5. Generating cost comparison EFR vs. advanced PWR.

Findings of recent work have led to improvement in LMFR designs and safety. For example, an advanced design of the system: main vessel - safety vessel - vault was developed for EFR (Fig. 6).

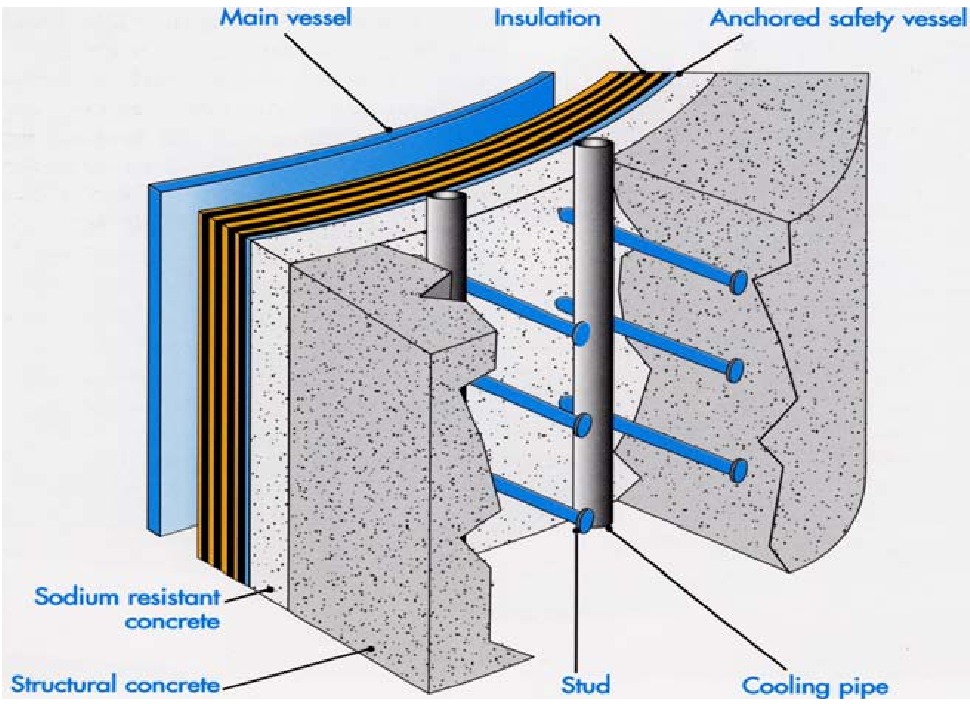


FIG. 6. EFR: safety vessel and vault [22].

The main vessel is completely surrounded by a leak tight safety vessel anchored to the surface of the concrete vault. A layer of metallic insulation covers the inside surface of the safety vessel, which reduces both the heat losses and the thermal cycle skin fatigue. A layer of sodium resistant concrete is provided between the safety vessel and the structural concrete of the vault. The structural concrete is kept cool by the vault cooling system. The top entry loop type design was selected for the Japanese DFBR because of the following considerations [23]:

— Major primary components such as the intermediate heat exchanger (IHX) and the pumps are outside of the reactor vessel, and this facilitates maintenance and repair;
— The system has flexibility to introduce such innovative technologies as the electromagnetic pump integrated component, which is needed for commercialization of the FR; and
— Experience gained at the prototype MONJU must be fully utilized. Considering that the top entry system is quite a new concept, the conceptual design study, the evaluation study of commercialization prospects and the water hydraulic tests using models of thermalhydraulic properties specific to the top entry system were conducted.

In progress of the loop type LMFR design development, Japan is now in a position to embark on an in-depth study of an advanced plant configuration - a compact loop type LMFR design JSFR (Fig. 7).

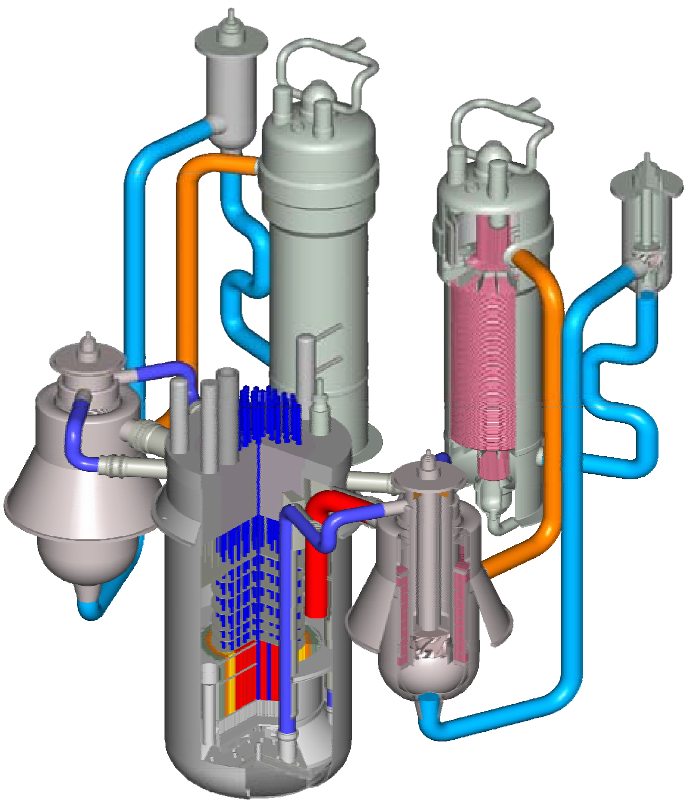


FIG. 7. Bird's eye view of JSFR-1500 NSSS [3].

To achieve the economic target, several innovative technologies and LMFR design improvement measures have been adopted [3]. The reduction of plant material is accomplished by adopting the following technologies:

— Shortening the piping length and reduction of the number loops by adopting 12 Cr steel which has low thermal expansion with high strength;
— Development of integrated intermediate heat exchanger (IHX) with mechanical pump.

Although there are differences in conceptual design approaches, a number of common topics could be identified among the conceptual designs. These include improvements with regard to safety margins, design simplifications and cost reduction.

Safety margins improvements include the consideration of core catcher design for excluding the recriticality, cooling capabilities, passive backup reactivity shutdown, decay heat removal systems and a strong inherent negative reactivity feedback with temperature rise. Economic competitiveness should be improved by the following design efforts: the optimization of the number of cooling loops and equipments, and achievements of reducing its weights as well as the building volumes, findings of structural materials, fuel technology and core design to achieve high burnup, and limiting the number of safety graded systems.

The development of simple, reliable, efficient and flexible systems and components is a primary objective in the design of advanced fast power reactors. Systems and components for developmental design (advanced reactor designs which range from moderate modification of existing designs to entirely new design concepts) in general require much extensive testing and demonstration to verify component and system performance. Key issues are scaling effects for simulating plant configuration, design life and interactions among different systems.

1.4. SAFETY RESEARCH ISSUES FOR ADVANCED LMFR DEVELOPMENT

The TWG-FR has paid much attention to reactor characteristics relevant to LMFR safety. Some examples of topics of Specialists and Technical Committee Meetings held within the framework of the TWG-FR (formerly IWG-FR) include:

— Operational safety of sodium circuits;
— Reliability of decay heat removal systems;
— Role of fission products in whole core accidents;
— Demonstration of structural integrity under Normal and Faulted Conditions;
— Sodium fires and prevention;
— Evaluation of radioactive materials release and sodium fires in fast reactors;
— Passive and active features of LMFRs;
— Material-coolant interaction and Material Movement and Relocation;
— Reactor core with near zero sodium void Effect; and
— Primary coolant pipe rupture event in liquid metal cooled fast reactors.

The meetings with above mentioned topics cover a wide area of LMFR technology and have demonstrated fruitful international cooperation. A series of international conferences, symposiums and topical meetings on fast reactor safety under the sponsorship or in cooperation with the IAEA have been held in the past: Aix-en-Provence (1967), Beverly Hills (1973), Seattle (1979), Lyon (1982), Guernsey (1986), Snowbird (1990) and Obninsk (1994).

1.4.1. Progress made since the past

Minimization of the risk to the public by further improving the safety level of advanced nuclear systems is one of the ultimate goals of safety research. Although large progress has been made since the time that the first prototype LMFRs: the 750 MW(th) BN-350, the 250 MW(e) PFR, the 255 MW(e) Phénix, and 600 MW(e) BN-600 were put into operation more than two decades before, the search for improvements in performance and reliability by design measures has called for continued R&D in the broad area of the LMFR technology with a special emphasis on nuclear safety. To draw the findings of the international meetings on fast reactors and related fuel cycle [25] and advanced reactor safety [23, 25], the following advances have been made:

— The amount of available data on LMFR technology has been expanded significantly. Experimental design and analytical work were carried out for all important components and systems including steam generators, sodium pumps, intermediate heat exchangers, sodium and aerosol leak detection, fuel failure detection, hydrogen detection after failure of steam generator tubes, sodium boiling detection, temperature measurement. The most important achievements have been reached on liquid metal technology and mixed oxide fuels.

— A large number of tests and analytical work has significantly improved the understanding of passive and natural safety characteristics of the LMFR. Two types of passive safety features of the LMFR for prevention and mitigation of core disruptive accident (CDA) are under consideration for advanced LMFRs: heat removal from the core by natural convection; and strong negative reactivity feedback mechanisms to control and/or restrict the core power in emergency situations. To demonstrate the functioning of the decay heat removal (DHR) systems, an experimental and theoretical programme has been carried out with in-core and out-of-core tests in water and sodium in differently scaled systems, including full-scale DHR systems of the EFR. At least 25 experimental rigs have been built in Germany, France, U.K., U.S., Russian Federation, and India to study the characteristics of different DHR systems. As concluded at the TWG-FR Specialists Meeting on Evaluation of DHR by Natural Convection [27], the existing experimental data and the analytical work show that the decay heat removal by pure natural convection is feasible. Concerning the objective of passive safety, DHR by pure natural convection is an essential feature to enhance the reliability of DHR. The rationale of safety strategy for advanced LMFR is to employ completely mechanical and physical laws rather than engineered systems whose reliability is subject to human action. Passive reactivity shutdown systems have been developed and demonstrated for the BN-800 (suspended by coolant flow) and PRISM (gas expansion module in the core). A design study of enhancement of absorber rod drive line expansion was made for the EFR. Above a certain (switching) temperature its thermal expansion is about 3 times more than a natural thermal expansion relative to the core [28].

— Three-dimensional codes to allow calculations of steady states or thermal-hydraulic transients concerning the hot plenum and cold collector, fuel subassemblies and secondary loops have been improved as a consequence of a large number of separate effect tests and detailed numerical studies. Subsequent developments continue to decrease uncertainties, particularly for gas entrainment phenomena, temperature fluctuations for thermal striping and core outlet area modelling.

Research on liquid metal-cooled reactors during the last decades has significantly improved our understanding of LMFR safety. The achievement of the past safety research has been effectively used to develop a system of safety analysis methods which were used to evaluate the safety characteristics of the existing and advanced fast reactors. It is predicted that LMFRs currently being designed can achieve a very high degree of safety. However, in spite of all the progress made on LMFR technology and in particular on safety development, the quest for excellence calls for further work. The IAEA gives support within the framework of its Statute and the available means, on topics to be briefly addressed below. As to the challenges for future advanced LMFR, it is important to realize that all modern nuclear power plants employ the defense-in-depth safety strategy which relies upon maintaining the structural integrity of the three principal barriers preventing release of radioactive fission products: the fuel cladding, the primary system boundary and the containment. Failure of these barriers primarily results from mechanical and thermal loads, and safety research and nuclear power design studies show that significant margins have been provided to avoid such failures. For LMFRs there are a number of safety issues which influenced fast reactor licensing and safety analysis and some of them had been discussed above.

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2. PROTOTYPE FAST REACTOR

2.1. DESIGN FEATURES AND REVIEW OF OPERATING HISTORY

The primary circuit of the prototype fast breeder reactor (PFR) was a pool type and had an inventory of 900 tons of sodium. The equipment was stationary fixed on the upper plate from which the reactor vessel is suspended. This design requires a complex thermal expansion compensation system of skirts, inside which tube bundles of heat exchangers were inserted. For this purpose the PFR reactor has bellows operating at high temperature and being in-maintainable. Compensation of pressure pipelines thermal expansions was achieved by means of the bends. In view of that several pipelines connect the pump to the pressure plenum.

One of the main thermo-mechanical problems concerning the top support reactor design was to create acceptable operational temperature conditions for the upper bearing plate, from which the vessel is suspended and on which the equipment is installed (pumps, heat exchangers, rotating plugs, etc.). For thermal shielding of the bearing plate internal surface in the PFR reactor multi layer steel foil insulation was used, which turned out to be complicated and expensive.

Experimental investigations and analysis of heat and mass transfer from the coolant surface to the upper plate carried out at the design stage revealed a number of interesting effects. These investigations showed that there is a great temperature drop between the sodium surface and the plate (200-300°C). The main part of this drop falls at the region of near-sodium boundary layer and the rest part - at the near-the-plate boundary layer. In the central part of the gas plenum vigorous mixing results in uniform temperature.

The experiments showed an active process of aerosol formation due to sodium evaporation from the surface, its consideration near the surface and sodium particles fog formation in the gas plenum. Thus, in the gas plenum a complex heat transfer takes place as a result of convection, aerosol mass transfer and radiation.

The main mechanism of sodium surface and cover plate heat transfer is thermal radiation. Its intensity to a considerable extent is determined by the emissivity of the sodium surface, cover plate and sodium fog. With allowance for specific features of heat and mass transfer in the gas plenum the analysis showed that at the sodium surface temperature of 520°C the temperature of the plate surface may be 350–450°C. Changes of surface effective emissivity from 0.03 up to 0.1 results in increasing heat flux from 700 to 1250 W/m^2, changes of optical thickness within the range of 0.1–10 have a weak effect on the heat flux value and they influence the gas temperature only at high values of the plate effective emissivity. The radiation-induced heat flux is of prevailing importance. The sodium fog significantly (2.5 times) reduces the heat flux density thus screening the plate from the sodium surface radiation.

On the basis of the experimental investigations and analysis it was concluded that with allowance for fog screening properties in the reactor gas plenum the thermal insulation design could be simplified or due to ascertaining data on radiating properties of the hot sodium surface and that of the cover plate surface.

Figure 1 shows a cross-section of the primary circuit. Heat from the 600 MW(th)-rated core was transported by sodium primary coolant pumped by three electrically-driven (1 MW) mechanical pumps to six intermediate heat exchangers (IHXs), which were grouped in three non-adjacent pairs and each pair was connected with the relevant secondary circuit. Sodium entered the core at a temperature of 400-430°C; the core temperature rise was about 160°C.

The primary sodium was contained in a primary vessel of 18/8/1 stainless steel, 12.2 m in diameter and 14.0 m deep, surrounded by a guard vessel made from medium carbon boiler grade steel. Sodium purity was controlled by means of an external cold trap loops.

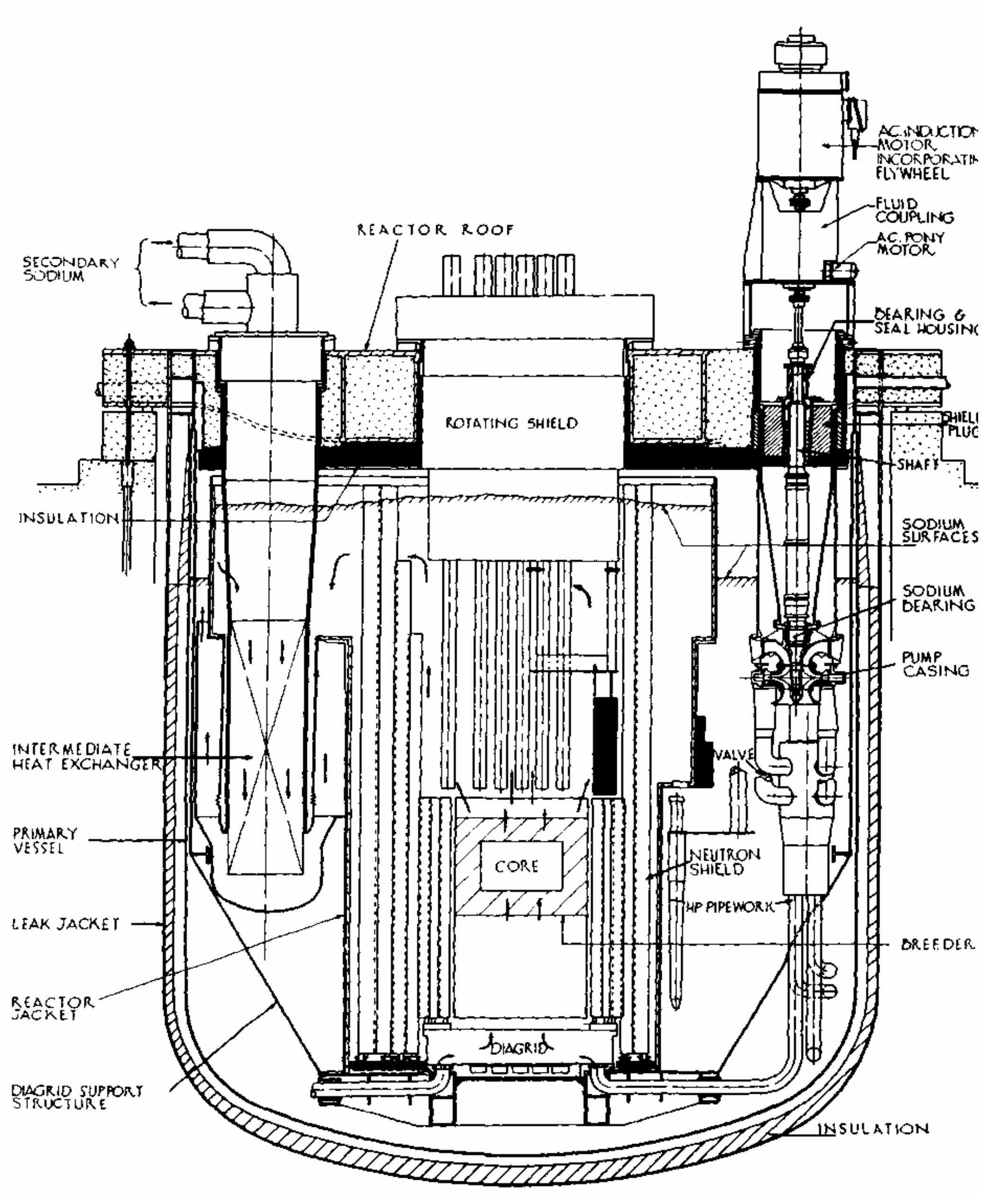

FIG. 1. Elevation through PFR primary circuit.

Secondary sodium flowed through the shell side of each IHX and transported heat to the steam generators consisting of an evaporator, a superheater and a reheater. There were three secondary circuits, each containing about 75 tons of sodium which was circulated by a mechanical pump similar to the primary sodium pumps and each coupling a pair of IHXs.

Superheated steam from the three circuits flowed to a common header to drive a 300 MW turbo-alternator. The main feed was via a 100% duty steam-driven pump with 10% electric and 10% steam-driven pumps for start-up and post-trip use (later, a 50% capacity auxiliary electrically-driven pump was installed as back-up). Appropriate water conditions were provided by a full-flow polishing plant, and the feed-heating by sets of low pressure direct contact and high pressure tube units. The condenser was cooled by seawater.

The reactor core and its surrounding blanket were made up from an array of hexagonal MOX subassemblies, 142 mm across flats. The assemblies were of a size appropriate to a full-scale commercial reactor, and provided a core 910 mm high and about 1550 mm in diameter. Control was exercised through five boron carbide absorber rods, and a further five similar rods were available to shutdown the reactor.

A radial breeder (blanket) surrounded the core and was itself bounded by stainless steel reflector assemblies to improve neutron economy. Outboard of the core and blanket was a graphite shield which essentially eliminated neutron activation of major removable components such as the primary pumps, valves and the IHXs, the secondary sodium, and the primary vessel itself. Special loops filled with eutectic sodium-potassium alloy (liquid at room temperature) as coolant were provided to reject decay heat from the primary coolant via air-cooled heat exchangers to the atmosphere after reactor shutdown if the steam generators were not available.

Fuel could be transferred from an adjacent preparation facility, the irradiated fuel cave (IFC), to a storage rotor within the primary vessel while the reactor was operating. This rotor reduced the time required for refuelling operations and, when irradiated fuel was discharged from the rotor to the IFC, reduced the number and complexity of the transfer flask movements because irradiated fuel removed from the core could be left to cool in the rotor before being moved to the IFC after the reactor had resumed operation.

Transfers between the core and the storage rotor used a vertical lift pantograph charge machine working through a single rotating plug in the reactor roof; for such moves the reactor had to be shut down to allow the charge machine to be installed. Fuel discharged from the rotor after irradiation was first stored under sodium in the IFC until either it had been examined and returned to the reactor for further irradiation or was cool enough to be prepared for reprocessing (including steam cleaning to remove all traces of sodium) and then moved to a buffer store to await transfer to a reprocessing plant, also located on the site. The time from start of construction to filing of the primary circuit had been seven years compared with the four years planned, the delay being due principally to difficulties experienced in the welding of the reactor vessel roof. The nuclear power plant construction cost was about 40 million pounds (1974), with an additional 5 million pounds for the installation of a 170 km long high voltage transmission line to connect with the main grid.

Commissioning had proceed without major problems though a water test of the test of the sodium side one of the circuit had revealed a gas entrainment problem requiring modification of all shut-of-circuits, and bearing problems on one of the primary pumps and two of the secondary pumps. The approach to criticality began in February 1974 and was first achieved on March 1974. Physics parameters for the core and for the reactivity effectiveness of, and interaction between the control and shut-of rods agreed with prediction within the expected uncertainties. The hot dynamic test was completed in June 1974.

The operating history of the PFR power plant can be divided into two phases. For the first ten years (1974-1983), electrical output was limited, mainly because of a series of leaks in the steam generator units, and the highest load factor in any year was 12% in 1978 (Fig. 2). After 1984, with the steam generator weld problems dealt with, plant performance improved, in the final year of operation the load factor was about 57%.

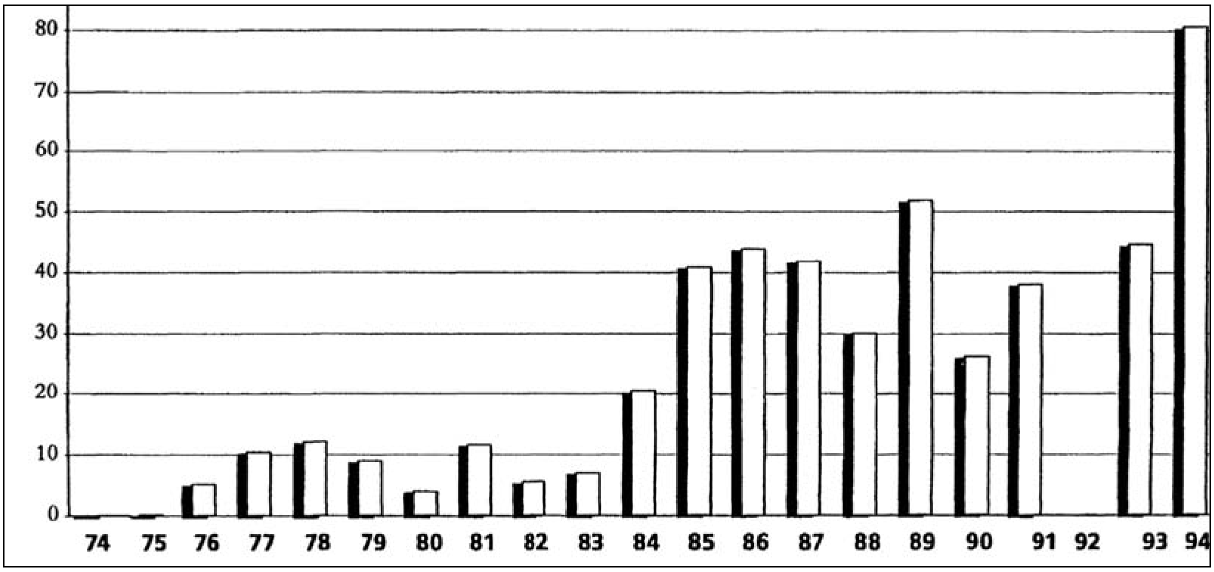

FIG. 2. PFR annual load factors 1974-1994
(1994 for three months' operation only-before decommissioning), %.

In 1985, PFR was able to operate, for the first time since the commissioning period, with a full set of steam generator units. In the second decade of operation there was one major outage, in 1991/92. In this period, unto 1991 the reactor and primary circuit equipment were responsible for only a very small fraction of unplanned outage time; on 25 June 1991, a leakage of oil from a bearing of one of the primary pumps into the primary sodium led interruption of reactor operation for 18 months. PFR was started up for the last time on 14 January 1994. Figure 2 shows a histogram of annual load factors. The major incidents and unforeseen events over the 20 years of operation were as follows:

— Series of steam generator leaks;
— Sodium aerosol deposits in control rods;
— Major oil leak into the primary circuit;
— Cracks in the air heat exchangers of the decay heat rejection loops;
— Sodium mixing and vibrations problems.

2.2. REVIEW OF THE PFR STEAM GENERATOR DESIGN CONCEPT AND OPERATING HISTORY

PFR had three separate forced-circulation steam generators, each of which contains an evaporator, a superheater, and a reheater. Figure 3 shows the general arrangement and Fig. 4 shows a superheater in more detail.

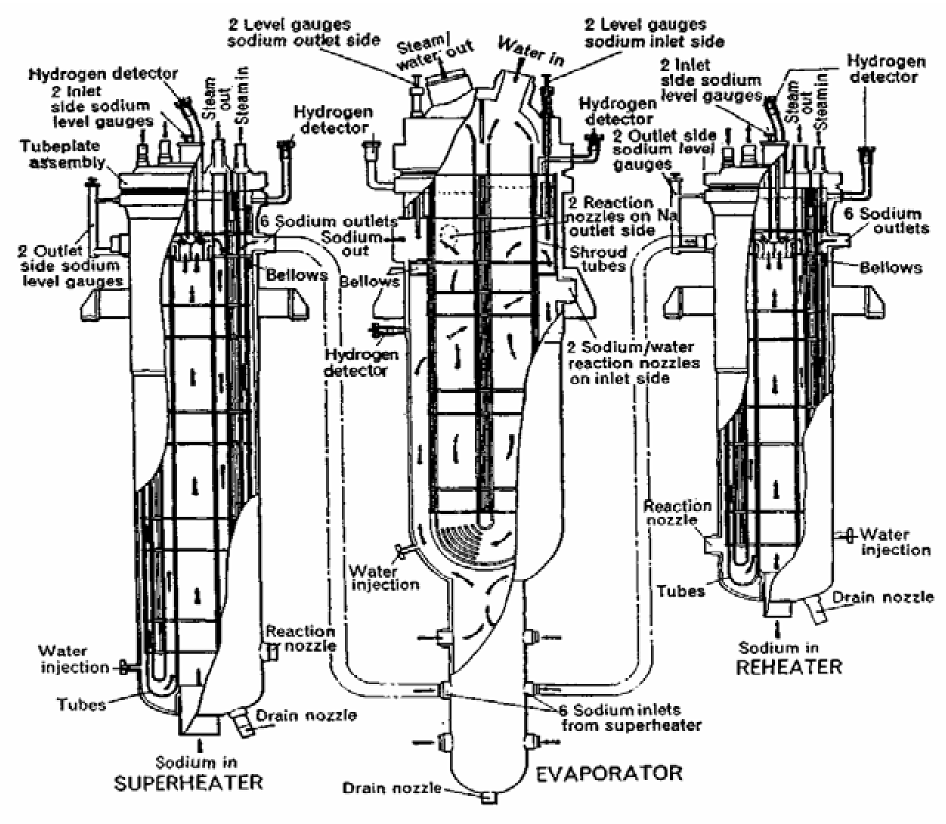

FIG. 3. PFR steam generators.

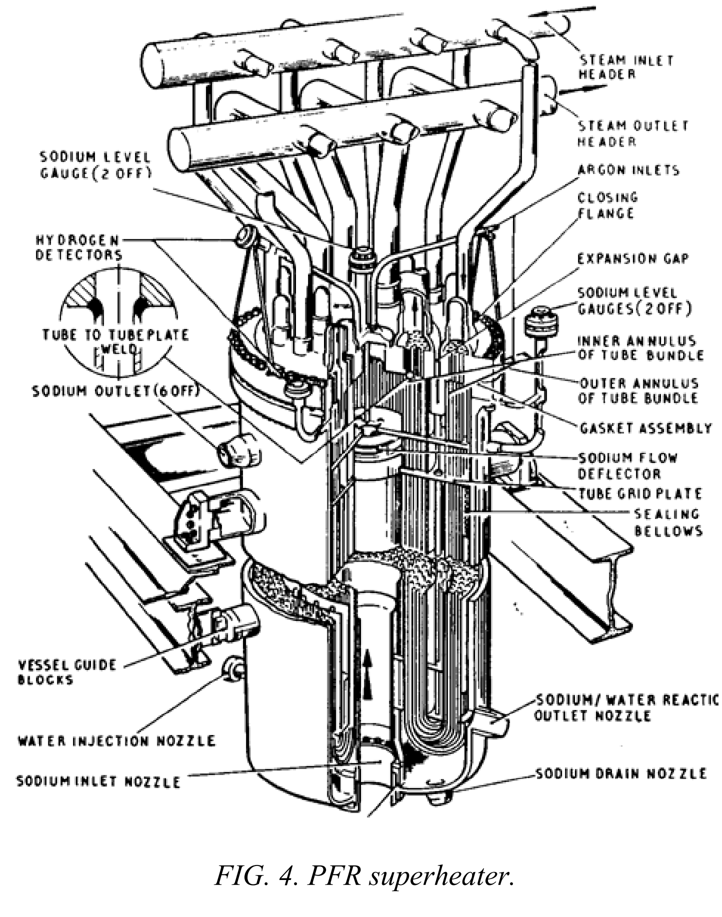

FIG. 4. PFR superheater.

2.2.1. PFR steam generator: choice of the design concept

All three heat exchangers are of “U” tube configuration in which the tubes were welded to tubeplates. There were no other welds in the tubes. All three tube bundles can be removed from their shells by breaking a flanged joint situated at tubeplate level that they are of different configurations. In the evaporator the tubes are arranged so that there is a simple diametrical baffle across the shell to provide counter-flow; the other two units (superheater and reheater) achieved the same objective with annular baffle and annular steam headers.

A free sodium level was established below the tubeplate. The main idea was, that the gas spaces which contain all the tube-to-tubeplate welds, to protect welds by keeping the sodium away from them as well as to improve the sensitivity of detection of the leaks in the tube-to-tubeplate welds. The UK experts believed that limitation of welds to the gas space was essential in PFR, where one of the main objectives was to prove the single – wall concept.

The general philosophy of achieving reliability and high availability as well as leading to simple design required detailed consideration of possible manufacturing and fabrication problems. It was realised at the outset that the tube-to-tube-plate joint would be a critical area.

Three types of the tube-to-tubeplate joint and parameter tolerances set because of material and tube size differences between the units were considered at the design stage (Fig. 5). An explosive welding was rejected because the UK designers considered that it was impossible to remove the sodium side crevice.

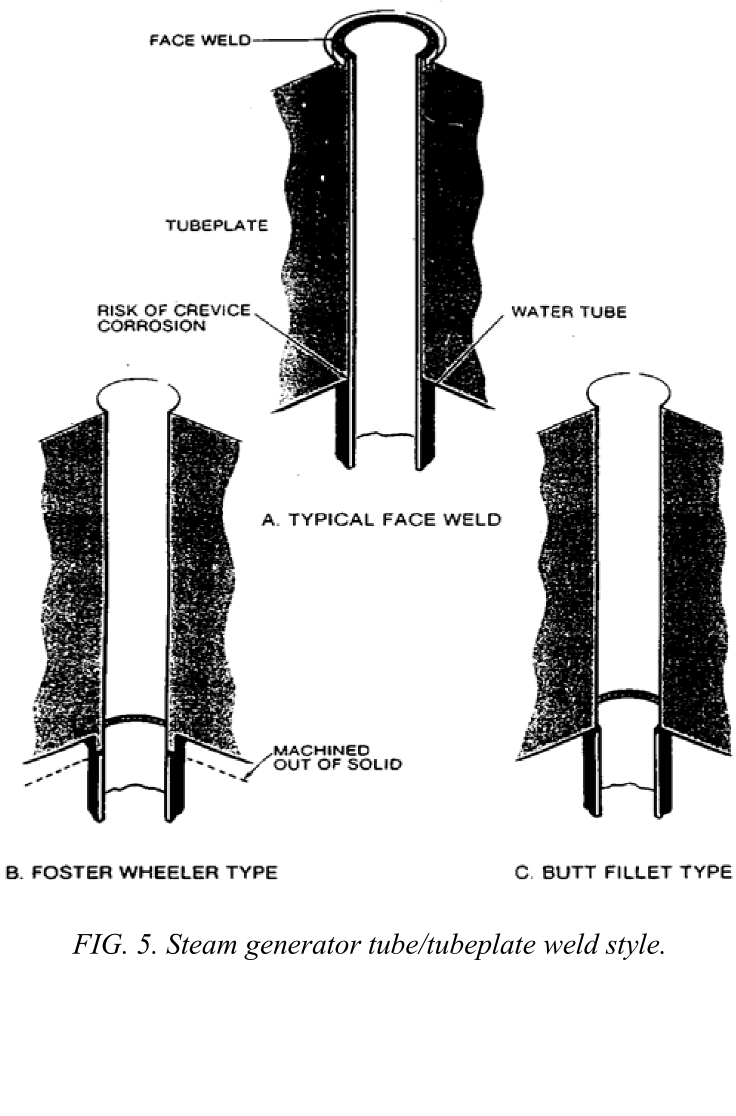

FIG. 5. Steam generator tube/tubeplate weld style.

Therefore two types of tube/tube-plate weld: ‘stub/butt’ weld and ‘butt/fillet’ weld were selected for detail consideration (Fig. 6 a-b). Both welds rely on internal bore welding using tungsten inert-gas welding without the addition of filler wire. Mixed Nb + Ti-stabilized 2 1/4Cr-1Mo tubing was welded.

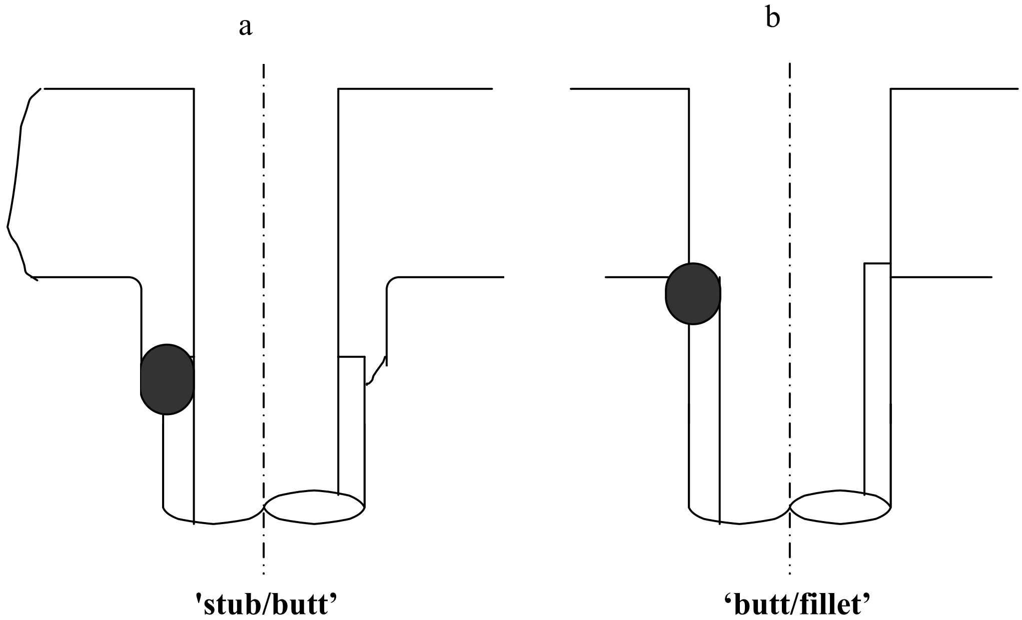

FIG. 6a,b. PFR steam generator tube/plate weld joints.

Some experts believed (and this was confirmed by PFR SGs operation) that 'stub/butt' weld (Fig. 6a) provides a favourable mass balance for welding, the post-weld heat treated after manufacture can be readily provided and the joint gives maximum ligament efficiency. Initially single welds were made using a 200°C preheat and a post-weld heat treatment of 30 min at 700°C, but subsequent trials showed that welds could be made with satisfactory profile and quality without any preheat.

Further tests were done on test blocks to simulate the conditions in service. No preheating was used and radiographically acceptable and gas-tight welds were produced. There was a tendency to produce porous welds, and in the main these could be satisfactorily repaired by re-running. Specimens cut from these welds showed that the ultimate tensile strength was 32.2 tons/in^2 at room temperature, but this was reduced to 26.2 tons/in^2 at 480°C. Other mechanical tests e.g. bend and fatigue tests gave satisfactory results.

In parallel with the development work on the ‘stub/butt’ weld, work was undertaken to investigate an alternative geometry. It was found that the machining costs for the ‘stub/fillet’ weld type of joint were considerably cheaper than for the ‘stub/butt' and thus additionally there was no possible weakness due to the cross-grain of the machined stub. On the other hand, this joint has the disadvantage of relatively poor ligament efficiency compared with the ‘stub/butt’ joint and the joint geometry was not ideal for either optimum coolant flow conditions or for subsequent non-destructive examination

The ‘butt/fillet’ weld was used for the design (Fig. 6b). The welding conditions were closely controlled with particular attention being paid to fit of components, location of electrode, arc length and rate of heat input. Tolerances and fit are rigidly controlled by inspection whilst the electrode location can be preset. Welding parameters were controlled automatically within the limits required to produce consistent welds. A similar type of welding control has been applied to fuel-element assemblies where the accuracy of control was such that service failures have been restricted to 1 in 30 000.

The design of each evaporator unit required the welding of 498 U-tubes 39-ft -in. long into a 7-ft 4-in. diameter tube plate. Furnace stress relief was not considered practical due to the difference in section and the handling problems of a U-tube bundle with such a thick header. Furthermore, a suitable local stress-relieving treatment was not considered possible and tests were initiated to assess the performance of the unstressed relieved welds.

One of the main disadvantages with the ‘butt /fillet' weld compared with the ‘stub/butt' weld was the difficulty in carrying out an adequate non-destructive test. A high degree of quality control and reliability was maintained due to the automatic control of the important welding variables. Additionally, with this type of joint the weld profile is a good indication of weld quality and care was taken to ensure that the welding parameters were controlled within acceptable limits. Special optical devices have been developed for visual examination of both weld surfaces.

A permanent record could be kept of the bore examination by means of a TV camera and a video recorder. The camera and endoscope were mounted in a housing which locates the endoscope at the centre of the weld and allows focusing and both axial and rotational movement. The lighting, focusing and drive-direction were controlled remotely by means of a miniature transistorized console. The picture was displayed on an 11-inch monitor and stored on magnetic tape. A sound channel was used to add a commentary to identify the weld and to draw attention to any anomalies. The play-back is of good quality and cracks are clearly defined.

Despite this high degree of quality control, development work aimed at finding a reliable non-destructive testing technique to a successful conclusion. Two systems were possible: one an ultrasonic technique using twin finely tuned shear-wave probes and the other radio graphic using a small thulium-170 isotope source. The success of radiographic technique depends on the production of small 1-mm×1-mm right cylinders of thulium-170 sources, at the initial source strength of 3 1/2 Cr and the development of a film cassette capable of being wrapped.

Using this new improved technique crack indications were observed in the tail-off region of the weld. Electro-polishing of the outer weld profiles and careful sectioning showed that these defects always occurred on the outside surface of the weld and were typically 0.010 in. long and 0.002 in. deep. Initially it was thought that these cracks were due to the presence of small quantities of Nb-rich eutectic giving low hot ductility on the rerun during the tail-off of the weld, but subsequent tests showed that significantly greater quantities of eutectic than that present in the weld did not affect ductility. Nevertheless, modification of the startup and tail-off techniques eliminated the tendency to produce these fine cracks. Experience in the manufacture of 2 000 defect-free welds confirmed the ability to satisfactorily weld the stabilized steel in these geometries.

Thulium radiography was used for general weld soundness, visual examination ensured the correct weld profile and a check was maintained on a percentage of the welds and by destructive examination of production-control welds to ensure that the modified tail-off technique has eliminated the micro-cracks referred to above.

2.2.2. Review of steam generators operating history

First criticality of the PFR reactor was achieved on 3 March 1974. In October 1974, during early steam commissioning, a leak was detected in superheater 3.

Up to 1976 there were failures in gas-space leaks in one evaporator, two superheaters and one reheater. These early failures were believed to have been due to manufacturing faults.

A total of 37 gas-space leaks were experienced in PFR SG units in the period 1974 to 1984 with 33 of these occurring in evaporators, 3 in superheaters and 1 in a reheater. All the gas-space leaks originated at the welds between the tubes and the tubeplates.

In the case of the austenitic superheaters and reheaters, the leaks gave rise to considerable concerns about the design. Although both the damaged superheaters continued in use up to 1986, having had the leaking tubes plugged, one of the superheaters and the reheater had suffered from caustic stress corrosion cracking of the tube plate caused by the products of the sodium-water reaction. The superheater was salvaged by grinding out the cracks and thoroughly washing the tube plate with hot sodium to remove the reaction products.

Damage to the reheater tube plate was so extensive that the tube bundle was scrapped. It was replaced by a plug in the empty reheater vessel until a replacement was fitted in 1984. For this period the plant had to be operated with reduced reheat capacity. Following the early failures of tube-to-tubeplate welds in the two superheaters and the reheater no further failures occurred in the austenitic units until 1986, when a superheater tube leaked while the unit was being pressurised with steam prior to being put on line. This incident is described below.

In the period 1984–1987 all the six austenitic tube bundles were replaced by new tube bundles (Fig. 7a). The design benefited from the early experience of caustic stress corrosion following the leaks in the austenitic units.

This unfavourable experience, so early in the operating life of the steam generating plant, led to a decision to order a complete replacement set of superheater and reheater tube bundles of modified design to be fabricated in 9Cr-lMo ferritic steel. As in the original units, the new units were to have no under-sodium welds in the steam tubes, but, in addition, the need for difficult tube-to-tubeplate welds, and, indeed, for any welds separating the steam and sodium environments, was eliminated by passing the steam tube through a sleeve projecting above the tubeplate, removing the joint to the steam header from the sodium environment.

Each sleeve was brazed to the steam tube and welded to a seal plate (Fig. 7b). The new design also incorporated improvements to reduce flow induced vibration of the tube bundles. These units were delivered to site in 1984, at which time it became possible to replace the missing reheated and thereby restore the full complement of steam generator units. The replacement work was completed by 1987.

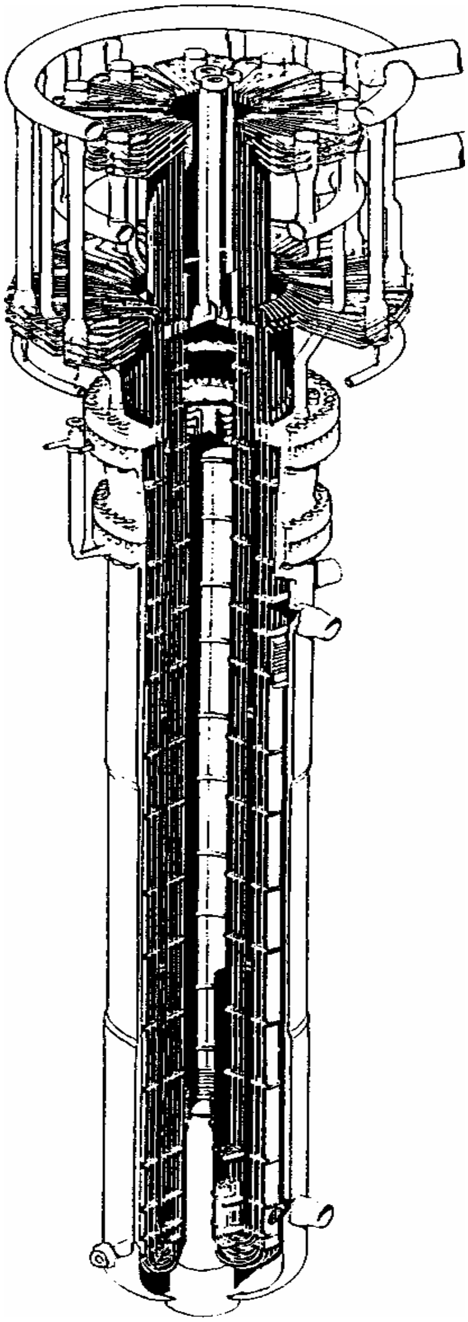

Fig. 7a

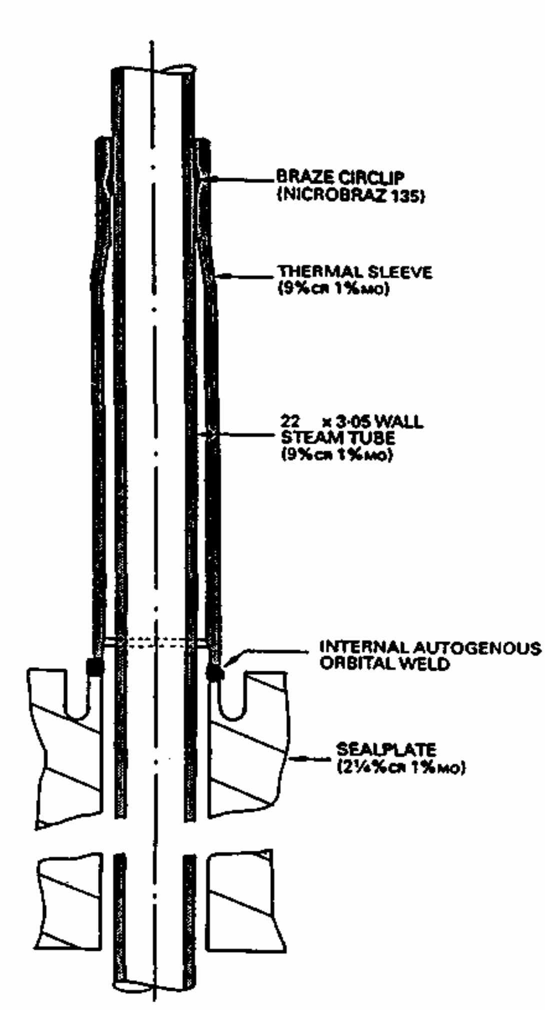

Fig. 7b

FIG. 7a,b. PFR replacement reheater tube bundle, reheater thermal sleeve.

The 33 leaks experienced in the ferritic steel evaporator units were relatively benign as ferritic steel is not so subject to caustic stress corrosion, but the effect on availability while leaking tubes were being repaired was considerable. The evaporator gas space leaks were all associated with cracking of the tube-to-tubeplate welds. These were hard and had high residual stresses because there was no post-weld heat treatment. None of the evaporator leaks gave evidence of wastage damage to the neighbouring tubes, probably because they were detected early by the installed gas-space hydrogen detection system. This was based on katharometers and was very sensitive, being capable of detecting leaks as small as 0.1 mg/s. The leaks were repaired by plugging the affected steam tubes. Nevertheless it appeared that one leak would, after a few days or weeks of further operation, cause others. It was concluded that residual caustic reaction products in the gas space above the sodium caused further cracking of welds and initiated more leaks after an incubation period. Sodium flooding of the tubeplate at a temperature in excess of 400°C for periods in excess of 24 hours had some success in removing reaction products. It was also required to wash out sodium hydrides which could lead to false hydrogen detection signals when they dissociated at high operating temperatures. This did not cure the problem completely, however, and eventually it was concluded that washing with hot sodium did not remove caustic material from the roots of pre-existing fine cracks in the welds, so that in the presence of the residual stresses corrosion continued and the cracks grew to give rise to further leaks.

With leaks having occurred in all three evaporators and a sodium-side origin of cracking in evidence, it was decided that the most practicable solution was to by-pass the suspect tube-to-tubeplate welds by using a sleeve. Each sleeve was a precisely machined cylindrical tube of a 9Cr-lMo steel inserted into the tubeplate and extending downwards into the top 75 mm of the tube. The upper end of the sleeve was explosively welded to the tubeplate and the lower end was braised into the inside of the tube (Figs 8 and 9).

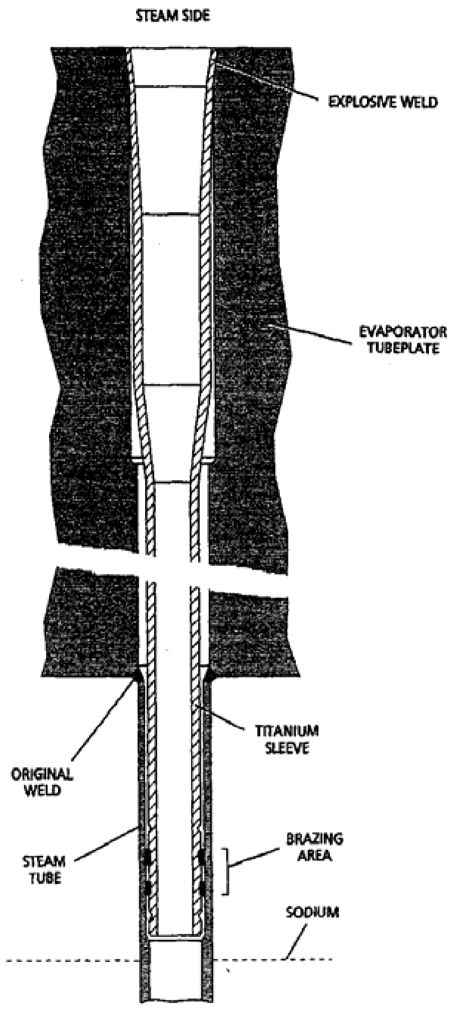

FIG. 8.

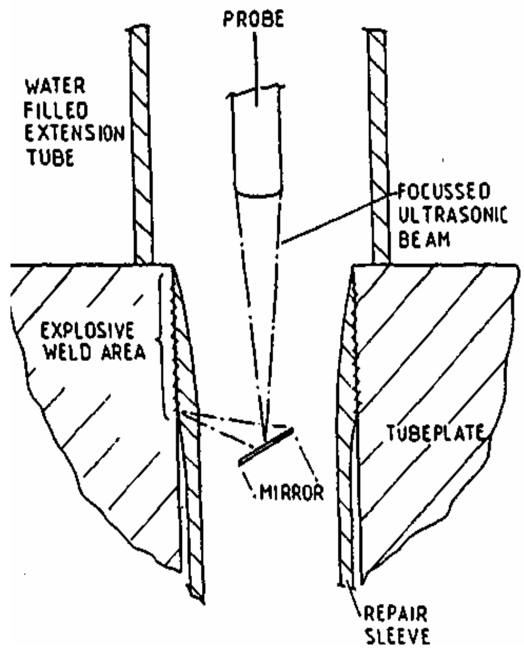

FIG. 9.

FIGs 8 and 9. PFR evaporator weld repair sleeve and explosive weld examination.

Following extensive laboratory trials, four experimental sleeves were first fitted to operational evaporators towards the end of 1980. A further 11 evaporator weld leaks occurred in 1981 and a further 41 sleeves were fitted to by-pass defective and suspect welds. In parallel, work was in progress to examine whether the sleeving technique could be applied on a routine basis. A trial installation of 200 sleeves was conducted on the spare evaporator tube bundle in the latter part of 1982. A decision was then made to sleeve all tube-to-tubeplate junctions in the three evaporators.

The problem was finally solved by the fitting of sleeves which spanned the original welds, as shown in Fig. 8. In all 3 000 sleeves were fitted over a 14 month period. The work was completed early in 1984 after a 14 month campaign. While sleeving was underway the station operated on a single circuit. Following sleeving no further problems were experienced with the evaporators.

The type of direct tube-to-tubeplate weld (the 'butt/fillet' weld) adopted initially at PFR, which could not be heat treated after manufacture, should be avoided in future fast reactors. Austenitic steels are in question for LMFR steam generators because of the high risk of caustic stress corrosion damage following even small leaks.

2.2.3. The under-sodium leak in PFR superheater

An under-sodium leak occurred in PFR superheater 2, one of the original units made from austenitic steel (Fig. 3), in February 1987. It provided valuable information on the behaviour of sodium water reactions in an operating steam generator and led to a complete reassessment of the design basis steam generator accident for subsequent fast reactors.

On 27 February 1987 PFR was being operated at full power when a sodium-water reaction trip was caused by the rupture of a bursting disc on the stem side of superheater 2. This initiated a dump of the steam and sodium in the secondary circuit and automatic shutdown of the plant. Shutdown to a safe state took approximately 10 s, as designed. It was confirmed shortly after the incident that a large under-sodium leak had occurred in superheater 2.

Figure 3 shows one of the original PFR superheaters. After the sodium circuit had been cleaned to remove reaction products, the superheater tube bundle was removed from its vessel in a nitrogen-filled bag and examined. This revealed that between two tubes support grids one of the six baffle plates forming the central sodium inlet duct had become detached, and the remaining 5 plates in this region were deformed. Considerable distortion of steam tubes could be seen through the aperture left by the missing baffle plate.

An incident at the PFR steam superheater, however, made to treat more attentively tubes behaviour analysis in the large water-into-sodium leak zone in a steam generator within a time interval following the shock load effect. As a result of a leak occurrence in one tube in the process of the accident 40 tubes had a through damage — a guillotine type rupture. According to the data from material research, the temperature in the reaction zone increased up to 1 300°C. In addition to 40 tubes having through damages, about 500 tubes were high temperature (in excess of 800°C) reaction products, on 70 tubes swelling was revealed.

As a result of the analysis of the whole set of data the British specialists presented the following version of the accident development. The initial leak occurred on the tube no. 16 (Figs 10 and 11).

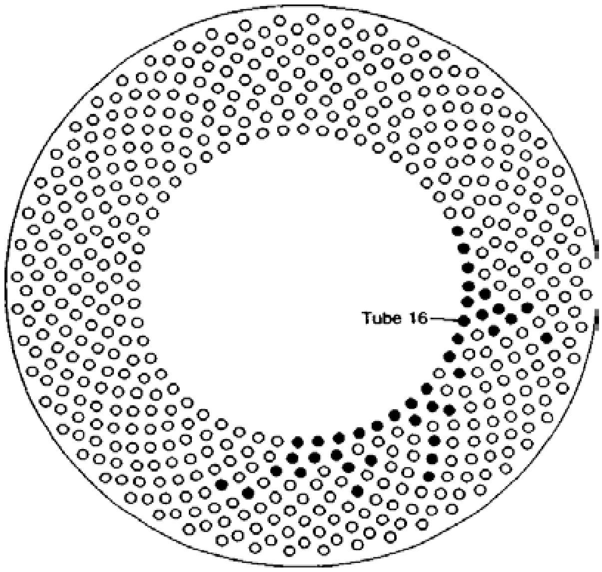

FIG. 10. The failed tubes in PFR superheater 2.

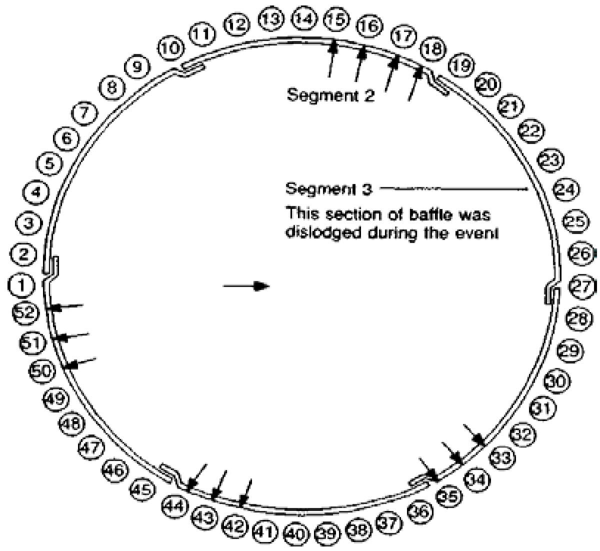

FIG. 11. The location of fretting marks on the PRR superheater 2 central baffle.

The cause was a crack due to wall wear by about 70% of its thickness. The leak was not detected because of a lack of hydrogen-in-sodium monitoring, and the system of hydrogen monitoring in the gas space proved to be too delayed in time (it operated after a rupture of the bursting disc). Such an initial leak could continue for several hours at a flow rate of 0.001 g/s and a few minutes at a flow rate of 1.0 g/s. Then the leak passed into an intermediate one and during about 20 s was at a level of 0.5 kg/s. The consequence of this leak was a guillotine type rupture of a few adjacent tubes weakened by abrasion wear due to vibration and by their being in the high temperature zone. The cause of the rupture was a high steam pressure (13 MPa).

The rest of the tubes were damaged in the process of steam generator shut down when after the cessation of water and steam flow rate the heat removal from the reaction zone was stopped. The process was aggravated by an inadequate design of the sodium and reaction products removal system. Altogether, about 150 kg of water penetrated into sodium. Thus it has been found that at an initial large leak, under some conditions the secondary failure of heat exchange tubes is possible because of metal overheating.

Experiments carried out by USA specialists have confirmed the above fact. Thus, one of the tests was carried out at an initial leak of 454 g/s. The first of the secondary leaks appeared on the target tube in 16 s and was caused both by corrosion and by a rupture under the effect of internal pressure. The traces of corrosion wastage were found on one more tube. All the other failures (a total of 23 tubes were failed) were of an “overheating-rupture” character. It was recognized that most of the secondary failures were the result of the reaction between steam leaking from the failed tubes and residual sodium after its draining. An important result of the experiments is also that despite a considerable number of damaged tubes the acoustic waves of pressure were determined by the initial large leak.

On the basis of a generalized analysis of the results of the experimental validation studies carried out on the EFR design, it has been found that the effect of a tube wall failure by the “overheating rupture” mechanism is possible at an appearance of the secondary (or of the primary) leak with a flow rate more than 80 g/s. And in this case the time from the appearance of this large leak up to a failure of adjacent tubes depends on the leak size. It was shown that with an increase of a leak from 100 to 4 000 g/s the above time interval reduces from 50 to 4 s. Thus, when developing an approach to the choice of a design basis leak value, the following should be taken into account:

(1) The rate of hydrodynamic processes in the sodium circuit (and, therefore, the circuit strength) is determined by the rate of an increase of an initial large leak. It is expedient, therefore, as an initial leak to take that appearing at an instantaneous guillotine failure of at least one tube.

(2) Steam generator tube bundle damage zone (the number of heat exchange tubes), as well as the amount of water penetrating into the sodium circuit (and, therefore, the scope of repair work) depend on the SG design features and on its safety system characteristics. It is desirable to aim at the most rapid possible termination of the sodium-water interaction reaction after the appearance of the secondary (or the primary) large leak.

Flow induced vibrations were at the origin of the most important under-sodium leak at PFR. Analysis of the event and examination of the superheater concerned has shown that flow induced vibration of a single tube resulted in the initial leak, which thereafter developed rapidly. All the original reheater and superheater tube bundles were replaced with new and much improved units.

The under-sodium leaks demonstrated that it is possible for a large number of tubes to fail due to overheating in a period of a few seconds. The incident led to a reassessment of the design basis accident for the steam generators of both PFR and EFR. In the case of PFR the design basis-accident was changed from a single double-ended guillotine fracture to 40 double-ended guillotine fractures spread over a period of 10 s.

2.3. EFFECTS OF SODIUM AEROSOL DEPOSITION IN LMFRs

Most of fast reactors have to some extent experienced problems related to sodium aerosol depositions. At BN-350 and BN-600 some deposits were observed in the gap between the large rotating plug and the reactor vessel roof, inducing difficulties of rotation. At KNK-II and Phénix, deposits have been found in the control rods between the shielding piston and the guide tube, causing difficulties with their insertion.

At PFR, the possibility of sodium aerosol deposition in those parts of the absorber drive exposed to the primary vessel gas blanket was recognized in the design and an argon purge flow was provided in these areas to prevent aerosol transport. However, it was not completely effective and the removal of small deposits on the magnet faces in the drive mechanisms was a routine procedure at PFR. This led to plant trips on a number of occasions due to absorbers dropping off their magnets during power operation.

The question of difficulties with the PFR control rods has been examined in two phases. During the first phase (up to late 1985) the results gave no cause for concern. Since 1985, the reactor operated at high power and high temperatures and greater problems were encountered.

It was believed at the beginning that sodium deposition was a main cause and after investigations confirmed this. Particular features to note are the roof insulation region, the weight sensing equipment and the dashpot. Initially, it was thought that the main issue concerning the freedom of rod movement would be bowing caused by neutron induced voidage (NIV) coupled with thermal effects. The computer code Peeble predicted maximum friction of 25 kg_f. However actual measured maximum friction was 40 kg_f and the distribution of measured friction with rod position was not in good agreement with the Peeble prediction.

PFR had 5 shut-off rods (normally fully raised) and 5 control rods inserted to control power. The rods were essentially identical B_4C assemblies supported by electromagnets (Fig. 12).

On a trip all ten rods dropped. Magnet current, apparent rod weight, rod release time and time of flight were measured by installed instrumentation

At all shutdowns and after plant trips the electromagnet pick-up and drop-off currents were measured. These were the minimum magnet currents at which the absorber could be raised and at which it dropped off after being raised. On the basis of these figures a decision was made on whether the magnet faces had to be cleaned before return to power. If required the drive and magnet assembly were removed by simple bagging techniques and the magnet face was cleaned in an argon purged glove box. The extension rod face was cleaned in situ using commercial “Scotchbrite” cleaning pads, again making use of a simple bagging technique.

After late 1985, when the high power operation started a sudden increase of friction was measured with values going up to 80 kg_f. At this time the possibility of sodium aerosols in the cover gas and deposition on the keys and key ways in the upper part of the mechanism were discussed. As an explanation it is suggested that sodium has deposited in the positions indicated in Fig. 13.

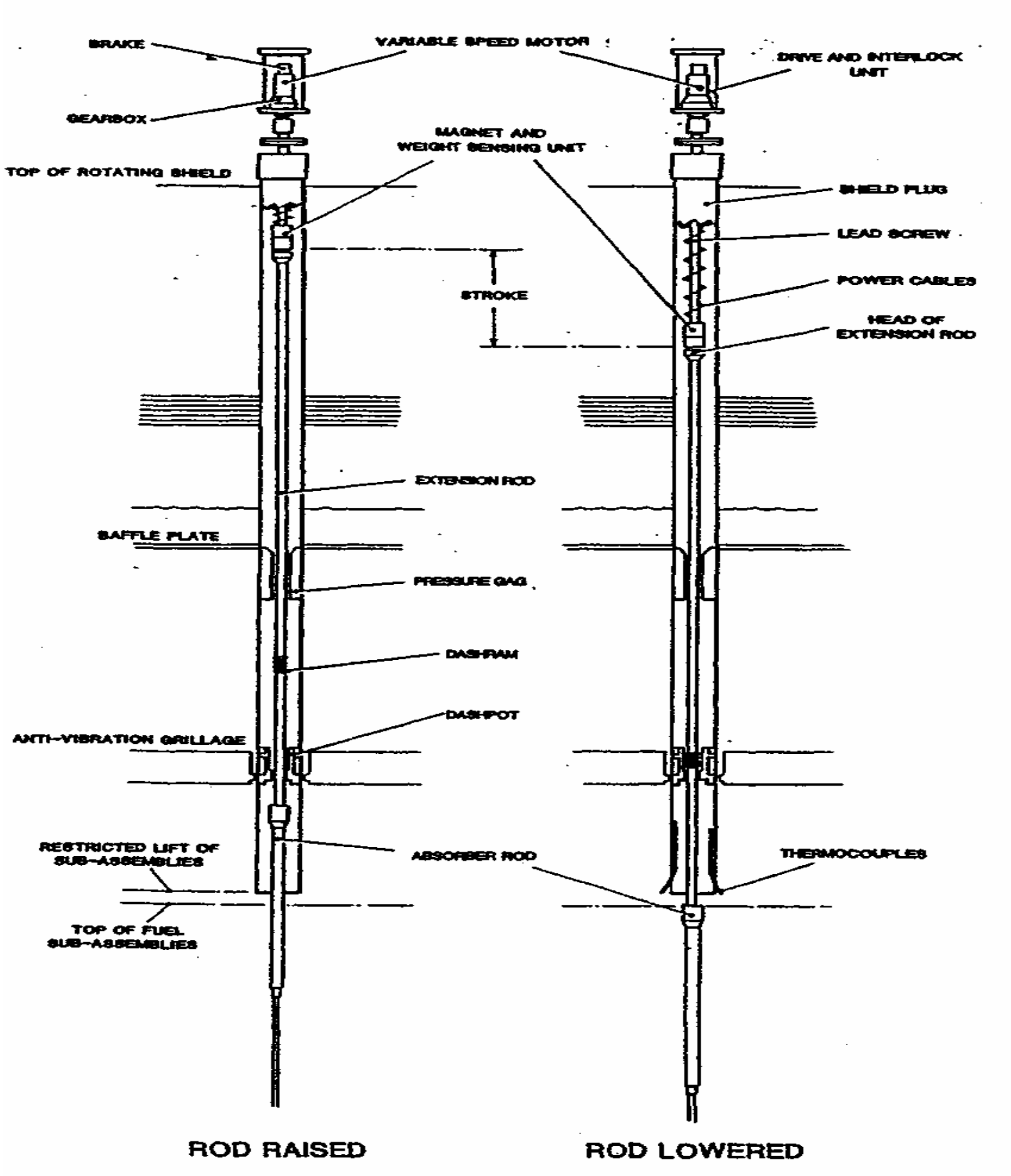

FIG. 12. The PFR absorber rods.

Deposits in these positions could give the distributions of friction measured, largely as a result of the detailed design of the keys which are attached to the "Latch/Delatch" tube and which engage in key ways in the extension rod. In 1988 a special glove box was made which allowed examination of the liner tubes and the extension rods. Examination of a number of rods confirmed that sodium deposits were present but in smaller quantities than expected and confined to the keyway of the extension rods. None were found on the liner tube as originally hypothesised. The sodium was soft and easily removed. Although the absence of deposits other than in the keyways was surprising, when they were removed the friction of the restored rods to normal. It took some 40 effective full power days (efpd) of operation for friction levels to begin to rise noticeably.

A difficult problem in operation was posed by sticking in the shutdown systems of KNK-II (Fig. 14).

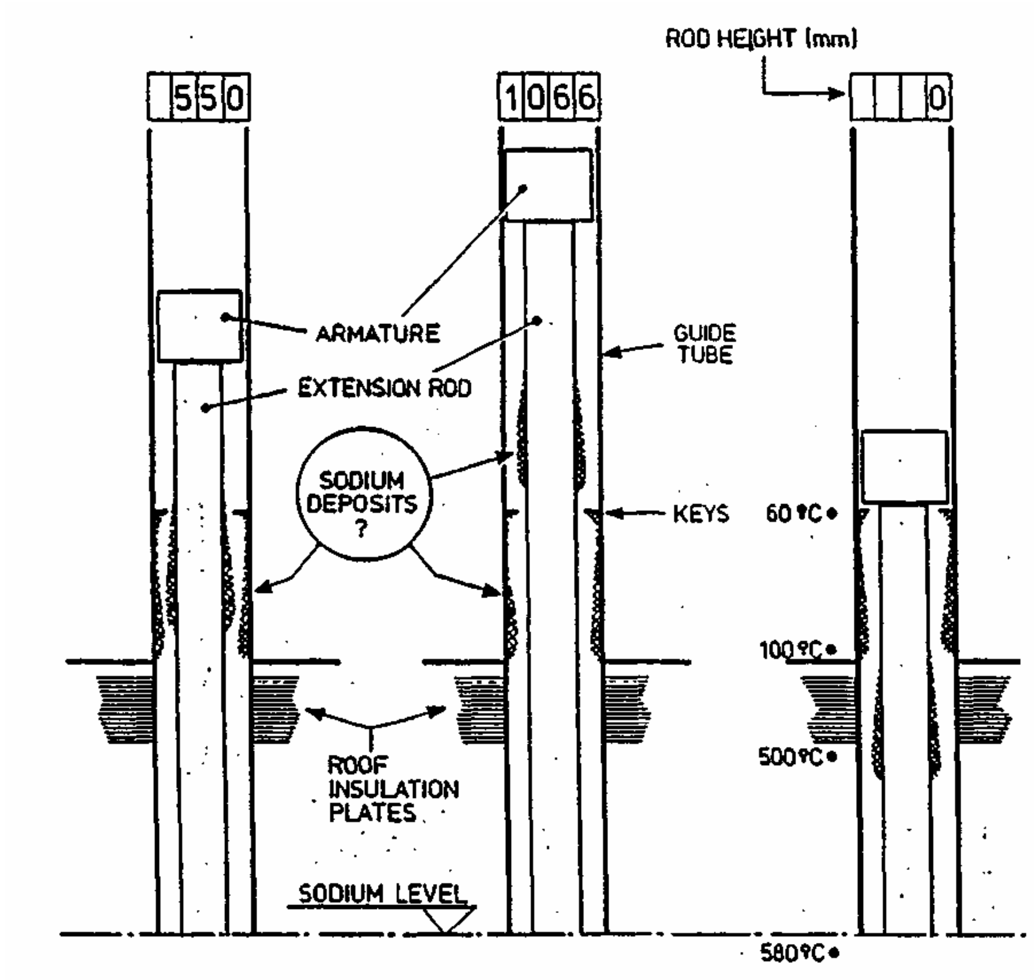

FIG. 13. Location of sodium aerosol deposits on a PFR absorber rod mechanism.

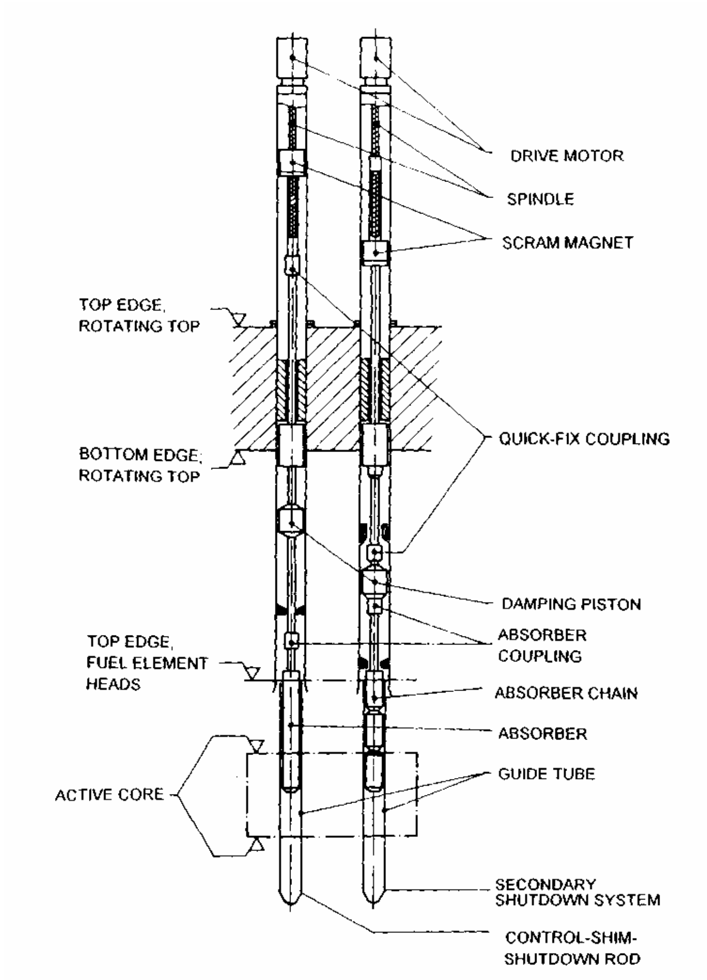

FIG. 14. Schematic representation of the shutdown rods in the KNK-II shutdown systems.

In December 1986, a control rod of the primary shutdown unit for the first time was found to stick while the reactor was shutdown. The cause was found to be sodium aerosols plated out during prior handling steps, when the rod actuating equipment had not been swept with gas.

In December 1988, deposits were found on a rod of the secondary shutdown system; they impaired the mobility of the component, but not the shutdown function. Probably the fact that the primary system had been opened for maintenance purposes a number of times before had caused the quality of the cover gas to deteriorate and thus produced the deposits. On January 1991 the scram at 15% power reactor operation was initiated by a sudden absorber movement after an obstruction in movement had been overcome. This blockage in the primary shutdown system again was caused by depositions in the rod actuating equipment in a phase in which the cover gas quality had been insufficient.

When the quality of the cover gas is insufficient, the sodium in the rod actuating equipment was oxidized to sodium oxide whose dough-like consistency impeded lifting movements of the equipment. This blockage in the primary shutdown system was caused by depositions in the rod actuating equipment in a phase in which the cover gas quality had been insufficient.

Movable parts within the primary reactor envelop, might be exposed to cover gas carrying considerable amounts of sodium aerosols, have to be protected by structural elements like expansion bellows or double seals with cover gas in between.

If narrow gaps between fixed and moving parts within the reactor vessel gas plenum are unavoidable and venting gas has to be applied, measures for the flow and quality surveillance of such venting gas are essential. In case of poor gas quality, caused by impurities like hydrogen, oxygen or methane, a possibility to switch over to clean gas from the liquid gas storage tank has to be installed. Depending on certain circumstances the gap has to be chosen such as to minimize possible convection of sodium aerosols.

Solving the problem of aerosol deposition is a difficult one because it is related to gas convection and to the temperature of the different structures. In the frame of the new project, an important effort is being made for the development of a high performance calculation code. If needed, design provisions should be incorporated such as tight annular spaces for the rotating plugs, large clearances between moving parts, heating devices to avoid solidification. Also, gas injections should be limited to a minimum in order to avoid oxide formation and deposits on moving parts. The experience with short-circuit devices for the detection of sodium leaks is good. Nevertheless, an important effort is still necessary to improve their sensitivity and reliability.

One solution is to use metallic (ferritic) reactor roof in which the component penetration are machined, the tolerances are more controlled, so smaller clearances are possible. It gives a further benefit of lower a sodium aerosol and heat transfer from the cover gas.

2.4. THE PRIMARY CIRCUIT OIL SPILL

The PFR primary sodium circulation system is shown in Figs 1 and 15. Figure 16 shows details of a primary sodium pump (PSP). The sodium from each PSP flows through filters and a stop valve to the diagrid, and thence to the fuel subassemblies. Each subassembly has a filter at its inlet. Figure 15 shows the relationship between the pump and subassembly filters. In 1974 primary sodium pump 2 (PSP 2) was removed from the reactor for modifications to its instrumentation and was noticed to be heavily contaminated by a black sooty deposit. In the same year the charge machine was removed, revealing that its immersed surface was black with adherent tarry lumps. During this period of operation some 65 L of oil had been lost from the pump upper seal oil systems, part of which is believed to have entered the reactor vessel. When the reactor was taken critical no effects of the oil were observed. It is suspected, however, that as a result of the spill the filter on PSP 2 valve failed due to high differential pressure because it became blocked by oil-sodium reaction products ("O" in Fig. 15). It is also thought that partial blockage of the pump casing overflow pipe was to lead to the major problem in 1991 (Fig. 16).

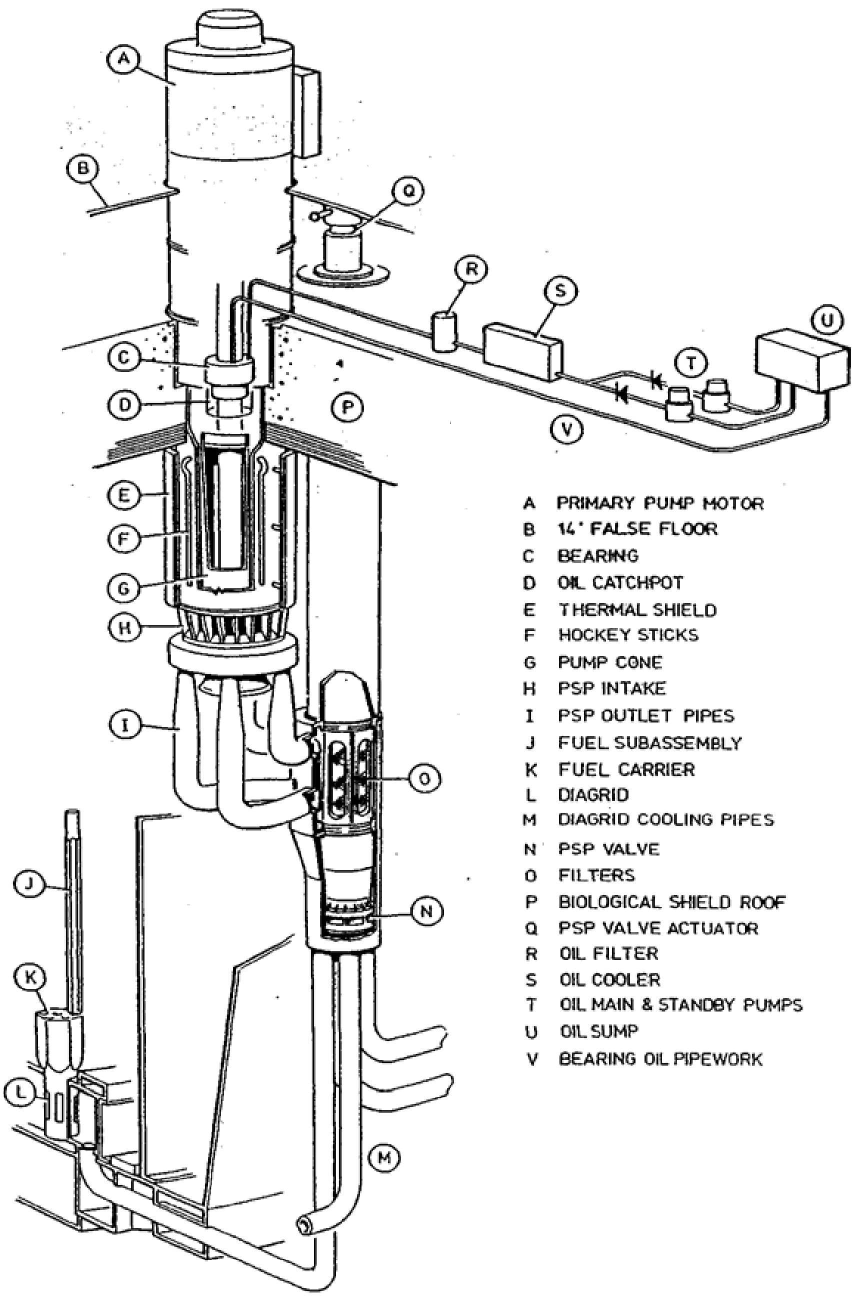

FIG. 15. PFR primary sodium pump and its associated valve and filter assembly.

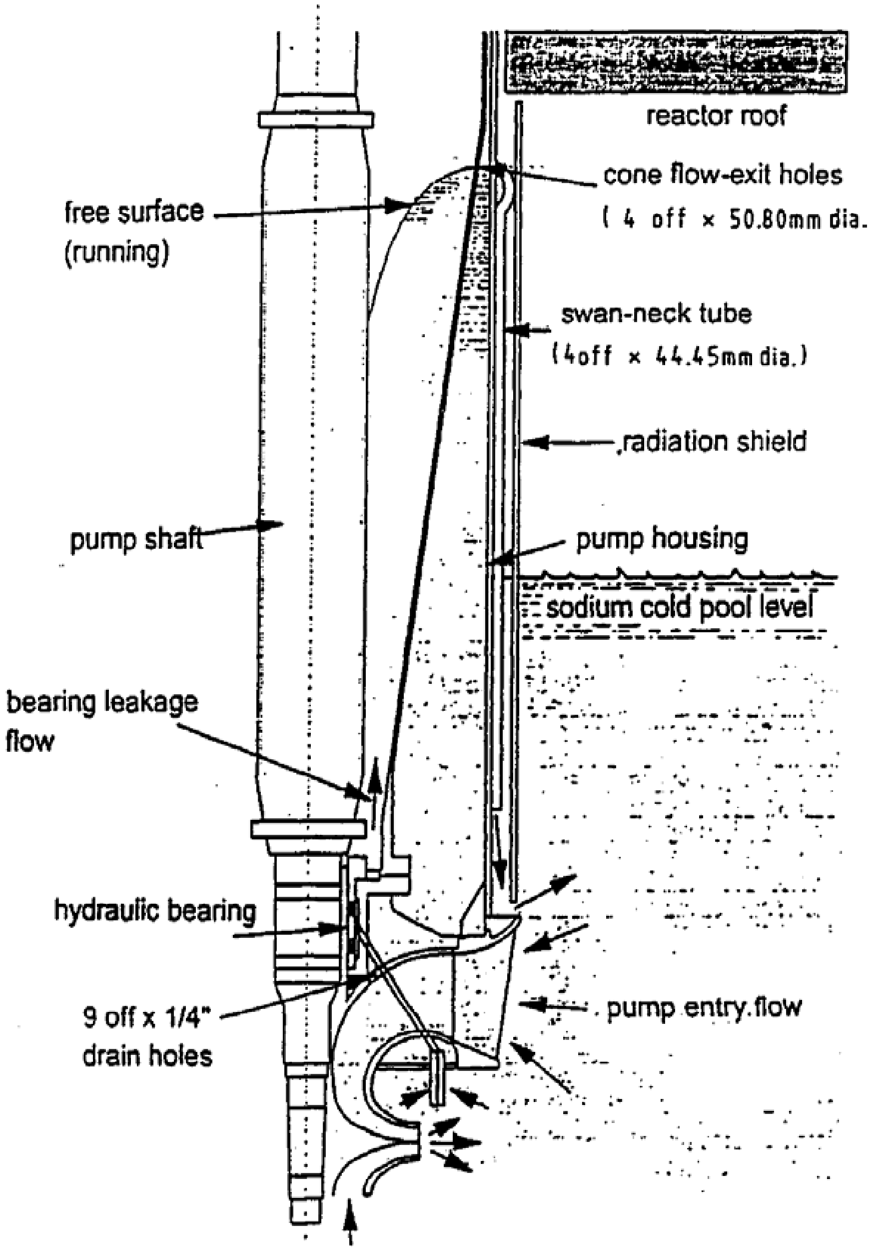

FIG. 16. Detail of a PFR primary sodium pump housing.

At the end of 1991, PFR had been shut down since 29 June when the reactor had been manually tripped following observation of overheating of the top bearing of PSP 2. An extensive examination of the circumstances which led to the leakage of primary pump bearing oil into the primary sodium circuit, has concluded that it involved two stages, the first on 25 June 1991 and the second on 29 June 1991. Prior to these events:

— There was a maximum indicated level of 17 L of oil in the pump drains tank above a in layer of sodium/oil sludge;
— On 24 June 1991 the main argon gas blanket cover gas flow to the reactor dropped to zero due to blockage or partial blockage of the two absolute filters and/or the drain line associated with the aerosol filter;
— On 25 June 1991 efforts were made by the operators to restore the cover gas flow by venting the system through the gaseous effluent system.

Venting the reactor cover gas system to the effluent system reduced the cover gas pressure in the pump support vessel, allowing sodium to rise up the annulus between the pump drive shaft and the drains tanks inner wall and enter the drains tank. This sodium would displace oil to a level above the upper lip of the drains tank. Restoration of the cover gas blanket pressure on 25 June then forced the 17 L of oil down the annulus between the pump shaft and the drains tank into the pump support vessel sodium. Over the next few days, a further 5.5 L of oil which were added to the seal oil sump could have reached the sodium in the pump vessel. A sudden loss of 12.7 L of oil from the pump lubrication system sump occurred on 29 June. This activated the loss-of-sump oil level alarms in the control room and led to the observed overheating of the top bearing which initiated the decision to manually trip the reactor. Examination showed no oil external to the system leading to an estimated total of 35.2 L of oil ingress into the primary sodium.

The remainder of 1991 had been spent in cleaning the primary sodium and in making preparations for examination of the three primary pump filters and the inlet filters on some of the fuel assemblies which had shown temperature increases at the time of the spillage. Removal of the pump filters was a major task which had never been attempted before.

All three primary pump filters were removed for examination. Deposits incorporating some carbon were found on all three. Some difficulty was experienced in refitting them because of distortion of some thermocouple guide tubes and it was necessary to cut these away. The success of the overall operation showed that the opacity (and, as it was the primary circuit, the activity) of the sodium coolant is not an insurmountable impediment to maintenance work. Examination of a number of fuel assemblies also showed deposits on the inlet filters of those which had shown temperature increases prior to shutdown but none on assemblies which had shown no temperature increases. However, the deposits were slight (about 0.5 g) and mainly of sodium. It was improbable that these were sufficient (even allowing for the possibility of some loss during handling) to cause the observed temperature rises and it is possible that roughening of the cladding surfaces by adsorption of oil degradation products might have been a contributory factor.

In parallel with these out of reactor examinations of fuel assemblies, a device was developed which would measure flow through each of the remaining fuel assemblies and through some of the blanket assemblies. Use of this device showed acceptable flows through all of them. Work on the reactor was supported by laboratory studies to examine the effects of temperature, sodium and irradiation on both new and degraded oil. These included tests on the PROTVA rig at the IPPE at Obninsk where oil was injected into sodium upstream of a PFR fuel assembly inlet filter, and studies on a National Nuclear Corporation (NNC) water rig to examine the blockage-forming potential of material similar to that observed in PFR.

A major effort was required to remove all three valve and filter assemblies from the reactor for examination. These were the longest components in the reactor vessel, at 12 meters, and required considerable care in handling. Examination showed that at least one panel of each valve filter had failed, and oil-related debris was found on all the filters. A number of the fuel subassemblies which had showed outlet temperature rises during the incident were removed, and oil-related debris was found on their inlet filters and wrappers. The result of the oil ingress was an 18-month shutdown while PSP valve and filter assemblies were removed and new filters were fitted. The pump seal oil systems were modified to prevent any further possibility of oil ingress, and alarm and trip systems were added to prevent blockage of the pump filters in order to protect the subassembly filters.

Very fine particles of carbon were found in primary sodium samples after 1974. These are believed to have come from the 1974 oil ingress, and it appears that in the long term oil debris breaks down into finely-divided carbon particles which are dispersed in the sodium, pass through the filters, and circulate without obvious effect.

Oil ingress into the primary circuits of an LMFR is undesirable because of the potential release of methane gas through the core causing reactivity effects, and possible blockage of the subassemblies by solid carbon debris. In the case of PFR no reactivity effects were seen, possibly because the oil was retained in the pump cone for a prolonged period and broken down slowly without the formation of large bubbles. In the long term oil bearings are probably best avoided. The EFR design was changed following the PFR oil ingress incident by the introduction of the innovative features of magnetic bearing and 'ferro-fluid' seals to eliminate oil completely and remove the potential hazard of its ingress into the sodium.

2.5. CRACKS IN THE PFR AIR HEAT EXCHANGERS

In the decay heat removal system of PFR, leaks were detected in the air heat exchange (heat exchange between the NaK circuit and the atmosphere). These were associated with anomalous temperature differences between tubes in the heat exchanger, due to aspects of the design together with difficulties in achieving filling with NaK. PFR had three thermal syphon decay heat rejection loops (Fig. 17).

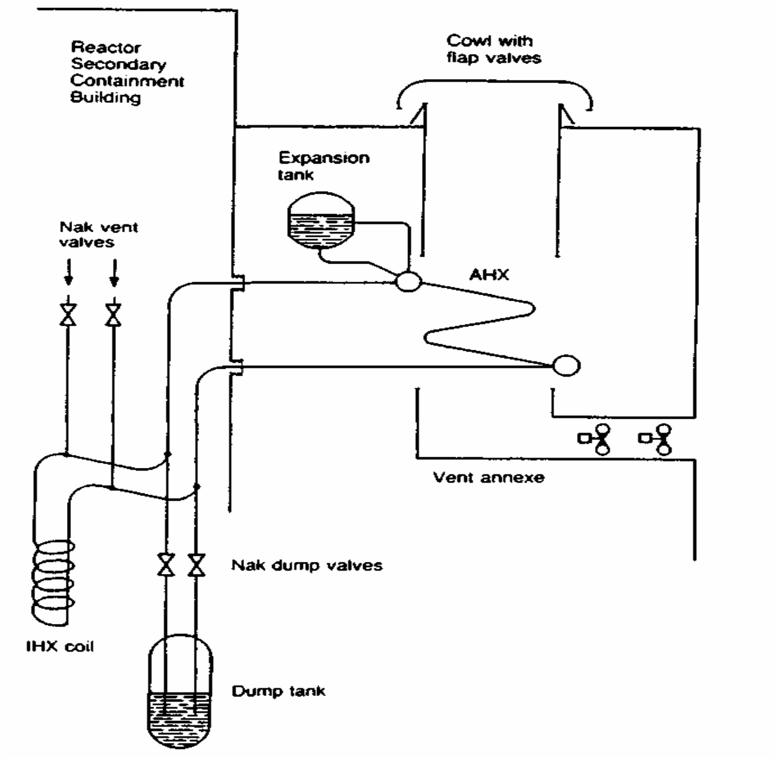

FIG. 17. Schematic diagram of a PFR thermal siphon decay h eat rejection loop.

The ten tons of NaK were contained in this system. Each consisted of a NaK filled loop connecting a heat exchanger coil positioned in the main reactor vessel adjacent to an intermediate heat exchanger to an air heat exchanger (AHX) on the roof of the reactor containment building. In the event of loss of electric power supplies each loop was capable of removing 1.5 MW of decay heat from the reactor by natural convection.

Each AHX was equipped with 2 fans connected to emergency diesel power supplies, which could enhance the decay heat removal to over 4 MW per loop. When the rector was operating normally heat removal was limited by dampers which restricted the airflow to the AHXs. Although the thermal syphon system operated well by 1984, it had become apparent that the AHXs suffered from a systematic fault leading to failures and leaks. Each AHX consisted of forty serpentine parallel tubes welded to pulled tees in two headers, as shown in Fig. 17. The tubes were finned along the straight lengths but plain at the bends, which were clamped together and supported. Further rigidity was provided by cleats which were welded to the tops of the fins on adjacent tubes. Flow of NaK was from the top down. Leaks were occurred at the welds between the tubes and the pulled tees in the headers. As an interim measure operational constraints were imposed as the frequency of failures could have invalidated the risk analysis in the safety report, and hence jeopardised the authorisation to operate the plant. Meanwhile the AHXs were heavily instrumented with strain gauges and thermocouples to identify the cause of the problem and indicate a solution.

The measurements indicated that the problems occurred essentially because the AHX tubes were in parallel, and were horizontal with no fall to ensure good filling. When the AHXs were filled, gas locks were occurring at the pipe bends. The gas-locked tubes remained cold, and as a result oxide impurities could be precipitated causing permanent blockages. Because of temperature differences between a cold blocked tube and the adjacent hot tubes to which it was clamped, large stresses were imposed. As a result the weakest point in the system, the weld between the tube and the header, was stressed, suffered cracking and eventually leaked. Replacement AHXs (RAHXs) were manufactured to an improved design which avoided the problem of gas locks and afforded greater toleration of loss of flow in individual tubes. The new RAHXs were fitted in 1986/1987. Operation was trouble free until 1996, when the thermal syphons were finally emptied for decommissioning. The following changes were made and are shown in Fig. 18.

— A 2° slope was given to the tubes to give better venting and drainage;
— Each tube was given individual support;
— The tube-header connections were reinforced;
— Larger diameter headers were fitted to give better NaK distribution.

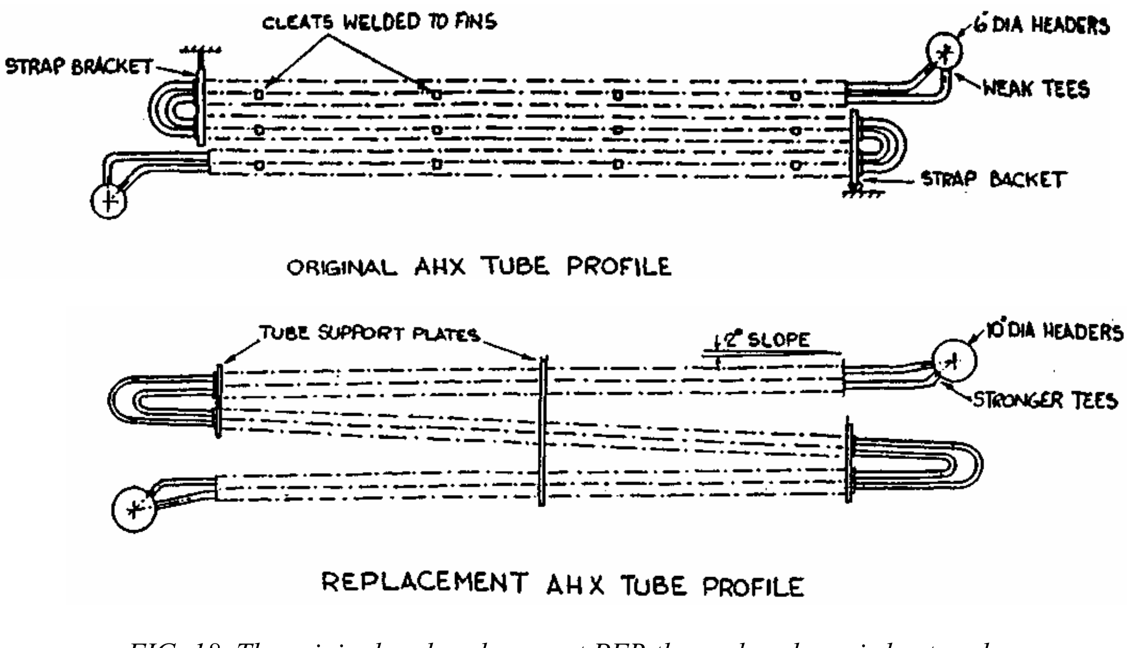

FIG. 18. The original and replacement PFR thermal syphon air heat exchangers.

In 1984 a common mode failure problem in the PFR thermal syphon AHXs was jeopardising the plant authorisation. A rapid research, development, manufacturing and installation programme solved the problem by 1986. It is notable that the design feature essential to solving the problem was the inclusion of a simple 2° slope on the tubes.

2.6. SODIUM MIXING PROBLEMS AND FLOW AND MECHANICALLY INDUCED VIBRATIONS

At PFR, whenever it was necessary to operate on less than three secondary circuits, there was a small flow of sodium at core outlet temperature past nominally shut sleeve isolation valves in the intermediate heat exchangers which are not in service. This provides the potential for mixing of sodium flows at core inlet and outlet temperatures and hence possible thermal cycling damage to the heat exchangers and containing pod structures. These effects were extensively studied in laboratory rigs and resulted in constraints on the core temperature rise when operating on less than three circuits. The performance of the intermediate heat exchangers has been examined both theoretically and experimentally to establish a satisfactory position.

Flow induced vibrations, together with mechanically induced vibrations have been an important cause of unplanned outage. Two of the PFR failures (one primary and one secondary pump) occurred during early commissioning. The primary pump failure resulted from incorrect finish machining of the shield plug, causing shaft distortion and damage to the hydrostatic bearing. Two secondary pump failures, the latest in April 1984, were due to hydrostatic bearing seizure caused by detachment of part of the bearing surface or a foreign body in the sodium. Table 1 summarizes the experience with sodium pump performance at PFR.

TABLE 1. SUMMARY OF THE EXPERIENCE WITH SODIUM PUMP PERFORMANCE AT PFR

Operating hours		Failures	Failures
Primary	Secondary	Primary	Secondary
405 965	231 960	1	2

2.7. THE EFFECT OF NEUTRON-INDUCED DISTORTION OF CORE COMPONENTS

The core is divided into modules of six subassemblies, each with an associated control rod or support position. The diagrid is provided with a hole at each one of these module positions into which a pressure chamber, described as the fuel element carrier, is fixed by a single bolt.

The fuel element carrier is shop-fabricated and can be removed from the reactor relatively easily should it be damaged. In the central position of the fuel element carrier, a support post has been fixed which is usually referred to as the leaning post and it carries some support pads at a position level with the bottom of the axial breeder of the core where the damage flux is relatively modest. In each carrier, there are six upper and lower bearings which carry the support spike of the six subassemblies. The lower bearing is offset outwards about 1/8 in. and the bottom spike of the subassembly consists of an upper bearing fixed rigidly to the sub-assembly envelope and a lower bearing mounted on an elastic cantilever.

Figure 19 shows schematically the support system for the PFR core. As the subassembly is lowered into the fuel element carrier, the lower bearing of the subassembly is pushed across by the offset lower bearing of the fuel element carrier and this forces the subassembly against the pad provided on the leaning post. The pad loads are set high enough to prevent the subassembly vibrating due to the coolant forces generated at the subassembly exit. The lower bearing offset is, therefore, determined by this required load and the sum of all the tolerance effects, such as sub-assembly bow. The upper bearing is provided with a step a few thousandths of an inch high to ensure that the subassembly pivots correctly about this bush.

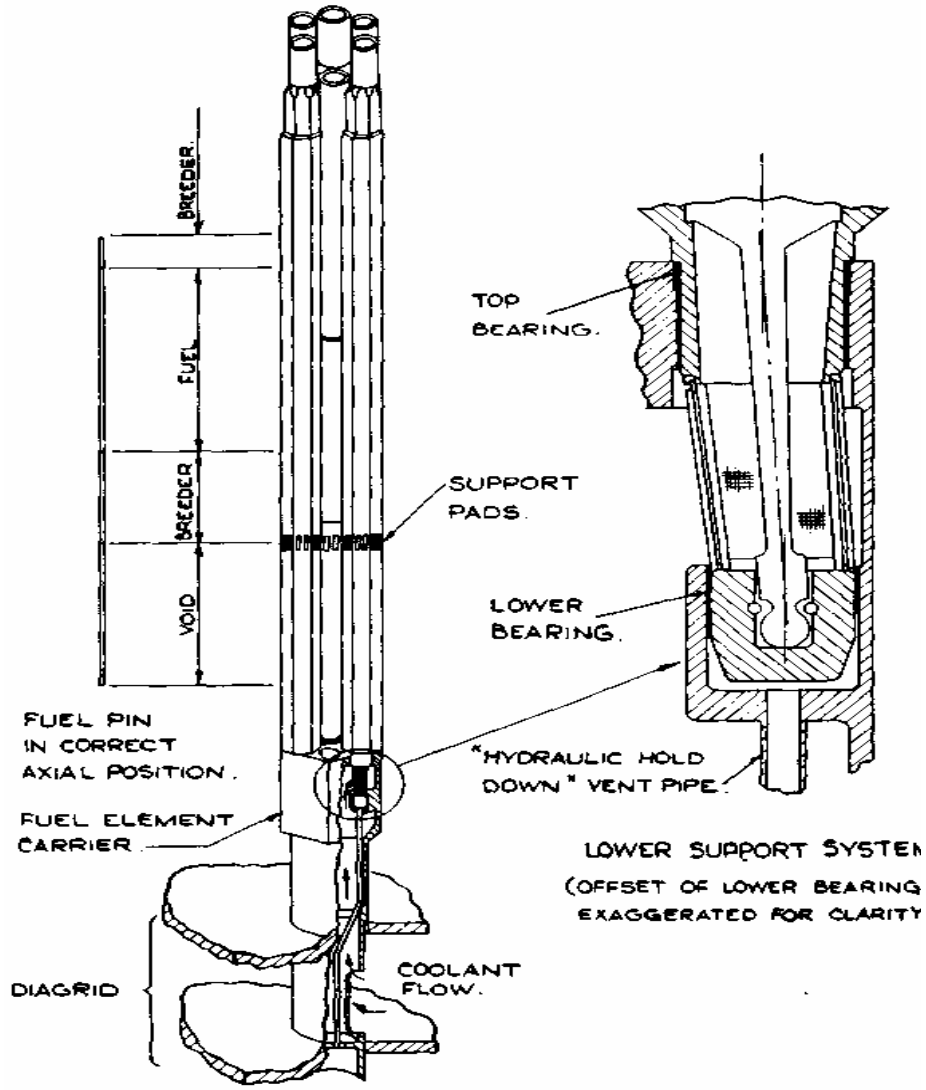

FIG. 19. PFR fuel subassembly support system.

Radiation damage resulting from the high neutron fluxes and operating temperatures of a fast reactor can give rise to dimensional changes in core components. The mechanisms involved are swelling caused by neutron-induced voidage (NIV) and radiation creep. These phenomena affect core components by causing axial extension, bowing in transverse gradients of neutron flux or temperature, and dilation. NIV was first detected during post-irradiation examination of components from the Dounreay fast reactor (DFR) in 1965. PFR had been designed in 1963 without taking account of the need to accommodate the effects of NIV.

In consequence calculation routes had to be developed to predict the distortion of PFR core components so that they could be managed in such as way that operation would not be impeded. In particular it was essential to be able to ensure that no core component was at risk of becoming so distorted that it interfered with the movement of the absorber rods or could not be removed. The calculations, including the important effects of interaction between components, were based on empirical material deformation rules obtained from post irradiation examination of irradiated components. They were successful in guiding operations except when problems arose due to unexpectedly rapid growth of particular materials.

It was necessary to predict the bowing of fuel subassemblies in order to prevent handling problems. The operating limit was 14 mm bow at the subassembly shoulder. Bows beyond 21 mm at the subassembly shoulder would have presented difficulties when it came to extraction from the core. Subassemblies were routinely rotated through 180° part way through their residence in the core in order to correct the bowing.

Immediately before refuelling operations in PFR three sweep arms were employed to ensure that there were no obstructions above the core which would prevent rotation of the rotating shield. At the start of a reload in 1988 the sweep arms were found to contact or partially contact objects in two core positions. These positions were identified as containing subassemblies with cold-worked EN58B steel wrappers, with a calculated dose of greater than 60 displacements per atom (dpa).

Using a special tool the heights of all subassemblies of the same material were checked. A distinct trend of rapid increase of growth at doses above 50 dpa was revealed, although not all subassemblies were affected. Two subassemblies in particular, “JRA” and “GYN”, had measured growths of about 40 mm. As a result all components with predicted doses likely to exceed 50 dpa by the end of the next run were removed from the core. Considerable difficulty was experienced in handling the severely distorted components and special tools had to be manufactured for their extraction and removal from the reactor.

Although the absorber rods and associated components in PFR were manufactured from nimonic PE16, an alloy known to be subject to low swelling, it was important to ensure that NIV distortion would not prejudice operation of the system or hinder rod drop in a scram. The major cause for concern was distortion of the guide tube in which the absorber rod moved, either by NIV bowing or by pressure on it from adjacent bowed fuel subassemblies. In addition to the calculations, regular exercising of the absorber rods over their full stroke gave assurance that no such problems were arising.

On only one occasion there were observable effects in the operation of an absorber. In this instance a shut off rod developed unusually high and increasing friction at the top of its stroke while being exercised. Although the rod operated correctly during subsequent trips, indicated that the problem was probably caused by interaction of the guide tube with an adjacent distorted fuel subassembly.

The subassembly was discharged, and PIE confirmed the analysis. It was another subassembly clad in cold-worked EN58B, with higher than expected swelling. The allowed doses for EN 58B was reduced to prevent further problems of this sort. Large differences in NIV swelling rates could occur in different batches of the same material.

This led to handling problems in the case of components made of cold-worked EN58B. Materials chosen later in the lifetime of PFR, such as nimonic PE 16, had considerably lower swelling rates. Components manufactured from the ferritic steel FV 448, which was under test at the time of PFR closure, had extremely low swelling rates. NIV distortion was not expected to be life-limiting for this material.

2.8. FUEL DEVELOPMENT

The first fuel assemblies containing plutonium from PFR recycled into new oxide were loaded into PFR in June 1982, thereby closing the reactor fuel cycle. In March 1994, the reprocessing plant had treated a total of over 23 tons of oxide fuel, recovering more than 3.5 tons of plutonium, from 239 fuel assemblies, 144 mixer breeders and 10 radial breeder assemblies. The highest burnup in the fuel was 17.6%. Typical cooling times were 270-360 days, but the shortest cooling time was 136 days when some short-cooled fuel was reprocessing experimentally to study movements of iodine species. The plant has been operated since 1980 in campaigns to keep pace with fuel discharges from PFR. The highest throughput in any year so far was 4.79 tons in the period from April 1993 to March 1994.

The PFR fuel reprocessing plant proved the technical feasibility of oxide fuel reprocessing via a Purex cycle, with recovery of over 99.5% of the plutonium. This high recovery was also reflected in the low amounts of plutonium in the liquid and solid waste streams from the plant. The amount of radioactivity discharged to the environment was always about an order of magnitude less than the licensed limits. The plant is subject to IAEA and Euratom safeguards.

One of the principal tasks of PFR was to demonstrate a reliable, safe and robust fuel capable of routinely achieving a high burnup target. Successful completion of this task was one of PFR's major achievements. The major advances made in the second decade of PFR operations resulted from the introduction of Nimonic PE16 as the reference cladding alloy. The fuel assembly discharged at 15.9% burnup in 1986, mentioned earlier, gave the first indications of the benefit of this change. The pins showed maximum diametral increases of only 1%, with uniformly low diametral change profiles showing little pin-to-pin variability, compared with the 5-8% (maximum) diametral changes and highly peaked profiles showed by earlier examinations of pins clad in cold-worked M316 steel and irradiated to half the exposure.

Destructive examination of the pins indicated that the fuel column was stable, that internal corrosion was low and that there was no evidence of any fuel/clad mechanical interaction resulting from containment of a high burnup swelling fuel in a non-distending cladding tube. This confirmed that higher burnups were probably feasible. Measurements of the PE16 wrapper showed trivial length increase, across-flats distension and bowing, and revealed no potentially life-limiting changes. By 1990, irradiations of PE16 clad fuel pins in driver assemblies and in experimental subassemblies (Sas) had achieved more than 17 and 21% burnup, respectively. Even with displacement doses of the order of 130 displacements per atom (dpa) no life-limiting features could be identified in either the wrappers or the cladding.

Meanwhile materials irradiation experiments had shown ferritic/martensitic steels to be particularly resistant to void swelling. The high temperature mechanical strength of these materials seemed to preclude their use as pin cladding. The application as wrapper materials seemed to be practicable. Accordingly, in the late 1980s driver fuel assemblies with wrappers of the ferritic/martensitic steel FV448 and pins clad in Nimonic PE16 were introduced into PFR. By the end of the reactor's working life, over 20 such assemblies had been loaded and seven of these had exceeded the 15% burnup (110 dpa) target; one had achieved the then world record, for a mixed-oxide driver charge assembly, of 19.8% burnup (155 dpa). Statistics illustrate the undoubted success of the fuels development program in PFR. Approximately 98 000 pins were irradiated and, of these, over 40 000 exceeded the original 7.5% target burnup. The introduction of PE16 as cladding allowed over 2 400 pins to attain burnups in excess of 15% with about 320 of these having successfully exceeded 20% at the end of operations. The peak burnup (in lead pins irradiated in an experimental SA) was 23.2%.

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3. PHENIX AND SUPER-PHENIX REACTORS

3.1. PHENIX REACTOR

3.1.1. Commissioning and design features

A description of the French demonstration prototype sodium cooled fast breeder reactor Phénix has been given in an earlier IAEA publication and a comprehensive listing of design and operational parameters is presented in another. Therefore, it is not proposed to give an extended description of the plant here, three figure show the general arrangement of the reactor: isometric view, the elevation through Phénix primary circuit and flow diagram, describes below.

The reactor plant Phénix with a nominal ~ 255 MW(e) power rating (565 MW(th)), was firstly connected to the electricity grid on 13 December 1973; the nominal power was reached on 12 March 1974, 18 days ahead of plan.

The nuclear power plant (NPP) construction cost was 800 million franc (approximately 3.2 billion franc 2000, that is equivalent to ~ 512 million Euro). The planned budget was exceeded by less than 10%. The NPP was generally operated at the power tolerated by reactor and equipment, with comparatively high load factor. Phénix has currently provided about 100 000 hours of grid-connected operation representing 3 900 equivalent full power days at operating temperatures of 560°C for the reactor hot structures. The plant has achieved the objectives of demonstration of fast breeder reactor technology which were set at the time of construction, including the following significant achievements:

— An average burnup was increased from 6%[1] heavy atoms (h.a.) to 11% h.a. at the center of the core and to 13.5% h.a. at the periphery of the core. These levels were reached with eight cores of fuel which was 170 000 fuel pins; experimental pins and subassemblies have reached maximum burnup exceeding 17% h.a. and an irradiation dose of 150-160 dpa (displacement per atom) on the fuel cladding;
— The measurement of the breeding ratio made at the time of dissolution of the fuel evacuated from the plant, gave a true value of 1.16 (projected 1.13);
— The fuel cycle, based on mixed oxide fuel and PUREX reprocessing, has been closed and the first fuel subassembly made with reprocessed plutonium was loaded in the reactor in January 1980. About of five of Phénix cores (~ 25 tons) were reprocessed.
— The successful and regular operation at the highest temperatures and nominal power until 1990 resulted in validation of the pool concept option and did much knowledge regarding the high temperature design and structural material of fast reactors;
— The highest in the nuclear power engineering practice gross/net plant’s thermal efficiency of 45.3/42.3% during the periods of stabilized operations with nominal parameters. On the average, gross/net plant’s thermal efficiency is equal to 40/38% owing to operations at 2/3 the rated power now and then.

From 1992, the role of Phénix as an irradiation facility has been emphasized, particularly in support of the CEA R&D programme in the context of line 1 of the 30 December 1991 law on long-lived radioactive waste management. The first experiment, called SUPERFACT, led to the incineration of minor actinides (neptunium and americium). This programme was further strengthened in 1998, to compensate for the shutdown of Super-Phénix. It involves

[1] Planning at the design stage the high burnup was a big risk owing to stainless steel fuel pins cladding swelling observed in the Rapsodie in 1971.

transmutation of minor actinides and long-lived fission products. Since 1993, the reactor power has been limited to 350 MW(th), 145 MW(e) on two secondary loop operations.

3.1.1.1. Reactor block (Fig. 1a,b)

The Phénix reactor block is of an integrated (pool) design except for a few auxiliary circuits. The entire primary sodium system, containing 800 tons of radioactive sodium, is enclosed in the main reactor vessel. The reactor block is suspended to the slab via 21 hangers. These hangers have three welds, which are difficult to access.

The reactor vessel dimensions: inside diameter×height: 11.82×12 m, wall thickness: 15 mm is under the upper slab, 15 m in diameter and 1.5 m high, on which the equipment is installed. The slab (~ 800 tons: 200 tons metal and 600 tons concrete) is installed on 22 ball- and -socket type pads. The vessel cover plate is under high temperature conditions, similar to the BN-600 reactor, but in this case it is unloaded. The slab bearing plate operates under low temperature because between the plate and the reactor cover there is insulation and besides this zone is cooled. The reactor vessel was manufactured on site workshop using 15 to 25 mm cynical sheets after prefabricated elements were examined in detail.

The pumps and heat exchangers are located on movable sliding support, sealing of the pump and heat exchanger penetrations is carried out by means of the bellows. A peculiar feature of this reactor top design is a massive cover plate, of 60 mm thick and 12 000 mm in diameter having significant thermal inertia. It restraints a power growth rate or the reactor starting up and may be a cause of some thermal stresses occurring in the junction of the vessel cylindrical part and the horizontal cover.

The sodium in the vessel is separated into two zones:

- Hot pool at the core outlet where the hot sodium flows into the intermediate heat exchangers;
- Cold pool taken from a peripheral annular space between the primary tank and the wall of the main reactor vessel, which contains the three main circuit circulating pumps and six heat exchangers, suspended from the upper slab;
- The leak tightness of the penetrations of the reactor vessel by the IHX is provided by an argon seal.

A number of other devices are located in the main vessel: the fuel transfer arm, the six control rods, neutron flux detectors, thermocouples, failed fuel detection and location devices, the core acoustic detection system components, etc. An argon gas atmosphere is maintained above the sodium surface to prevent any contact with air.

The main vessel is closed at the top by a flat roof with openings for pump and heat exchanger pipes. It is associated with the cylindrical seating of a rotating plug in the slab penetrations forming the top of the reactor block. An outer guard vessel surrounds the main vessel. It has the double function of containing any sodium escaping by leakage, and preventing a drop in the sodium level of the main vessel which might affect core cooling.

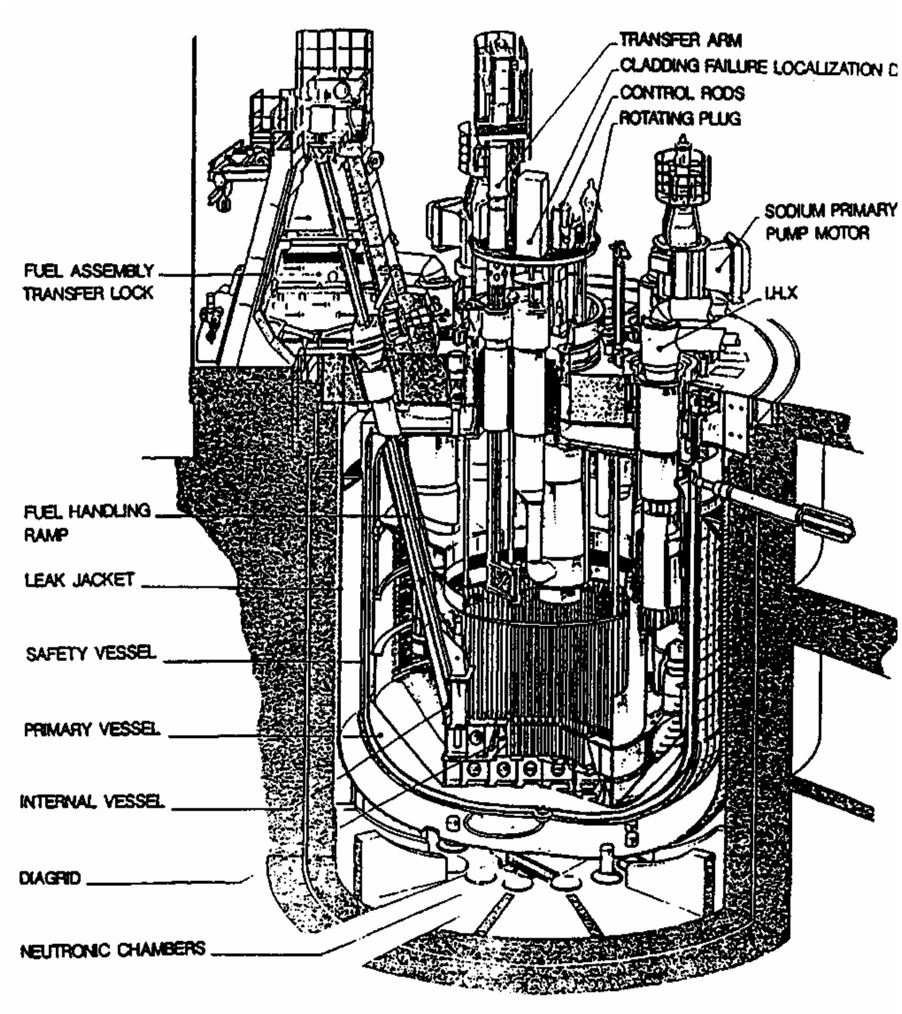
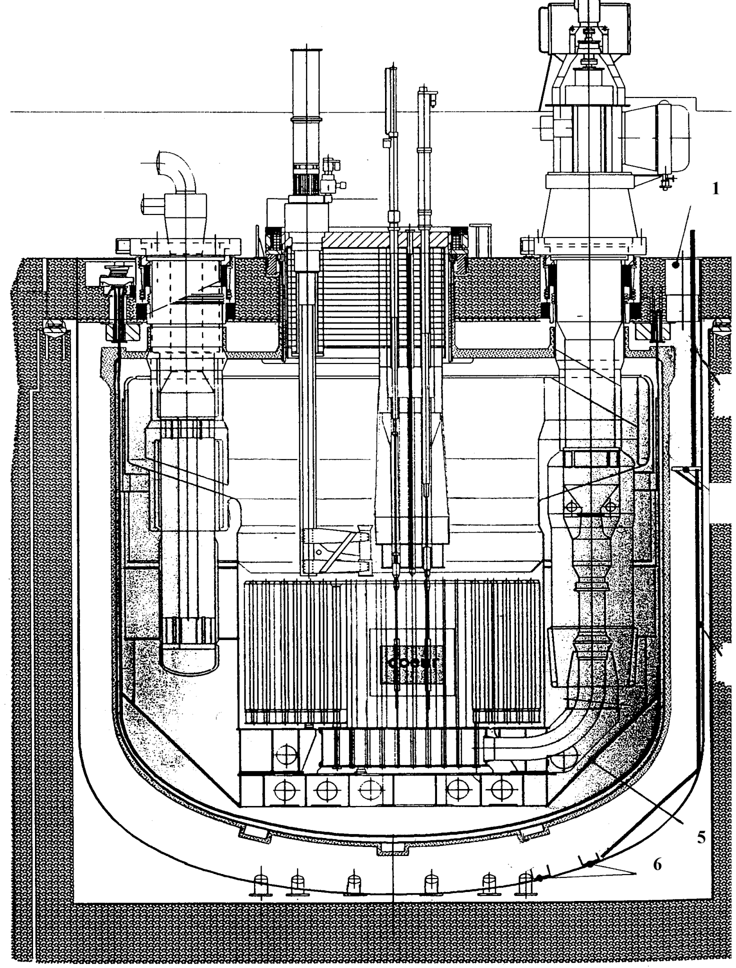


FIG. 1a. Phénix reactor block: isometric view.



1-manhole (Ó =5 00mm), 2-movable ladder, 3-catwalk, 4-fixed ladder, 5-conical skirt, 6-cable tracks

FIG. 1b. Elevation through Phénix primary circuit.

3.1.1.2. Heat transfer circuits (Fig. 2)

Owing to an intermediate circuit between the reactor and the steam generator, it is very likely to prevent an accidental interaction between the radioactive primary sodium and the water/steam in the electricity generating system. In such nuclear steam supply system design, the incident-secondary sodium-water reaction could be classified as a chemical incident in the non nuclear components.

The three primary sodium pumps are variable speed units (150 to 970 rpm) delivering about 950 kg/s at 825 rpm, which is their normal service speed. The circulating sodium enters the core at 400°C and moves from there, at 560°C, to six intermediate heat exchangers which are connected in pairs with three independent secondary loops.

Sodium must be kept very pure to prevent corrosion of the steel piping and plugging of circuit components. It is purified by cold traps operating on the principle of precipitation of any oxide in the sodium at low temperature.

Secondary sodium, which is not radioactive, is circulated by a mechanical pump with a flow delivery of 700 kg/s. It enters the intermediate heat exchangers (IHX) at 350°C and leaves at 550°C. Each secondary loop is connected to a steam generator consisting of an evaporator, superheater, and reheater, in 12 modules for each stage.

3.1.1.3. Reactor core

The reactor core, in which most of the reactor power is generated, is surrounded by a fertile blanket and neutron shielding to prevent activation of the secondary sodium flowing through the intermediate heat exchangers. The fuel is uranium dioxide mixed with plutonium dioxide (UO_2-PuO_2). It is contained in 103 subassemblies, each containing 217 pins, which in turn consist of a stack of sintered oxide pellets, 5.5 mm in diameter, enclosed in a stainless steel cladding.

The pins are assembled in clusters in a stainless steel outer shell, which also contains the upper and lower fertile blanket pins (depleted uranium oxide) and the upper neutron shielding. The radial blanket is composed of depleted uranium dioxide pellets measuring 12.15 mm in diameter, in 90 assemblies of 61 pins each. The structural components of these subassemblies are identical with those of the fissile subassemblies, with sodium flow through the spike inserted into the diagrid.

The first fuel load consisted of 50% MOX and 50% enriched UO_2, during the reloads the proportion of MOX fuel increased and after four years of operation were 100% MOX.

During the first two years operation was interrupted mainly by fuel reloading and by minor incidents including the core seven fuel pin failures. The pin failures were easy to locate by wet sipping at the subassembly outlets and less than three days loss of energy production.

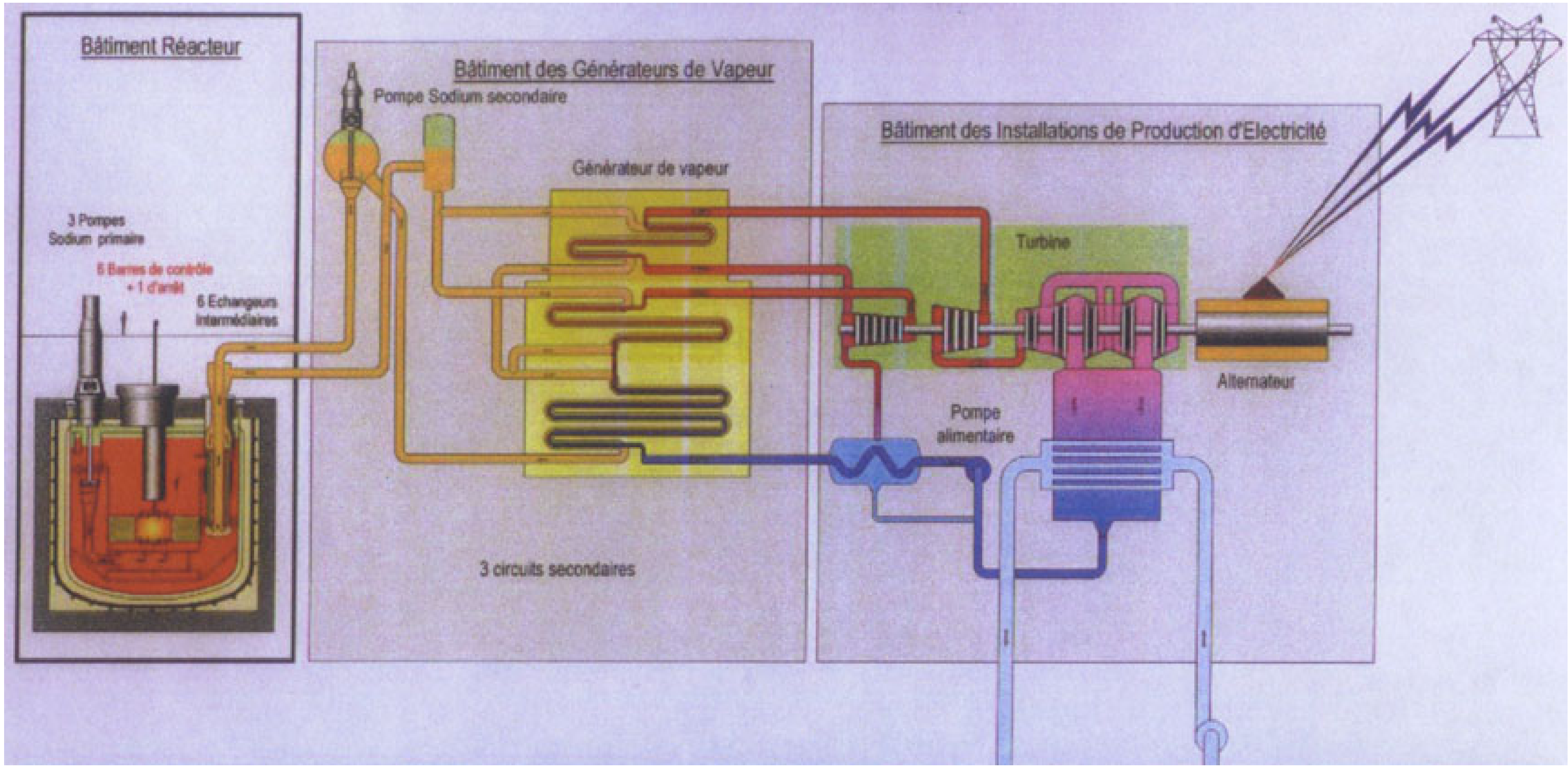


FIG. 2. Phénix flow diagram.

3.1.1.4. The emergency cooling system

An outer leak jacket, which was made of carbon 18 mm thick cynical shells aims at containing any radioactive products that might escape from the main vessel in the event of an accident. This containment is cooled by a water circuit which maintains the concrete of the reactor block at a low temperature, and which is capable of acting as a standby cooling circuit for decay heat removal after shutdown, should all the secondary sodium circuits being out of service.

Under faulted conditions the decay heat is removed from the double wall reactor tank to a third, so called safety tank, which has a water-filled tube coil system on its outer surface. The 12 cooling sub-circuits with a total length of 4 300 m of pipes were welded on the outside surface. To improve the heat transport between the double tank and the safety tank it was decided to replace the nitrogen by helium totally or partly. The emergency cooling system comprises two independent circuits, each supplying water to half the tubes and coils, three pumps, and two heat exchangers.

The two circuits are isolated from one another by a single valve. Of the three pumps two are normally operating, with the third on standby. In the event of loss of off-site power, the pumps are powered by the diesel generators after 3 seconds. Heat removed by the coolant is transferred to raw water circulating in heat exchangers. Lately, during inspection, repair and safety upgrade the two independent heat exchange circuit (air/water) for decay heat removal in the event of loss normal means have been installed and tested; a 1 500 kW air cooler and 720 kVA stand alone added.

The emergency cooling circuits have a secondary permanent function: during normal operation, treated water is circulated through them to remove heat from the cylindrical part of the safety vessel and its lower head, the roof slab including penetrations, the primary containment concrete penetrations for auxiliary system pipes, and the upper part of the vessel. The following major incidents or unforeseen events are shown in Table 1:

— Series of leaks of secondary circuit sodium in the IHX;
— Sodium-water reaction in the steam generators;
— Negative reactivity trips;
— Cracking of welded joints on certain parts on the main secondary pipes and some components, particularly in austenitic steel like 321 stabilized with titanium over about 30 years operation in three positions:

1973–1990	Demonstration of fast reactor technology and closed MOX fuel cycle;
1990–1993	Investigation after negative reactivity shutdowns; and
1993–present time	Renovation, test and operation with limited reactor power, 350 MW(th), 145 MW(e) on two secondary loop.

TABLE 1. THE MAIN EVENTS AND THE GRID-CONNECTED OPERATION TIME IN THE RELEVANT YEARS

Main events	74	75	76	77	78	82	83	84	86	88	89	90	98	00	03
IHX secondary circuit sodium leaks			**X X**	**X**	**X**			**X X**		**X**			**X**	**X**	
Steam generator leaks		$\mathbf{X}^1$ $\mathbf{X}^1$	$\mathbf{X}^1$ $\mathbf{X}^1$			$\mathbf{X}^2$ $\mathbf{X}^2$	$\mathbf{X}^2$ $\mathbf{X}^2$								$\mathbf{X}^2$
Secondary circuit main pipe sodium leaks	**X**	**X X**	**X**						**X X**	**X**					$\mathbf{X}^3$ $\mathbf{X}^4$
Negative reactivity shutdowns			**X**		**X**						**X** **X X**		**X**		
Grid-connected operation time, %	81	68	54	24	67	62	63	71	80	72	31	Investigation after reactivity transient, renovation, test and operation			

$\mathbf{X}^1$-water leaks into the evaporator box space through the sub-header's shell wall;
$\mathbf{X}^2$- sodium-water reaction;
$\mathbf{X}^3$-leak in the bellow of the sodium purification system valve;
$\mathbf{X}^4$-leak in the electromagnetic pump of the steam generator hydrogen detection circuit.

3.1.2. Intermediate heat exchangers operating experience

The intermediate heat exchanger (IHX) transfers heat from the primary radioactive coolant to the secondary non-radioactive coolant while maintaining a physical barrier between two. The IHXs are connected two by two to a secondary cooling circuit and suspended from the upper part of the slab. The design of the heat exchanger is shell and straight tube, counter flow, with the primary coolant on the shall side; 2 279 tubes (outer diameter×wall thickness: 14×1.0 mm, 5 300 mm length, austenitic steel 316) are fixed onto lower and upper tube plates by expansion and welding. At the top pf of the IHX, the second sodium enters a central tube and flows to the bottom of the distribution box with a convex bottom welded to the lower tube plate.

In the early years of operation the only serious loss of power generation was caused by the need to repair the IHX as a result of leaks of inactive secondary sodium. Leaks were detected in the annular inter space between the cold inlet and hot outlet ducts in two separate instances. Three successive incidents affected IHX: on 11 July 1976, on 3 October 1976 and on 31 August 1977 led to complete shutdown of the plant for about ten months to allow for disassembly and examination of the damaged IHX. The NPP restarted and operated at two thirds of its rated power with four repaired/modified IHX and dummy devices while repairs of the IHXs.

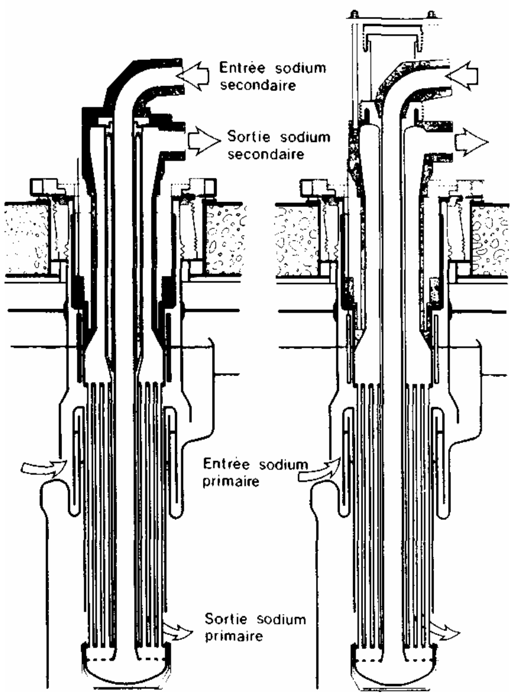
In March 1984 a new secondary sodium leak occurred in the IHX. The leak was found by detection of the sodium in the annular space separating cold and hot secondary sodium. The leak was very small and it was possible to continue the operating cycle for almost two months, at the end of which it was decided to remove the IHX and to replace it with a spare one.

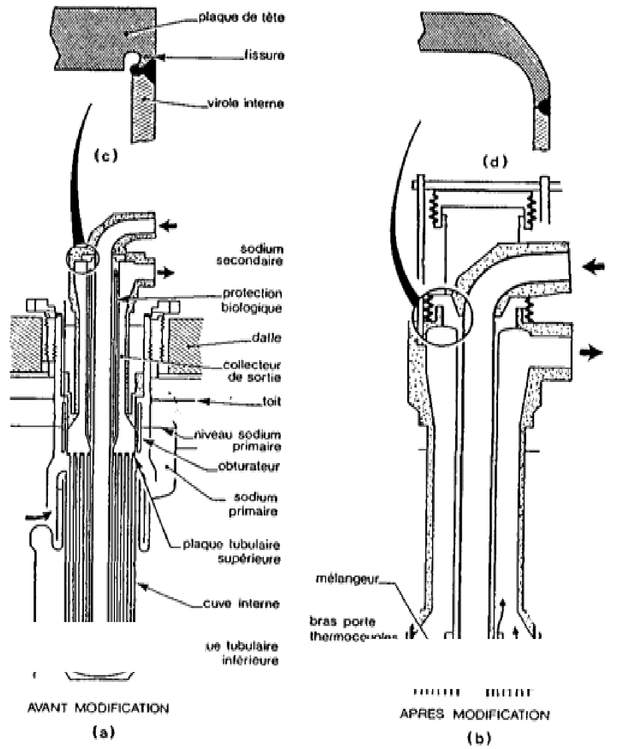
Replacement was carried out in July 1984, but in December 1984 the same defect appeared on this IHX, forcing the plant to operate at two-thirds power for nine months while repairs were made. It was concluded that these two successive leaks were not generic defects but were the result of two repairs made in 1977 after the first series of IHX incidents.

In October 1988, a leak of secondary sodium into the annular interspace of the IHX was detected. The IHX was removed and replaced by a spare new one. This new exchanger was equipped with 66 thermocouples, some of which were on the primary side of the tube bundle. As before, the cause of the leaks was found to be an improper mixing of the secondary sodium at the outlet of the tube bundle: an important radial temperature gradient existed between the outer and the inner sodium flows, the former being hotter. As a result, mechanical constraint developed, leading to overstressing of welds in parts of the secondary sodium duct.

Modification to a revised design with improved sodium mixing capabilities (by a mixing device) and flexible design elements at the secondary outlet were necessary for each intermediate heat exchanger (Fig. 3).

The removal and modification one-by-one of the Phénix IHX was perhaps rather difficult maintenance task foreseen on pool type fast reactors. Its accomplishment, with the minimum of reactor downtime, represents a conspicuous success and demonstrates the importance of foreseeing events at the design stage, making adequate provision of equipment and space for the repair, planning and learning from previous operation.





before repair *after repair* *before repair* *after repair*

FIG. 3. PHENIX intermediate heat exchanger (IXH): places of failure, repair and modification.
[(a)-before modification; (b)- after modification (flexible design elements and a flow-mixing device in the sodium header at the tube plate outlet)].

Two IHXs were experienced leak from the tube bundles during the test and renovation period: in November 1998 (reactor in operation) and in November 2000 (reactor shut down). The cracks were found in the tube inner wall in the expansion zone, at the rolling area under the upper tube plate. Experts concluded, that the cause of the cracking was corrosion under stress owing to presence of soda or polluted secondary sodium hydroxide, or sodium during the draining phases for these IHXs (several openings for work on the secondary loop between 1995 and 1997). The damage in the Phénix IHX was due to a design error leading to deformation and stresses because of the different heat expansion of the external and internal rings at the sodium exit to the secondary system; the different thermal dilatation of two plates cause the crack[2].

Apparently, for a future IHX design a variable flow distributed should be provided inside the IHX tubes with a higher secondary or/and lower primary flow on the outer rows to improve the temperature distribution in the tube bundle; a mixing device could be also helpfully at the secondary outlet. Feedback from these incidents and developed technology was very valuable for next French fast reactor Super-Phénix.

3.1.3. Steam generator operating experience

The steam plant is composed of three steam generators (SGs); each with its own independent secondary sodium system. The SG is modular in each stage, which made it easier to replace. Each SG comprises of three stages: evaporator stage, superheater stage, and reheater stage. Phénix SG specifications are shown in Table 2.

TABLE 2. PHENIX SG SPECIFICATIONS

Item	Evaporator	Superheater	Reheater
Heat transfer area, m^2	12×26.3	12×14.6	12×18.5
Outer diameter and wal1 thickness of tube, mm	28×4	31.8×3.6	42.4×2.0
Tube length, m	60.8	26.9	21.09
Steam water inlet temp., °C	246	375	308
Steam outlet temp., °C	376	512	512
Steam outlet pressure, bars	175	168	34
Sodium inlet temp., °C	475	550	550
Sodium outlet temp., °C	350	475	475
Sodium flow rate, kg/s	740	423	317
Steam, water flow rate, t/h	250	250	223

Each evaporator is associated with a reheater and superheater; stage is made up of 12 modules in parallel. A module consists of a shell containing seven steam tubes surrounded by sodium (Fig. 4).

[2]INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Liquid Metal Cooled Fast Breeder Reactors. Technical Reports Series No. 246, IAEA, Vienna, (1985), p.65.

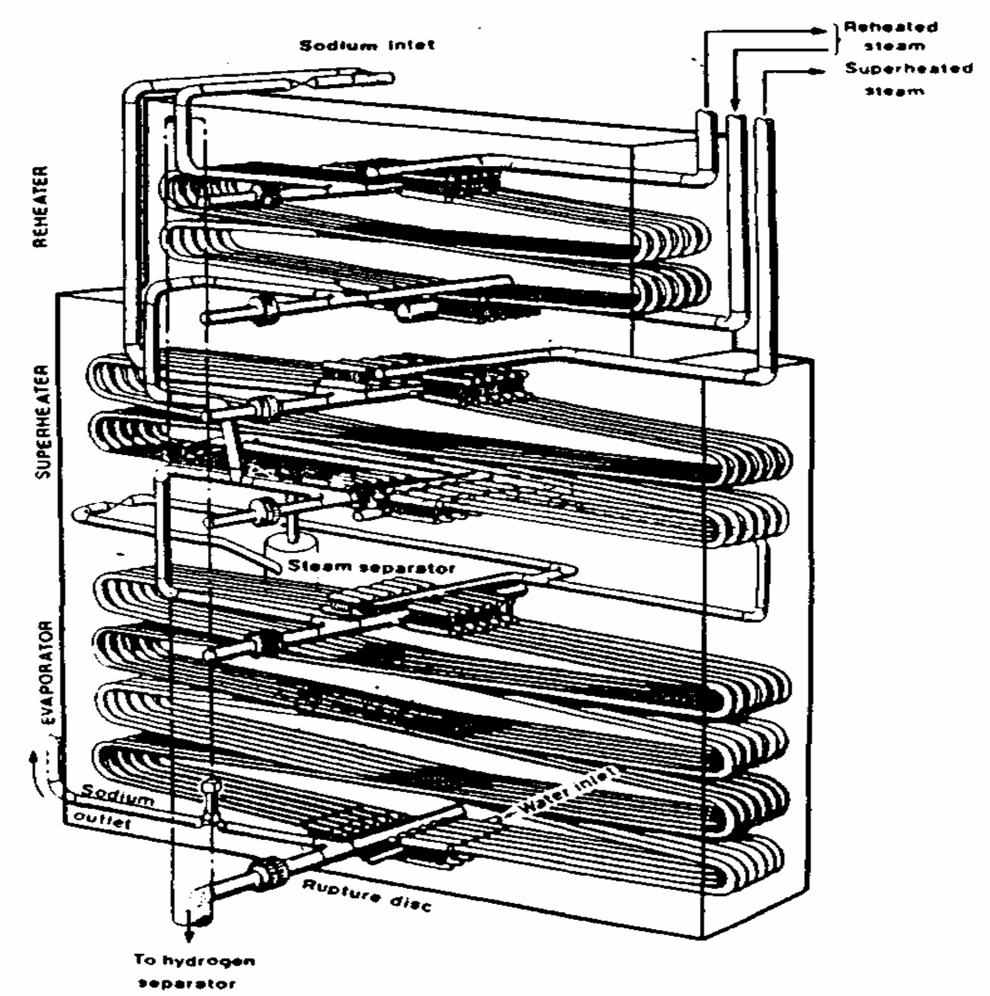


FIG. 4. Phénix SG design.

The SG is designed to be cooled with water or air during reactor shutdown. The hot sodium from the intermediate heat exchangers is distributed first between the superheater and reheater stages, and then the entire flow enters the evaporator stage. On the steam side, the stages are in series; the reheater stage is supplied by steam at partially reduced pressure from the turbine high pressure cylinder outlet.

The evaporator is made from ferritic steel 2.25 Cr-1 Mo whereas the superheater and reheater are made from austenitic steel 321 H. The superheater and reheater modules are S-shaped, or two S-shaped modules in the case of evaporators. Each module is composed of seven tubes placed inside a shell with water/steam flowing in the tubes and sodium flowing outside the tubes in the opposite direction. The steam generators are fitted with water leak protection system and detection devices to detect water or steam leaks in the sodium: hydrogen detects in the argon blanket of the expansion tank of each secondary circuit.

Main operating parameters under rated conditions are shown in Table 3.

TABLE 3. PHENIX MAIN OPERATING PARAMETERS UNDER RATED CONDITIONS

Characteristic	Value
Sodium inlet/outlet temperature, °C	550/350
Sodium flow rate in each SG, kg/s	740
Water inlet temperature, °C	246
Water flow rate in each SG, kg/s	210
Steam outlet temperature of the evaporator, °C	376
Superheated steam temperature, °C	512
Superheated steam pressure, bar	165
Reheated steam temperature, °C	512
Reheated steam pressure, bars	34

Four water leaks were happened in the evaporator inlet of the SGs between November 1975 and September 1976: subheader under frame was wearing out that was provoked by perturbations generated by the water flow distribution orifice plates in the evaporator tubes. The 252 orifices plates were replaced by new ones with improved design between October 1976 and February 1977. In 1982/83 leaks in reheater modules led to a period of operation on two circuits. On 29 April 1982 a reheater leak was detected by the hydrogen detection system; the secondary sodium side as well as the water side was dried out. The pressure of the nitrogen, which ought to have replaced the water, did not reach the required value of 0.7 MPa owing to the failure in the opening of a check valve in the steam generator. This allowed large quantities of sodium during the refill of the secondary system to penetrate into the water system. Twelve reheater modules and a related steam piping were polluted by the sodium-water reaction products. Finally, 6 L of sodium reached the air and caught the fire. This was extinguished within 2 minutes. The chronological order of the next three leaks occurrence:

Leak no.2: 16 December 1982 on reheater module 12 of SG 1;
Leak no.3: 15 February 1983 on reheater module 12 of SG 3;
Leak no.4: 20 March 1983 on reheater module 11 of SG 1.

In the case of leak no. 4, it should be noted that the no. 12 modules were new or had been removed. The following remarks may be made for a leak no. 1, 29 April 1982: SG isolation dry out was requested by the operator with about 5 minutes delay; this explains the large amount of wastage on the adjacent tube and the onset of wastage on the module shell. As pointed out before, after dry out, nitrogen injection into the reheater was not carried out due to a faulty non-return valve, and the large amount of sodium entered the steam tubes and the reheater connection pipes.

Only the first sodium-water reaction led to complications, with sodium contaminating the steam-side of the reheater modules. The radiographies detected holes in two tubes ~ 2 cm^2 that damaged the module’s shell. Nevertheless, the incident had no consequences for plant safety. When the leaks occurred, the SG had ~ 5 0000 hours of operation under rated conditions. Subsequent to this first leak, signal processing by the computer was improved for greater rapidity and operating instructions were also reviewed. These improvements proved beneficial during leaks nos 2, 3 and 4. Remarks for leaks nos 2, 3 and 4 are as follows: equipment and operators reacted well. Operator response time was only a few seconds.

The SG isolation-dryout sequence performed well. The number of failed tubes were one, and two for leaks nos 2, 3 and 4 respectively. The four leaks showed several common points. The fault occurred:

— On the reheater stages on the bends (Fig. 5);
— A short time (1 to 5 days) after a power plant start-up;
— Three times on module 12, once on module 11 (module 12 inlet pipe is the first of the reheater steam inlet main pipe);
— At a butt weld beads of the tubes and on one of the two welds on the hottest tube part.

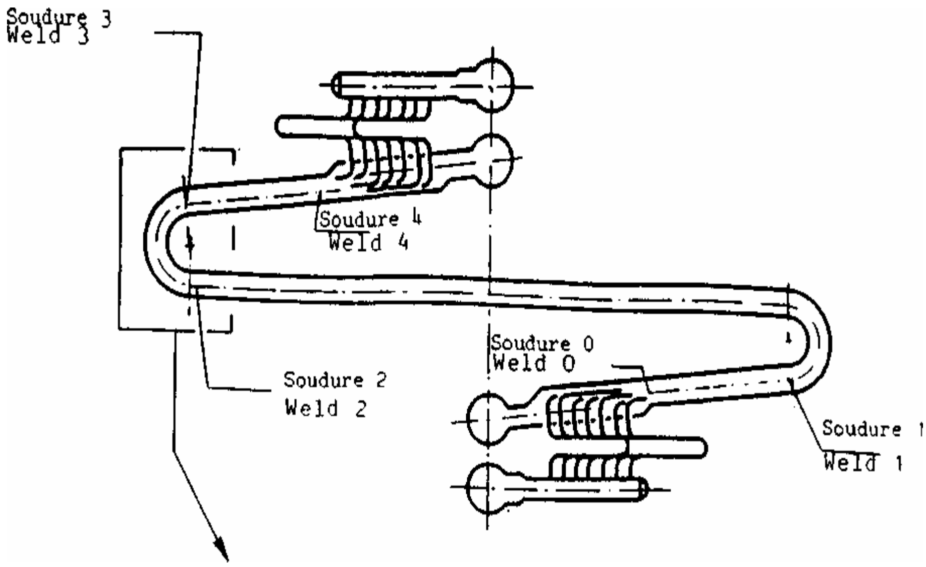


FIG. 5a

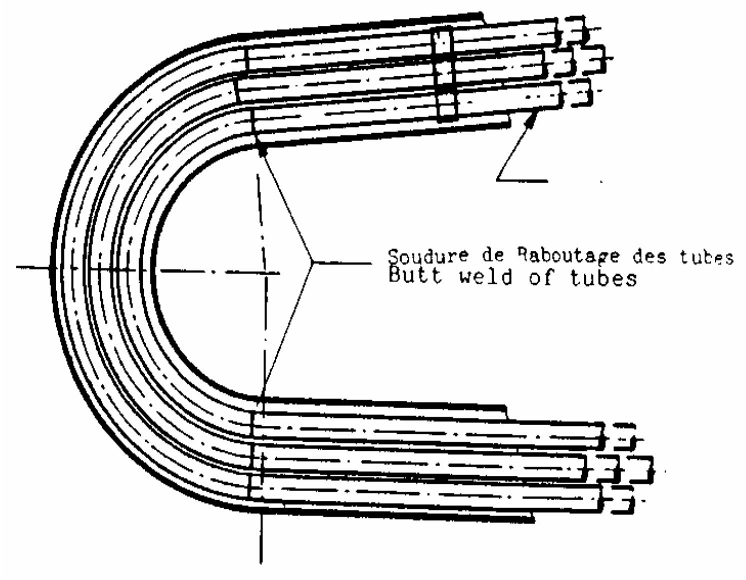


FIG. 5b

FIG. 5a,b. Phénix SG: location of leaks.

After expert analysis of various possible causes such as tube vibration, blocked differential expansion, chemical corrosion, etc., the most probable appeared to be the penetration of water into the reheater during the power plant start-up phase when the turbine bypass system is operating: when water abruptly turns to steam in the reheater tubes, that caused thermal shock which affect their resistance. The sodium outlet temperatures of the modules showed fast fluctuations with an increasing amplitude from module no. l to no. 12 (Fig. 7). These water induced thermal shocks and temperature variations caused major thermal stresses where the steel is thicker and generated a through crack. It was concluded that the failures were owing to a combination of operating condition and the quality of some welds.

Two main precautions were taken to prevent reoccurrence of the fault:

- At the design stage by specifying a maximum excess thickness of weld beads;
- During operation by introducing a procedure of purging and preheating the steam pipes.

The corrective actions concern after SGs leak:

1983: Time gain in hydrogen detection signal processing

- Introduction of double insulation of the water supply pipe to the evaporator stages;
- Addition of a second nitrogen injection system to each SG stages;
- Addition of a second decompression valve to the evaporator stages;
- Systematic sodium dump of the affected circuit five minutes after the trip. This delay corresponds to the minimum time needed to homogenize the sodium and to avoid local sodium hydroxide concentrations;
- Dryout of the other SG five minutes after the dryout of the damaged SG. This operation prevents any steam ingress into the damaged circuit in case of leaking isolation valves.

1984: Processing of the hydrogen detection signals by two dedicated computers auctioning the rapid shutdown and the isolation and dryout of the SG.

1985: Installation of the GENEVA monitoring system. As well as the permanent signal monitoring this computer allows all the routine tests to be carried out (e.g. calibration by injection of hydrogen, control of the diffusion rate); development of an experimental acoustic detection system.

1988: Installation of an induction heater in circuit No. l evaporator stage; the modular SG design facilitated the location of the faulty module and minimized the outage while taking a module out of service.

The cost of four sodium-water reaction incidents in steam generators was a total of six months of reactor shutdown and nine months of reactor restricted operations at two-thirds power.

A fifth sodium-water reaction in the reheater of the SG no. 1, module no. 12 took place on 13 September 2003 during operation on power level of ~ 300 MW(th): about two kg of water came into contact with the sodium. As in the previous cases, the failure was located at the upper bend of a reheater module and were due to a combination of operating condition and an initial manufacturing defect in the weld. A wastage in the shell across the tube hole was observed (Fig. 6a,b). The damage module was replaced with a new one.

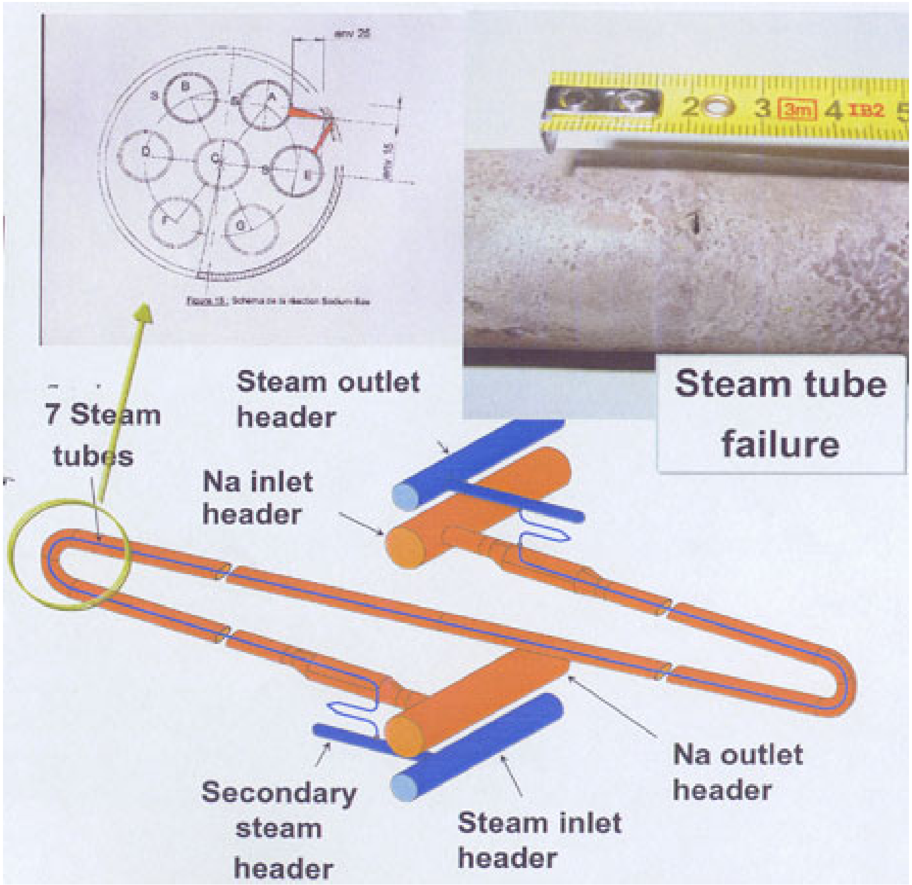


FIG. 6a. Phénix SG tube failure, - a fifth sodium-water reaction (September 2003).

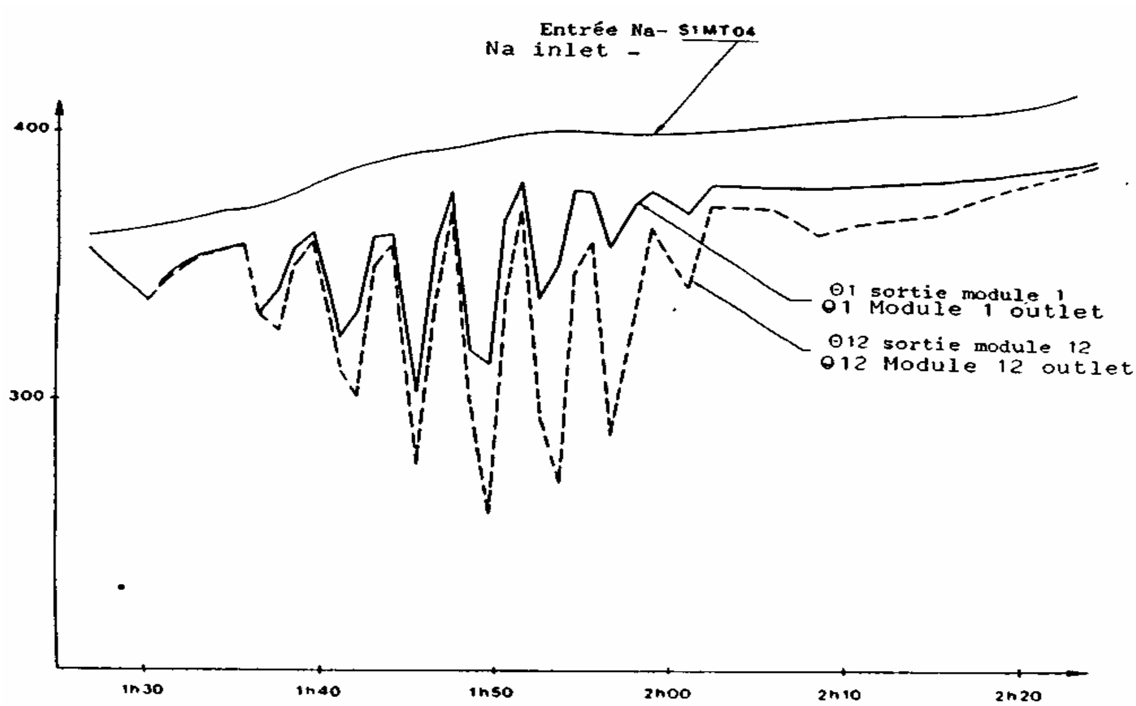


FIG. 6b. Phénix SG: sodium outlet temperature curves for reheater modules 1 and 2.

3.1.4. Secondary circuit operating experience

Each of three independent secondary cooling systems, which transfers heat from the primary system to the steam generator and isolates the steam supply from the activated primary sodium, composing two intermediate heat exchangers, pump, buffer and expansion tanks, auxiliary circuits and connecting pipe work (outer diameter × wal1 thickness: 510×6.0 mm; material: hot leg-stainless steel 321, cold leg-stainless steel 304). The secondary sodium flows inside the intermediate heat exchanger tubes and outside the steam generator tubes.

Minor secondary sodium leakages were detected three times, in September 1974, and then in March and July 1975. Leakage was traced to a joining weld on a large diameter pipe delivering sodium to the reheater part with 450 mm control butterfly valve. Approximately 20 L of sodium was lost on the first two events and only about one L the second events with no sodium fire outside. Repairs were ineffective and the valves in all three systems were replaced by diaphragms.

In March 1986, a leak occurred on the secondary circuit buffer tank. The repair of the tank needed a difficult operation because the sodium aerosols had corroded the tank wall that required welding a new metal on the spherical tank wall. Later three tanks were replaced by new ones 316 steel.

On 5 May 1986 during full power operation, a leak occurred from one of the main pipes of secondary circuit. It appeared at the entrance of the main pipe carrying sodium into the superheater part of the steam generator and was detected by the leak detection system. Around 50 kg of sodium leaked and froze in the thermal insulation casing surrounding the pipe. No local signs of leakage were visible. Regular patrolling was stepped up in the area until the next scheduled shutdown on 19 May 1986.

The circuit was drained and the insulation removed from the pipe. Several tens of kilograms of mixture of sodified sodium and insulation material were found. The leak was located in a weld seam joining the 500 mm diameter tee fitting to the inlet header in the steam generator unit. The tee fitting consists of two AISI 321 half shell castings 19 mm thick, while the AISI 321 sleeve is 7 mm thick.

Relatively marked corrosion was observed on the wall of the tee fitting, especially around the perimeter of the solidified sodium mass. Corrosion damage locally reached several millimetres. Dye penetrant, gamma radiography and metallographic examinations were conducted in-situ prior to removal of the tee fitting. The results indicated that the crack ~ 120 mm ran along the weld seam in the heat-affected zone on the inlet sleeve side of the steam generator. The intergranular crack extended on either side of the upper generatrix, and measured 120 mm in length. After removal of the tee fitting, the following metallurgical examinations were carried out:

- Macrographic examination of the inner and outer cracked surfaces;
- Micrographic examination in planes perpendicular to the weld seam at different positions in the crack;
- Fractographic examination of a 2 mm thick cross section through the weld seam;
- Ferrite and carbon content analysis.

In addition, the area of the tee fitting that had been corroded by the sodium-insulation mixture was dimensionally checked by an ultrasonic thickness measurement device. The results suggested that the cause of the leak was a delayed reheat cracking mechanism liable to appear in Ti stabilized austenitic stainless steels submitted to mechanical stresses[3].

Concerning the external corrosion of the tee fitting, dimensional measurements indicated that the zone in contact with sodium covered about 1 m^2. The most highly corroded area was around the perimeter of the contact zone, and the maximum corrosion depth reached 7.4 mm out of an initial thickness of 19 mm. The age of the leak was estimated at a few thousand hours. The tee-piece was replaced with a new one specially made. The corresponding parts of secondary loops 1 and 2 were also examined, but no defects were found in the tee fittings. The plant was operated at two-thirds of rated power with only two secondary loops until the end of August 1986, pending delivery of a replacement tee fitting. The tee fitting was replaced. The main secondary pipe leak detection system was improved by continuous monitoring of the electrical isolation of the preheating “pyrotenax”.

In the event of a sodium leak the heating element was generally damaged by corrosion products. Monitoring the electrical isolation of the two pyrotenax units provides an additional means of detection. The units are located on the pipe midplane: in case of leakage from the upper portion of the weld, they are affected more quickly than the leak detection wire along the lower generatrix. An operating error caused a sodium leak in October 1988.

The Phénix plant, like the other sodium cooled reactors, had significant experience with sodium leaks. Since the facility had started up, there had been some twenty leaks, approximately one per year of power operation: the leaks in IHX and SG- the most damaging caused the largest production loses. The aim of leak before break (LBB) argument in liquid metal cooled fast reactors safety analysis is to show significant lines of defence which allows rejecting in the residual risk an unacceptable of main vessels and piping. The important conclusion which could be derived from the Phénix secondary circuit leaks is as follows: the applicability of LBB argument should be confirmed in terms of stress analysis, risk of corrosion, leak detection efficiency.

3.1.5. Pumps operating experience

At Phénix, vibrations were detected on one primary pump and were shown to be due to a design defect which allowed the hydrostatic bearing bush to become separated from the shaft as a result of expansion during thermal transients. Modifications were made to four primary pumps including the spare. A similar failure has occurred in 1987 on a secondary pump. Table 4 summarizes the Phénix sodium pump performance.

[3] “This is a defect triggered in the root pass of a weld in the immediate neighbourhood of the contact zone, and which spreads radially between the grains when under stress in service. This only occurs with certain materials such as 321 type stainless steel, as it caused by hardening of the steel due to tine precipitations of titanium carbide inside the crystalline structure. The hardening causes the plastic deformation capacity to be transferred to the periphery of the grains. For this to happen, the following conditions have to be met:

- a high operating temperature (> 475°C for 321 steel);
- a geometrical discontinuity at the weld root;
- strain hardening at the weld root showing significant shrinkage;
- heavy local load, which may be due to welding stresses; and
- a defect in the weld root (e.g. a small shrinkage crack)” *[Jean-Francois Savage, Phenix 30 years of history: the heart of reactor, CEA/EDF, Valrno-BP 17171-30207 Bagnols-sur-Ceze cedex], p.220.*

TABLE 4. PHENIX SODIUM PUMP PERFORMANCE

Operating hours		Failures
Primary 472 830	Secondary 390 200	Three-primary, one-secondary

3.1.6. Negative reactivity shutdowns

On three occasions in summer 1989, the reactor was stopped by automatic emergency shutdown, the negative reactivity threshold (-10 pcm) being exceeded. This reactivity variation was very fast: first a minimum after 50 ms followed by an increasing oscillation, and then a decrease, caused by the control rod drop, 200 ms after the start of the transient.

The first two events were thought to be spurious (a neutronic chamber fault) and the reactor was restarted. The normal plant instrumentation did not allow proper recording of the transient so following the second trip special instrumentation was installed. After the third trip, the reactor was shut down in order to identify the cause of the events. It was found that the phenomenon could be explained by gas entrainment through the core, after accumulation under the diagrid.

The void coefficient explained the transient loss of reactivity. After some reactor improvements related to this explanation, the reactor was allowed to restart at the end of 1989, but the event occurred again in September 1990, after 182 EFPD of operation.

Investigation of the negative reactivity shutdowns which had occurred since the reactor’s commissioning showed that two trips, in April 1976 and in June 1978, are similar to those in 1989 and 1990.

An expert committee was then set up, and an extensive study of all the possible phenomena was started. Also it was decided to fit the plant with special monitoring equipment including fast recording systems', and to perform tests. Around 200 data were concerned. Tests on vessel and component mock-up were also planned. The tests were performed with the reactor shut down, critical at zero power (since October 1991) and at 350 MW(th) power (for around 12 days - February 1993). In the same time, checks were performed on the plant, especially on the reactor, its components and auxiliaries. Reactor tests were very satisfactory. They proved the good behaviour of the instrumentation, and data are now stored as reference of steady state power and emergency shutdown conditions.

By the end of 1993, studies had not led to a clear explanation of the phenomenon: “false” reactivity variations (a “neutronic mask” between core and neutronic chambers, or a spurious signal) are thought to be impossible. Among "real" reactivity variations, a sodium void effect or variation of the relative displacement of fuel and control rods are also thought to be impossible. There remains only a radial core volume variation, the origin of which (a “pressure wave”) has not been found. These studies confirm that the reactor safety was not affected. Operation with special instrumentation seems to be the only way to understand the origin of these negative reactivity trips.

It was concluded that the phenomenon was harmless and could have had at least two possible causes: signal interference or mechanical movement by the subassemblies in the core.

3.1.7. Sodium aerosol deposit

Phénix has two types of control rod mechanisms (this principle being adopted in order to minimize the probability of common mode failures). One type of mechanism, which is equipped with bellows protecting it against aerosols, has never experienced any problems in 17 years of operation.

However, during the test and renovation time a leak in the bellows of the two control rod mechanisms was detected. This bellows located between the sleeve and the rod ensure the sealing between the sodium or the argon of the reactor cover gas and the argon inside the mechanism, so as to limit the presence of aerosols inside the mechanism.

The other mechanism has experienced two series of problems, both linked with sodium aerosols, although neither of these problems prevented the control rods from being normally inserted when required to shut down the reactor.

The first series of problems occurred in January 1974, a few months after the first criticality. Following a normal scram, two rods were found 15 mm above their lower position.

After extraction of the mechanisms, it was discovered that the cause of the blockage was the fresh argon flow of 600 L/h which was permanently being entered into the annular space between the rod and the sleeve (oddly enough in order to prevent any upward motion of eventual sodium aerosols in this annular space). The argon outlet in the hot pool was under sodium. So the argon flow had two consequences:

(a) Depression of the sodium level in the annular space between extension rod and sleeve; and
(b) With the argon flow oxygen impurities were transported and space was higher than before, returning to the overall level inside the reactor; thus all the moving part of the rod, going though the lower bearing, would be permanently bathed in the hot sodium.

The three mechanisms (of the second type) were cleaned and remounted and no other problems occurred until 1981. In 1981, one of the same types of mechanism showed new anomalies in its functioning, which were first detected during systematic measuring of the drop time of each rod. One rod exhibited longer than normal drop time (100 ms).

A series of tests was initiated. At first it was thought that the dash-pot bearing the whole mechanism at the end of the course was responsible, until it was realized that the cause was some friction force acting during the stroke of the rod. After dismantling it was observed that some aerosols had been deposited along the extension rod, being formed near the “labyrinth” with a clearance of 2 mm in diameter and spread over the length of the tube during the raising and lowering of the mechanism over seven years of operation (Fig. 7).

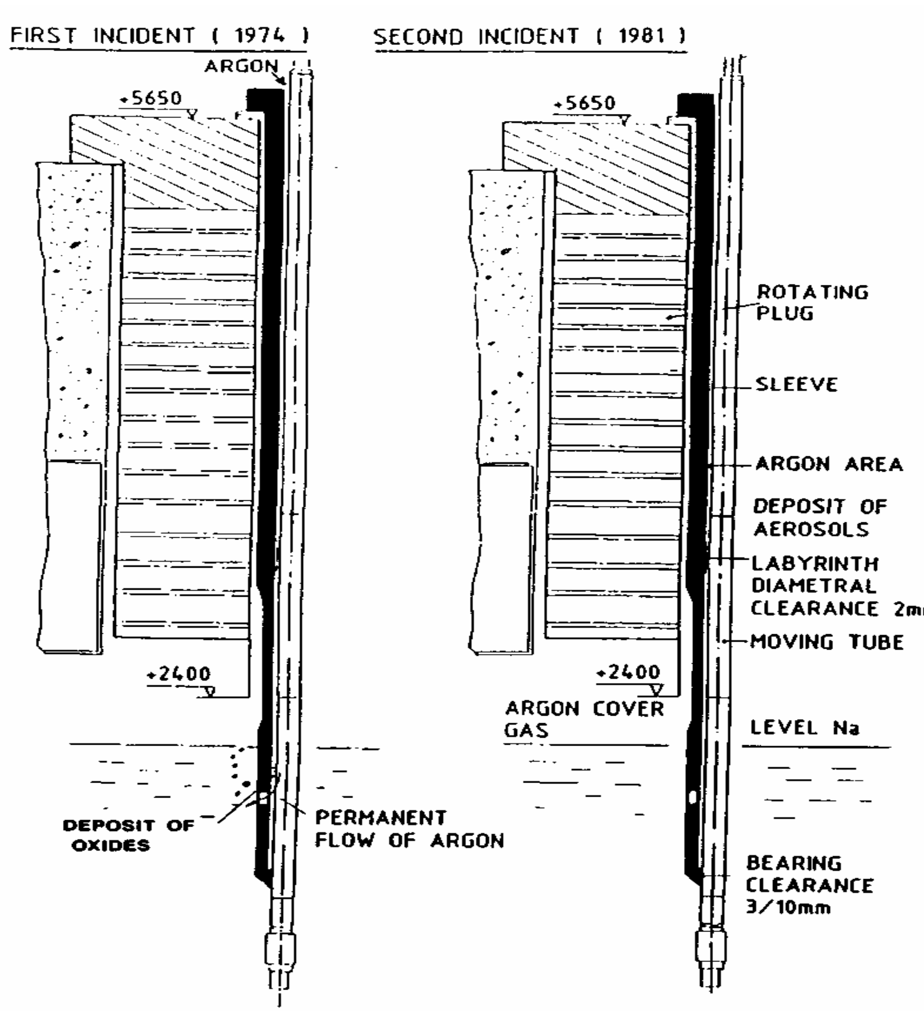


FIG. 7. Phénix: deposit of oxides on control rod mechanism.

This explains why the problem occurred progressively. It was decided this time to install on each mechanism a permanent device to measure with great accuracy the force needed to exercise the control rod.

3.1.8. Sodium circuits, reactor and equipment inspection and renovation

The Phénix reactor had been shutdown in September 1990 owing to the negative reactivity trips. Some times the reactor was in transient conditions at low power to maintain staff competence in reactor operation. Then the reactor was often kept in a state which was ready to start up. By this time the plant had accumulated about 100 000 hours, whereas it was designed for 140 000 hours of power operations. The operator proposed to start the plant back up. But each new request added a new safety issues. The delayed reheat cracks on the secondary circuit were some of them. Several objectives motivated the additional control, repair and manufacturing new components. At Phénix, renovation works as well as inspection, maintenance and repair activities have been done during 1993-2003 (J. Guidez and L Martin: Int. Conf. Fifty years of nuclear power- the next fifty years, Moscow/Obninsk, Russian Federation, 27 June-2July 2004):

— Special inspections of the welds of the secondary piping system, steam generator, reactor core support structures and the upper internal structures of the reactor block;
— The repair of the secondary piping system and steam generator;

- The portioning of the secondary sodium circuit in the steam generator to improve protection against large sodium fires;
- The seismic reinforcement of the plant building;
- The additional of safety control rod to the reactor;
- The installation of two redundant seismic resistant emergency cooling circuit and anti-wip system on the high pressure pipes;
- The defective equipments were repaired, replaced or left as is and justified by a non-propagation analysis for the defects, which backed up the non-destructive testing done on the site.

3.1.8.1. The secondary piping system

At the end of 1993, since the beginning of the inspection campaign in 1990, 1 200 meters of welds have been inspected, 4 000 gammagraphy films have been produced and interpreted. All circular welds, longitudinal welds and elbows, and 10% of the longitudinal welds of the straight parts were inspected by gammagraphy, due penetrant test, and where was possible by ultrasonic techniques. The pipes made of 304 steel were the first to be examined, for they were the longest and parts of them had been located inside the reactor and steam generator buildings. Very few small, non evolving manufacturing defects were revealed, and some ten welds were repaired.

In 1992 two cracks were observed on a 304 steel pipe downstream from the hydrogen detection return due to thermal stripping associated with the temperature fluctuations. The hot leg pipes around the steam generator up to including the buffer tanks, steam generator sodium headers and modules were made of 321 steel stabilized by titanium to improve the mechanical strength at high temperature.

From the analysis of these inspections and the metallography expertise of the dismounted welds, the following conclusions were drawn: 20 circular 321 stainless steel piping welds were found to contain cracks, in a total of 60 which were dismantled and 220 which were inspected; the cracks were located in the straight parts of elbows and tee-junction with thickness variations. Cracks were observed on 12% of the welds operating at 550°C and 3% of the welds operating at 475°C.

Non-destructive control revealed that cracks emerged at the root of the weld of the thermally affected zone. An ultrasound indicator detected the crack, but did not determine precisely the depth of the defect. Related expert studies showed that it was difficult to guarantee that crack under the operating stresses would not grow to a depth up to a thin wall of the pipe, which in Phénix is a 6 mm thick. The piping and SG portions of the 321 type steel were replaced by new 316 steel, which have been successfully used at SPX.

3.1.8.2. Steam generators

During testing of the reheater in SG 2, a leak appeared on the connection weld of five modules on the steam inlet side. This fault occurred in a well known manner and may be explained as follows: the welding operation between the steam collector, made of chromesco1 ferritic steel, and each of the reheater modules, made of AISI 321 austenitic steel stabilized with titanium, initiated microcracks in the weld seam by hot cracking.

In 1982 during a sodium-water reaction, sodium penetrated into the steam part of the modules subsequent to the malfunctioning of a check valve. This sodium was not completely eliminated during cleaning, especially in the microcracks that were present.

Inspection revealed cracks in some of the secondary circuit components made of 321 stainless steel, and conclusions repairs have been made. Several defects of the relaxation cracking type were displayed, which led to systematic replacement of such secondary piping and the SG sodium headers. A significant crack was detected following the examination of a SG no. 2 module. Studies for repair were undertaken in early 2001. Repairs were made on the 321 steel superheaters and reheaters (48 units in total) of SG no. 1 and no. 3. They involved the sodium inlet cones and the first hot bend in each module (Fig. 8) and control of the cold bend.

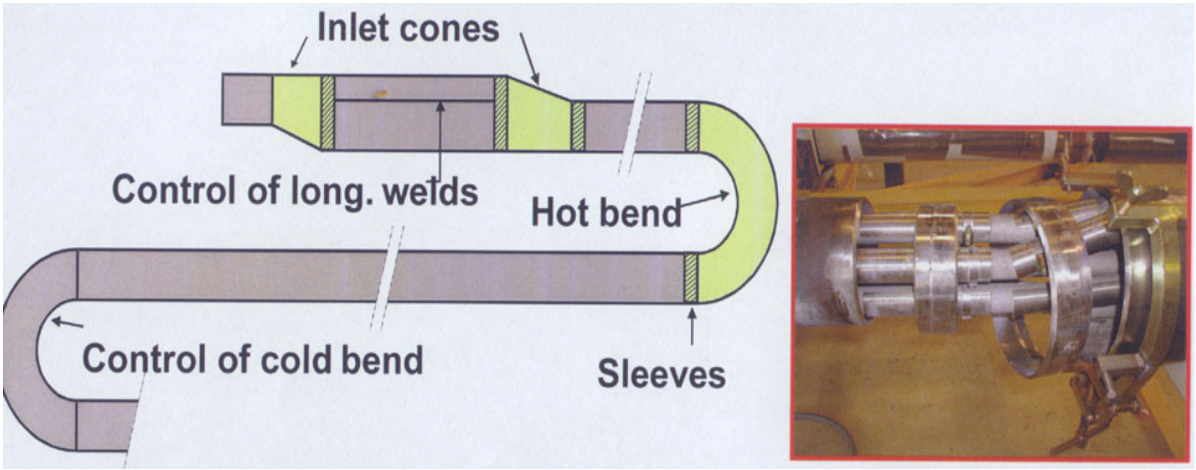


FIG. 8. Phénix steam generator repair and control.

The technological operations were as follows:

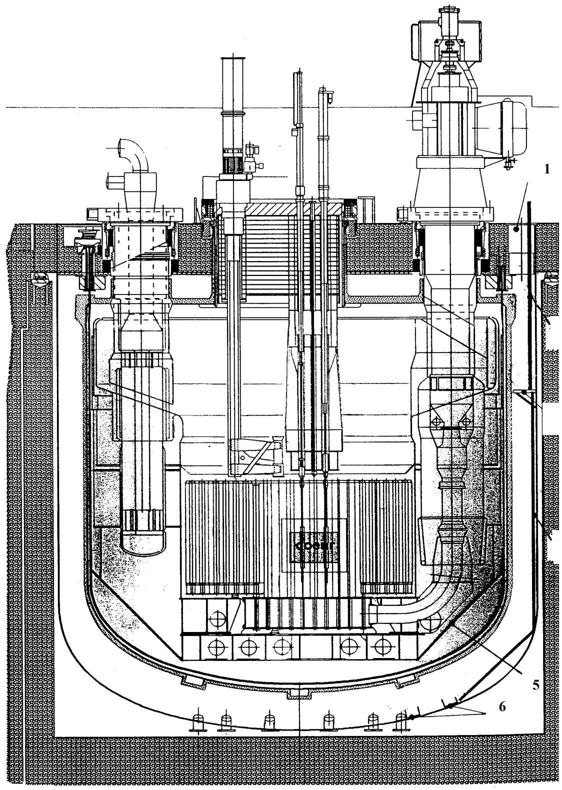
— Disassembly of the modules;
— Cleaning for residual sodium removal;
— Repair and control of some non replaced parts;
— Inspection of the new welds;
— Leak resistance and hydraulic tests for each module;
— Reassembly in the steam generator.

3.1.8.3. Reactor block

Four non-destructive tests have been carried out at the Phénix reactor:

— Ultrasonic inspection of the core support conical skirt (pos. 5 in Fig. 9a) welds under liquid sodium at ~ 150°C in period September-October 1999. Five holes specially were provided for these operations. The work was automated owing to a hot and irradiation environment.
— Inspection of the top portion and the handling shells of the main reactor vessel using small transducers.
— Inspection of the tubeplate in IHX using a photothermal camera.

Remote visual inspection (by periscopes) of the core cover plug and internals structures of the reactor in March-April 2001 with the partial (420 m^3 of ~ 900 m^3) draining of the primary sodium to the level of the fuel subassembly heads.



1-manhole (Ǿ = 500 mm)
2-movable ladder
3-catwalk
4-fixed ladder
5-conical skirt
6-cable tracks

FIG. 9a

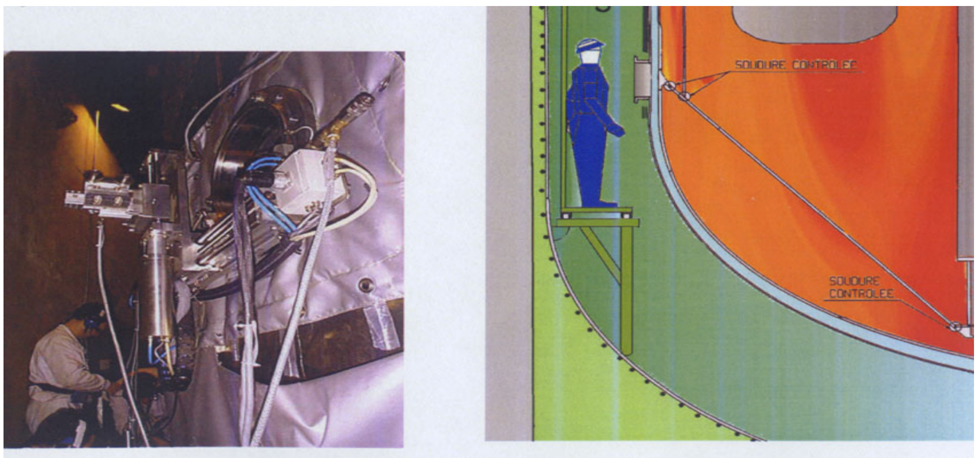


FIG. 9b

FIG. 9a,b. Phénix reactor conical skirt welds US inspection.

3.1.9. Protection for fires

During the test and renovation new studies and significant works on protection of the steam generator building against large sodium fires have been undertaken. The latter include (J Guidez and L Martin: Int. Conf. Fifty years of nuclear power-the next fifty years, Moscow/Obninsk, Russian Federation, 27 June-2 July 2004):

— The separation by steel insulated walls and doors firebreak of zones so as to limit the spreading of a large fire; two separation steam generator cells reconstructed (Fig. 10) to resist a major sodium fire with temperature on order 1 000°C for 30 minutes;
— The portioning or the housing of cable trays and building steel structures;
— The ventilation and the smoke cleaning circuits;
— Installation of multi-sampling detection circuit.

Most of the buildings were modified: offices and the control room, the handling, the reactor, the SG and auxiliary circuits buildings. A vast modernization programme at the Phénix NPP has been undertaken.

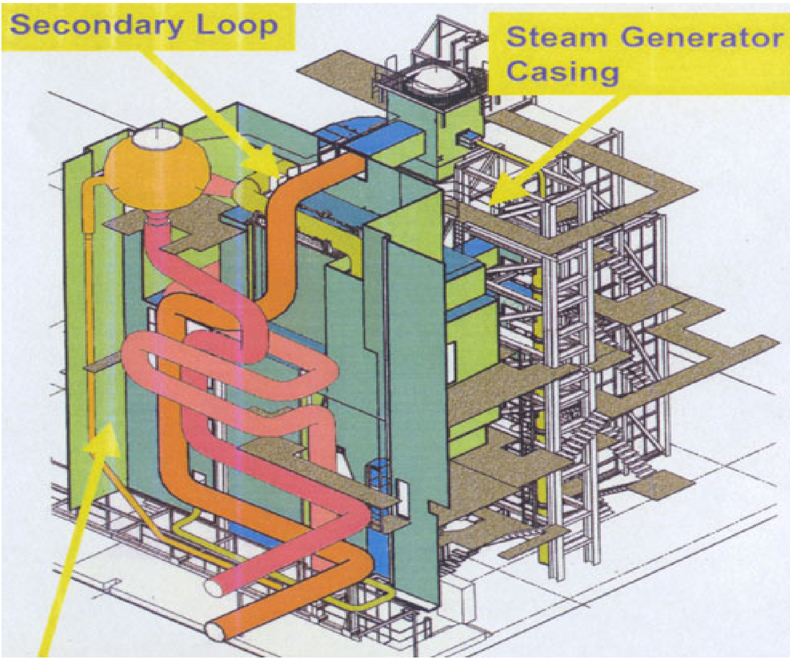


FIG. 10. Protection of the Phénix steam generator building against large sodium fires.

3.1.10. Plant statistics and conclusions

The French prototype fast breeder reactor until 1990 operated with a comparatively high operational characteristics and demonstrated the industrial feasibility of sodium cooled reactor technology (Figs 11 and 12).

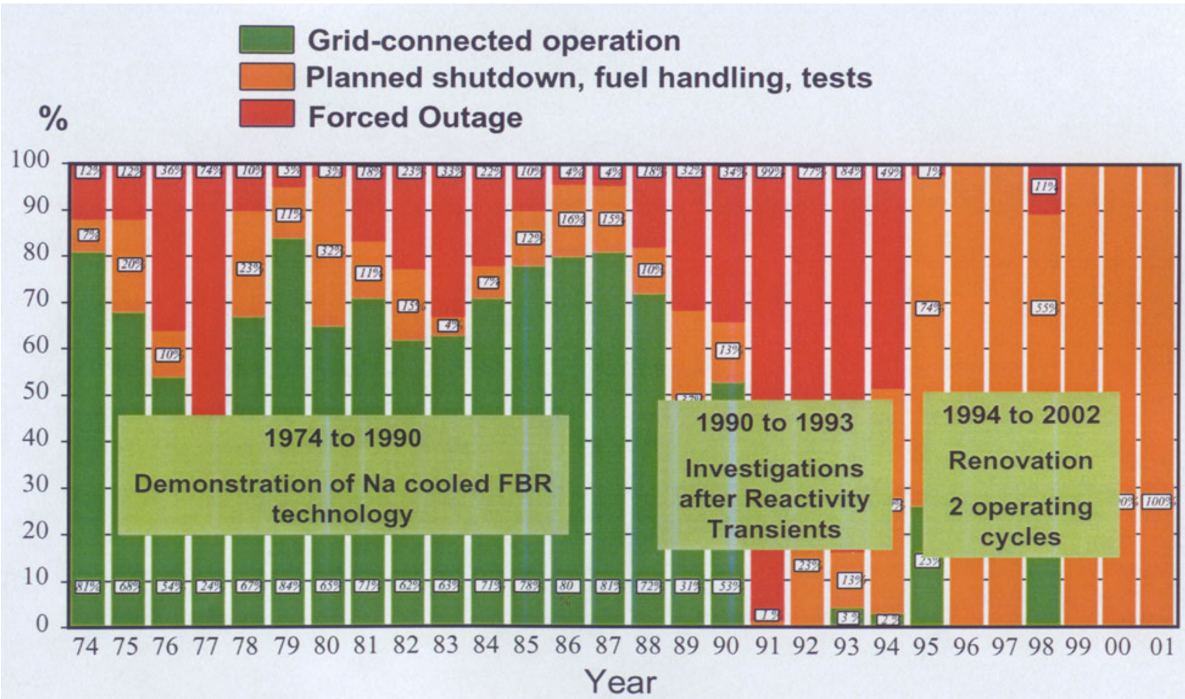


FIG. 11. Phénix operating histogram.

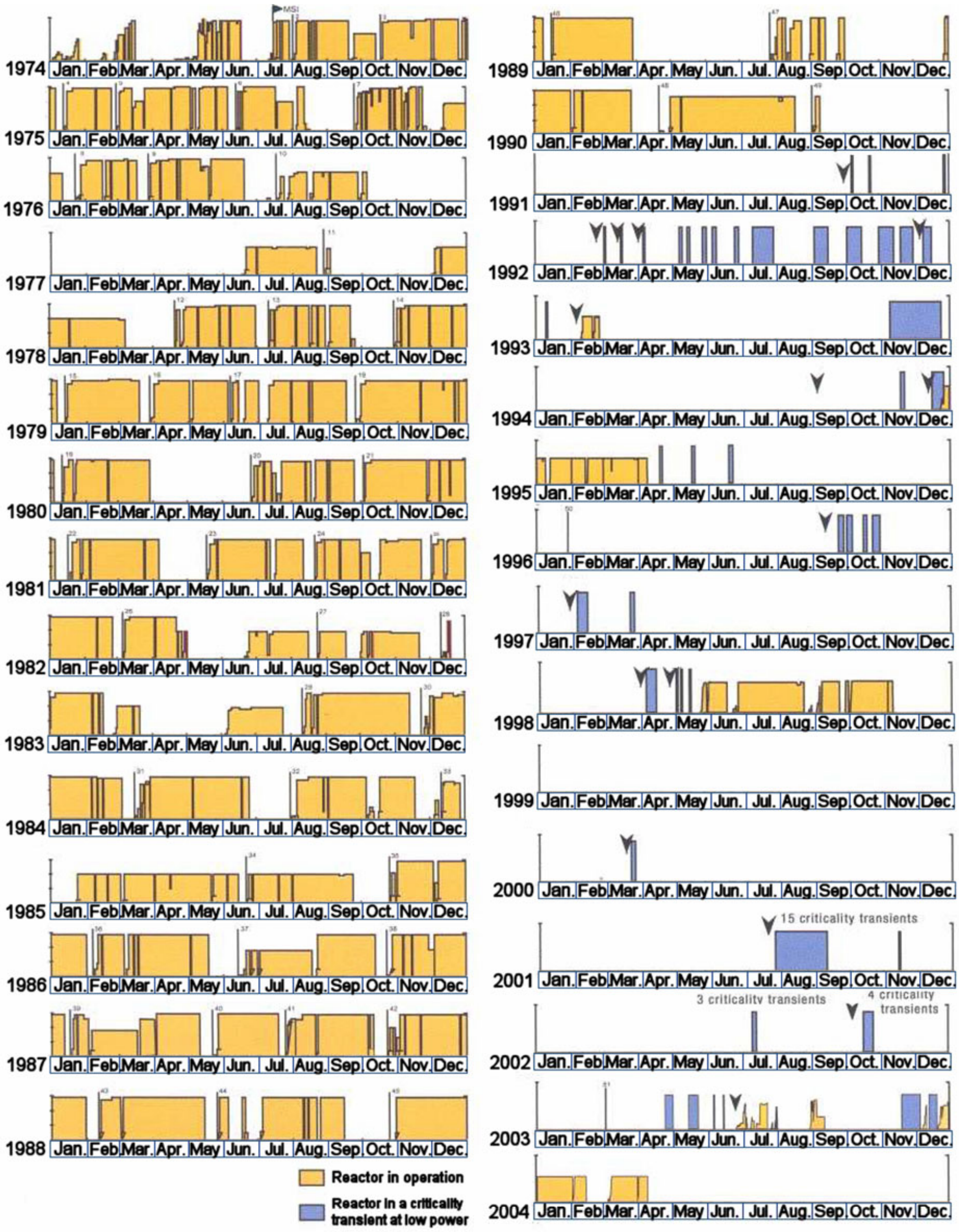


FIG. 12. Phénix power plant-operational chart 1974-2004.

The main production data of Phénix are shown in Table 5.

TABLE 5. PHENIX MAIN PRODUCTION DATA

Characteristic	Value
Effective full power days, EFPD	3 900 EFPD
Gross electrical energy production, GWh	22 424 087
Load factor (since commissioning in July 1974), %	~ 50
Number of irradiated subassemblies	829
Number of irradiated pins	166, 521
Burnup (maximal), h.a.	17%

In the period 1974-1989 the plant was connected to the grid ~ 70% of the time, 25% of which was at reduced power. On the average the periods of stabilized operations is equal to 40% owing to the long periods operations at two-thirds of the rated power (following the events encountered on the SGs, IHXs and secondary sodium pipe leaks).

Many judgments were made during the design of Phénix NPP with a much more limited data base than is available to those viewing the plant retrospectively. With the exception of the choice of 321 steel stabilized by titanium to improve the mechanical strength at high temperature and IHX design, these judgments are generally vindicated by expert evaluations conducted on the materials and the reactor components, after 100 000 hours of operation with projected modern parameters. In the future sodium cooled reactor designs the using of the 321 steel stabilized by titanium are probably best avoided.

The Phénix plant, like the other sodium cooled reactors, had significant experience with sodium leaks. Since the facility had started up, there had been more than twenty leaks, approximately one per year of power operation.

The methodology developed by team of designers, researchers and operational staff to extend reactor lifetime, the development and realization of special investigation/inspection and renovation programs have resulted in significant progress for R&D and greatly increased expertise from which the entire nuclear program will benefit. Phénix contributions continue as it provides the first experience in the transmutation of long-lived nuclear waste and the utilization of surplus plutonium in the real reactor conditions.

The Phénix reactor has been restarted in June 2003 after five years of being off line for a major inspection, repair and safety upgrade. The cost of the upgrade work (~ 3 million hours-men) was 250 million EUR. The Phénix reactor will be used to irradiate actinide transmutation experimental rods. There are 12 transmutation projects to be carried out between 2003 and the scheduled closure date of the reactor in 2009[4].

[4] Nucl. Eng. Int., August 2003, p. 6.

3.2. SUPER-PHENIX REACTOR

3.2.1. Design features and commissioning history

The Super-Phénix (SPX) plant was derived from Phénix is of a pool type. Changes were either necessitated by the increased size of the reactor, or they were to achieve definite in economic and safety performance. For instance:

— The primary sodium coolant purification units were within the main vessel;
— There were four helical steam generators (SGs), each with a power of 750 MW(th);
— Slightly larger subassemblies, designed to achieve a higher burnup;
— Design of the main vessel and roof were simplified;
— A dome over the upper part of the vessel was added to provide containment.

Main design and economical data are presented Tables 6 and 7, respectively. The cutaway view of the SPX reactor is shown in Fig. 13. The comparison of the SPX-1200 with the 1400 MW(e) PWR plants Paluel-1 and -2, built at the same period, showed that the cost per unit power installed of SPX approximately a factor of 2.7 higher than a PWR (Table 6)[5].

TABLE 6. SPX SPECIFICATIONS

Item	Value
Power (thermal/electric), MW(th)/ MW(e)	3000/1200
Thermal efficiency, %	40
Inner diameter/height of main vessel, m	21/19.5
No. of loop (prim./sec.)	4/4
No. of main pump (prim./sec.)	4/4
No. of IHXs	8
Sodium inventory (prim./sec.), tons	3 500/1 500
Sodium flow rate (prim./sec.), t/s	4×4.24/4×3.27
Primary sodium temp. (hot leg/cold leg), °C	545/395
Secondary sodium temp. (hot leg/cold leg), °C	525/345
Loop concept	classical
Steam temp. at turbine inlet, °C	487
Steam press. at turbine inlet, bar	177
Steam flow rate, kg/s	4×340
Feed water temperature, °C	237
Type of steam-water cycle	Steam reheating system
No. of SG per loop	1 once-through SG
Total mass, tons	194

[5] M. RAPIN, Fast breeder reactor economics, presented in the Royal Society Meeting on the Fast-neutron-breeder fission reactor, London, U.K., 24-25 May 1989; R. CARLE, Detailed design studies demonstrate major improvements in economics. Nucl. Eng. Int., February 1988.

TABLE 7. COST POWER GENERATION BY SPX AND PWR- P'4, centimes/kWh [1]

Cost category	PWR-P'4 [1 400MW(e)]		SPX-1 [1 200 MW(e)]
Capital investment	8.5		22.6
Fuel	5.3		10.0
Operating and maintenance (O&M)	3.2		5.0
Power generation costs	17.0		37.6
Capital investment SPX/PWR-P'4		2.66	
Power generation costs SPX-1/PWR-P'4		2.222	

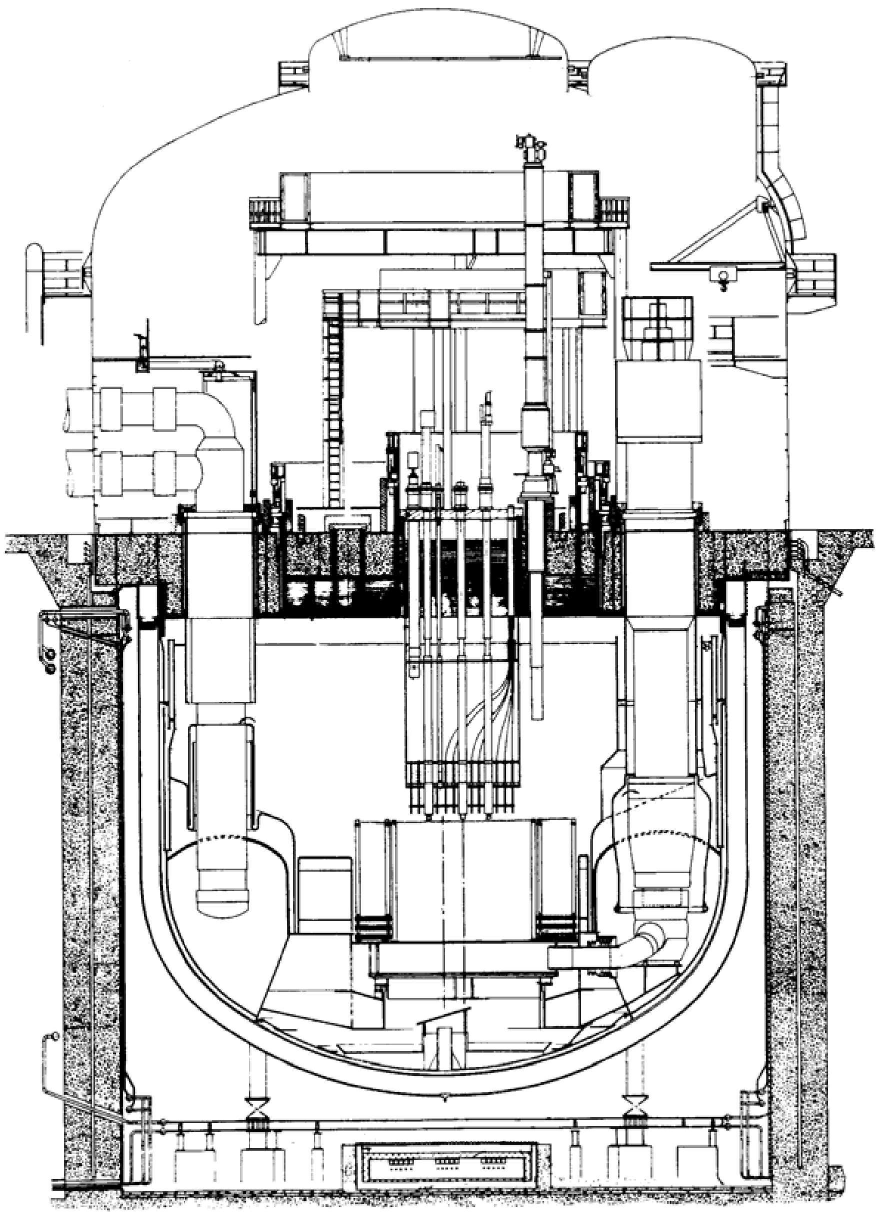


FIG. 13. SPX: cross-section of the primary circuits.

That is why, when designing the next European LMFR EFR, the problem of choosing an optimum type of arrangement, reactor and equipment design was raised again. The primary goal was clearly to cut the costs on the basis of the design and construction for SPX. The important step in this direction was the power rise from 1 200 to 1 500 MW(e).

Primary sodium coolant is entirely enclosed in the main stainless steel vessel which contains the core, and in which are installed four primary pumps and eight intermediate heat exchangers. The main vessel is closed above the free level of sodium and argon cover gas by the slab which contains in its central section two eccentric rotating plugs and the core cover plug which supports the control rod drive mechanisms and the core instrumentation. It is surrounded by the safety vessel, welded to the slab, which is itself topped by a metallic dome.

This dome can resist a pressure of 3 bar at 180°C. The safety vessel and the dome make up the primary boundary, and the reactor building in reinforced concrete constitutes the secondary boundary. The main design data are given in Table 3.

Successive sodium filling of the storage drum, two secondary loops, the reactor block and lastly the two remaining secondary loops, took from June to December 1984. Filling the reactor block took 2 months, from 23 August to 31 October; after prior heating to about 150°C by circulation of hot nitrogen. These filling operations were preceded by supply and on-site storage of 5 650 tons of sodium, transported by 291 tankers.

Isothermal tests started in January 1985 with an initial build-up in temperature limited to 395°C following the appearance of a phenomenon of vibration of the thermal shield in the main vessel during increased primary pump speed vibration of part of the SPX reactor internal structure occurred at certain flow rates, caused by small waves on the sodium surface exciting one of its resonant modes.

These vibrations were investigated using the MIR (visual examination by camera and displacement measurements through the "MIR" reactor inspection robot) vehicle for in-service inspection in the interspace between main and safety vessels, by adapting the focused ultrasonic transducers normally intended for the inspection of the main vessel weldments and by mock-up under water, rapidly attributed this phenomenon to a hydrodynamic coupling of the fluid and the structures, caused by vessel coolant sodium flow through the spillway.

Hydraulic studies showed that the sodium level at the main vessel cooling weir was a critical factor. By changing the flow adjusting gags in some 20 fuel subassemblies and thus increasing the flow rate over the weir, it was possible to reduce the fall over the weir, taking vibrations into the quiescent region for all normal operating conditions.

The first subassembly was loaded on 20 July and first criticality was achieved on 7 September 1985, with a core made up of 325 fissile subassemblies, as planned. In October 1985, 33 fuel subassemblies were added to constitute the core for power build-up; it was with this core that zero power testing was carried out at no more than 30 MW(th): control rod worth, neutron flux distribution and fission rates, calibration of neutron channels, measurements of reactivity and feedback coefficients. The "first nuclear" steam was received on 31st December 1985. The SPX plant was connected to the grid on 14 January 1986, full power was reached on 9 December 1986 and operated, but not without difficulty. Upon successful start in January 1986, SPX reactor operation was interrupted by incidents (the operating and administrative history of SPX is illustrated in Fig. 14), rated at level 1-2 of INES.

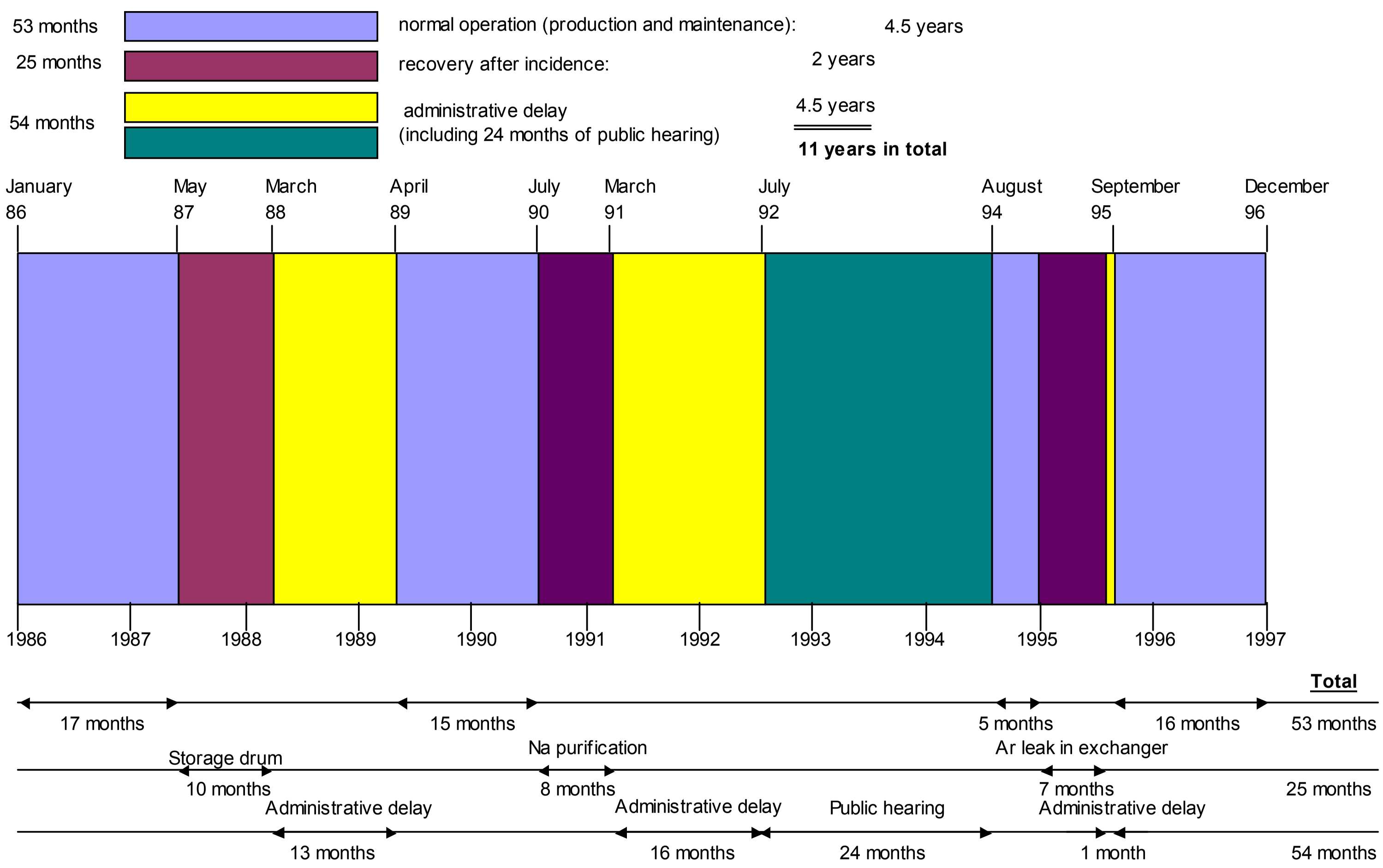


FIG. 14. The operating and administrative history of SPX.

During 11 years of SPX NPP existence, it was in operation for 4.5 years (energy production, correction of malfunctions, and design modifications). Special repair procedures for elimination of consequences of three incidents took 2 years; and administrative and social procedures concerning license issue for reactor start-up took 4.5 years, because of the nature of the accidents, as well as a very strong anti-nuclear social movement in the post Chernobyl period, and a sodium fire at the Almeria facility that coincided with reactor start-up tests and power increase. During 4 cycles (4.5 years), the SPX NPP produced 7.9 billion kWh (half of this energy was produced in 1996), i.e. a reactor was operated with load factor about 20%; 101 events and incidents have occurred at the reactor plant from the start of fuel loading (July 1985) till its closing down (July 1998). Out of these, 66 did not concern fast reactor specific features and, among 35 incidents concerning LMFR, 16 were related to sodium. Seven events were rated at level 1, and two events were rated at level 2 by the INES.

Three incidents marred the commissioning procedure and caused lengthy delays. In March 1987 there was a sodium leak from the used fuel storage drum. Two years after the incident plant was granted the permit to resume operation and reached its nominal power on 16 June, 1989. In June-July 1990 there was an air leak into an auxiliary circuit which caused extensive contamination of the primary sodium.

The plant has been technically ready to be restarted at the end of June 1991. Partly as a result of this experience the safety of the plant was thoroughly reviewed and modifications to improve the response to secondary sodium fires were made.

A public enquiry on renewal of the operating license was held and reported positively, and in 1994 the plant was granted the permit to restart. This was further delayed by a small leak of argon from the sealing bell surrounding one of the intermediate heat exchangers. This justified shutdown of the reactor in December 1994. Corrective action was taken and in the latter part of 1995 and through 1996 power was gradually raised to the full-power level. Since 24 December 1996 the reactor scheduled shutdown followed by legal cancellation of its operation license. It was made very clear that the reason of the shutdown was in no way with safety problems, but with economy: the government said that when uranium appears now durably cheap there is no need today to operate an industrial fast reactor prototype which “cost more than expected”.

The longest outages of the SPX-1 reactor were caused not only by and not because of using sodium as a coolant. Sodium bears relation neither to the fall of the turbine hall roof caused by snow load with extensive damage to the steam plant, nor to the vibration of water/steam piping system resulted from the water hammer. There were also some other incidents in the BOP, which have been reported at the international meetings. The frequency of the events was 7.8 per year over the entire period considered. As noted in the overall frequency of the events is similar to that observed on the PWR reactors, i.e. around 8 events declared per year, covering all after commercial start-up [reported at the IAEA meeting on Unusual occurrences during LMFR operation, 9-13 November 1998]. A large social response on any incident existed because in the time of SPX commissioning anti-nuclear sentiments raised by Chernobyl in some social groups and governmental structures were too strong. With downturns in energy demand and political changes in 1998 the French Government finally confirmed to discontinue it operation.

As a whole, the final operating experience of SPX-1 was incomplete but not as negative as sometimes reported: over eleven years of existence it has been operating during four and half years producing 7.9 billion kWh (half in 1996). Experience feedback on large components remains significant in spite of the short operating period. Primary and secondary pumps total more than 60 000 hours on main motor, and the continuous improvement of maintenance operations has allowed an increase in reliability and availability. As far as the steam generators are concerned, the sodium/water reaction detection systems have been improved on the basis of validated calculation codes through experience. Numerous draining and filling operations (more than 30 for the secondary loops and more than 20 for the decay heat removal emergency circuits) have allowed validation of the corresponding procedures. Knowledge of primary circuit behaviour has in fact been improved thanks to natural convection tests which showed that it was established in the core in about 5 minutes.

3.2.2. SPX steam generator: design and operating experience

The biggest change in SPX design was undoubtedly that to the SG. Of all the fast reactor design features this seems was the one that was the least well established and that showed the biggest difference in style and unit size between the various designs proposed by the major fast reactor countries.

The equipment of a large LMFR with economical means requires the use of SGs with high self power. The only experience in the world was SPX where helical alloy 800 units with 750 MW(th) power were installed and very successfully operated. Each 750 MW steam generator (SG) was a once-through unit comprising a vertical tube bundle, whose tubes were wound helically around a cylindrical support, and an outer cylindrical shell, penetrated by inlet and outlet thermal sleeves for each tube. This system avoids thick tubular plates (and associated problems) and resists the thermal shock during sodium transients. Table 8 shows the main design and Fig. 15 shows a cutaway view of SPX SGs, respectively.

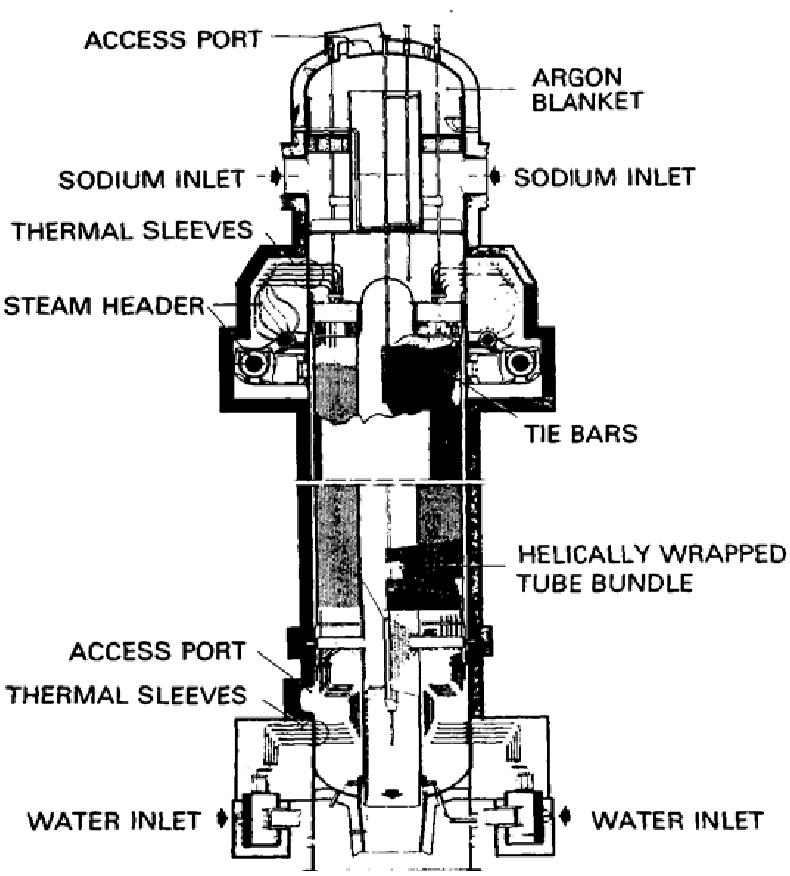


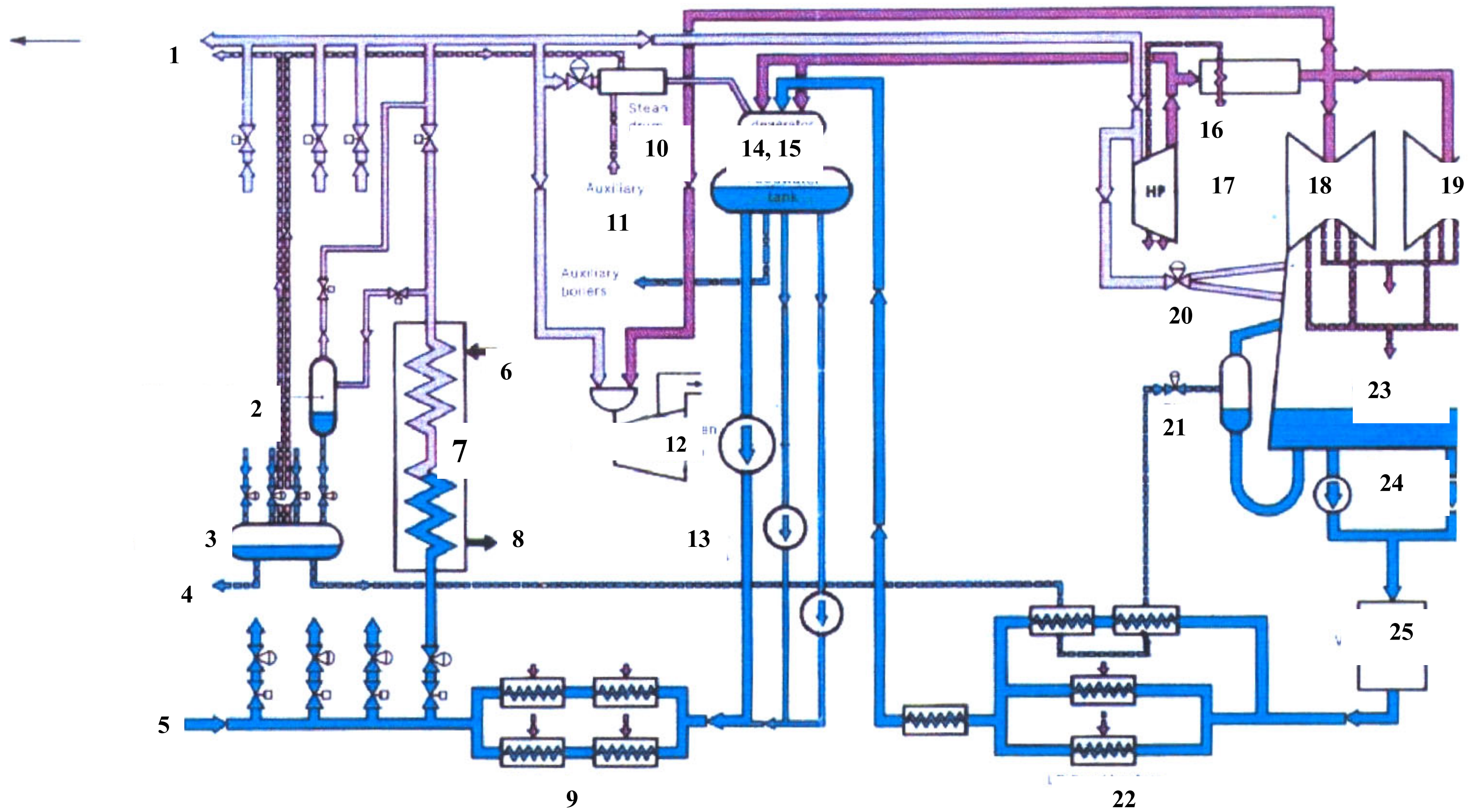
FIG. 15. SPX steam generator.

TABLE 8. SPECIFICATION OF SPX SG

Item	Value
Type of SG unit	Once-through
Thermal power, MW	750
Total length, m	22.4
Shell outside diameter, m	2.88
Shell material	Z 3 CND 18.12
No. of heat transfer tube, per unit	357
Length of heat transfer tubes, m	92
No. of welding on one tube	7
Out. diameter and wall thickness of tube, mm	25/2.6
Material of tube	Incolloy 800
Heat transfer area, m^2 / SG	2.565
No. of tube layers	17
Diameter of inner tube layer, m	1.17
Diameter of outer most tube layer, m	2.61
Steam/water inlet temperature, °C	237
Steam/water outlet temperature, °C	490
Steam water flow rate, kg/s	339.6
Steam/water outlet pressure, bar	184
Steam/water pressure loss, bar	349
Sodium inlet temperature, °C	525
Sodium outlet temperature, °C	345
Sodium flow rate, t/s	3.273

The SPX steam generators were of a new advanced design. Whereas the BN-600, Phénix and PFR steam generators comprise a number of small (modular solution), and divided into three sections - evaporator, superheater and reheator – this solution was not adopted for the SPX plant. Here, for further simplification, the evaporator and superheater were combined in a single unit. There was no sodium reheat, because the need for SG reliability increase and cost reduction by means of decrease of the number of units and sodium/steam heat transfer surface made it necessary using steam as the heating medium (extracted from the HP turbine, Fig. 16).

The advantages of the large diameter helical design are extremely good space utilization and good heat transfer properties. The tubes joined together in a manifold outside the steam generator vessel. This has the advantage that none of the walls containing sodium, apart from the tubes themselves, and certainly none of the sodium-containing welds, carries a high primary stress. Some disadvantages are the comparative complexity, and therefore cost, of the manifolds to connect several hundred tubes, and the difficulty of gaining access to individual tubes for inspection or repair and provision that has to be made for inspection during operation. In some SG designs (e.g. BN-600 and PFR) it was possible to avoid welds in the length of the tubes, where inspection is particularly difficult, but with helical tubes such welds are unavoidable.



1-turbine; 2-start up tank; 3-flash tank; 4, 5-water treatment plant; 6, 8-sodium inlet/outlet; 7-steam generator; 9, 22-HP, LP feed heaters; 10, 11-steam drum, auxiliary boiler; 12, 13-feed water pumps (12-turb. Driven); 14, 15-deaerator-feed water; 16-reheator/moisture separator; 17, 18, 19-HP, LP1, LP2 turbine; 20-bypass; 21-blowdown tank; 23-condenser; 24, 25-extraction pumps, water treatment

FIG. 16. SPX steam-water flow scheme.

As known, in the earlier years of the development of fast reactors there was a high incidence of leaks in the steam generators. Since leaks in sodium-heated steam generators are intolerable, however small they may be, whenever a leak occurs the unit in question has to be shut down and isolated for repair. This could be done without serious diminution of the power output only if there are a large number of small power separate units, any one of which can be isolated without reducing the total power significantly. The outstanding example of the advantages of the modular approach to steam generator design is afforded by the French Phénix and Russian BN-600 plants. This has three secondary sodium circuits, each with 12 (Phénix) and 8 (BN-600) separate steam generator modules, and each of these consists of separate evaporator, superheater and reheater sections, making a total of 108 (Phénix) and 72 (BN-600) separate heat exchangers. Partly because of this the availability of the above mentioned plants has been consistently high. However, these operational advantages are achieved at the expense of higher metal content and costs, and of complication of the SG layout. The metal content and cost of the high self power SG are reduced significantly by applying the modular approach to steam generator design (Table 9). It should be pointed out that some of leaks occurred where the tubes are attached to the tubeplate (BN-600 modules) and where the tube are welded to tube (Phénix SG- on the bends).

TABLE 9. SGs COMPARISON

Item	Phénix SG	BN-600 SG	SPX SG
Power (thermal/electric), MWth/MWe	186.3/85	490/200	750/300
Mass of SG, tons	225	600	194
Specific metal content, tons/MW(e)	2.64	3.0	0.65
Number of welds per 1 MW(e)	16. 0 (tube to tube)	27 (tubes to tube- plate attachments)	8.3 (tube to tube)

Each of the SPX tubes was ~ 92 m long. There were seven welds per tube, so SPX-1 had about 10 000 welds in comparison with ~ 4 000 in Phénix. The flow tubes were butt welded by the TIG (tungsten inert gas) process with welding metal. The operation was automatic and 100% radiographic control was carried out. A 45 MW SG prototype was built and had undergone very extensive operating tests for several years at the EdF test center. The welding and weld quality control procedures were developed and checked at this SG. That is why the SPX SGs were operated successfully; past errors were not repeated and good design solutions were incorporated. The outstanding success of the SPX operation has undoubtedly been the demonstration of reliable operation of SGs with high self power (750 MW(th)).

3.2.3. Storage drum operating experience

3.2.3.1. Description and role

The SPX fuel handling equipment was designed for a maximum residual power of about 28 kW per subassembly (Figs 17–19). In the initial design of fuel handling operations, the fuel storage drum played a double role:

— It ensured the transfer of all new and spent fuel subassemblies during reactor loading and unloading operations;
— It ensured the storage in sodium of spent fuel subassemblies to allow the decay of their residual power from 28 to 7.5 kW before cleaning and storage in the on-site water pool.

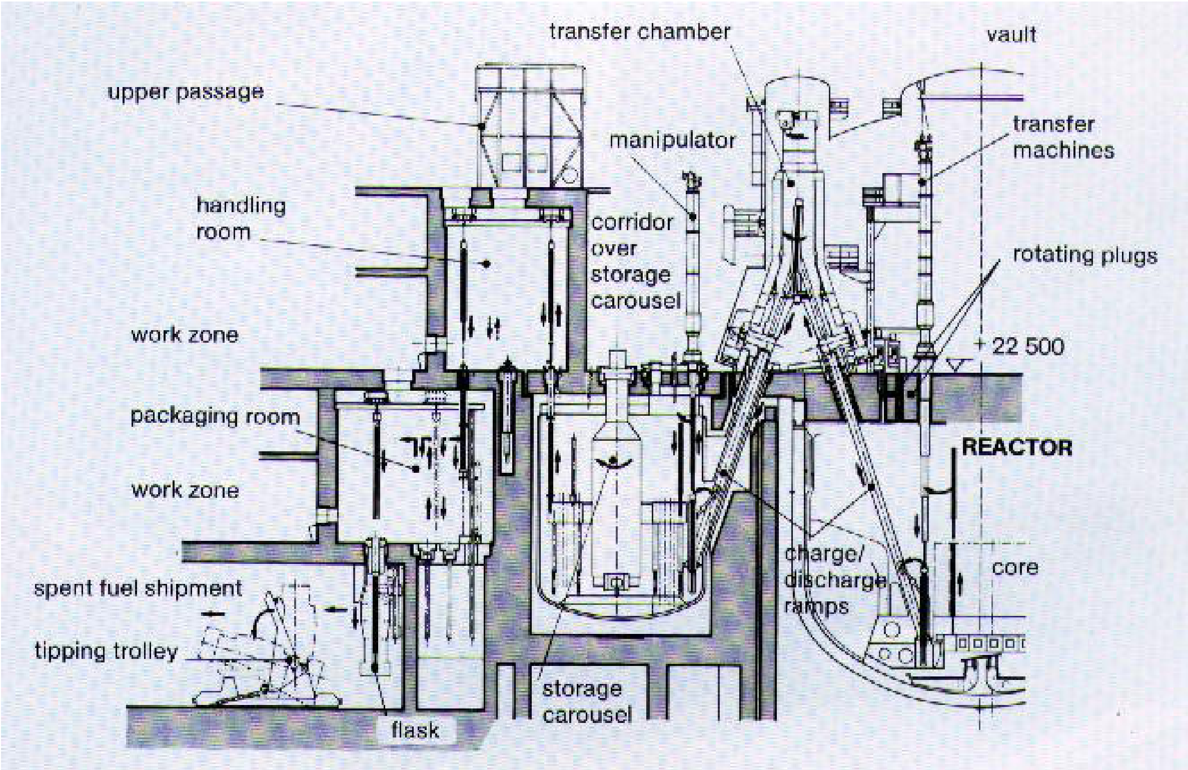


FIG. 17a

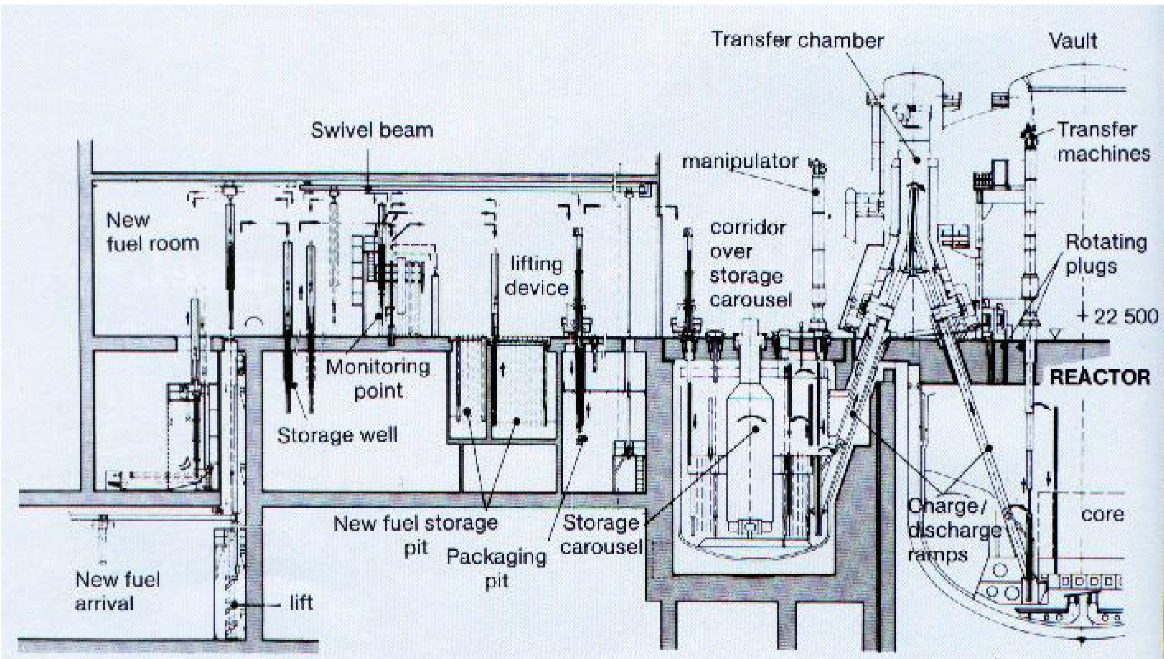


FIG. 17b

FIG. 17a,b. SPX new and spent fuel handling systems.

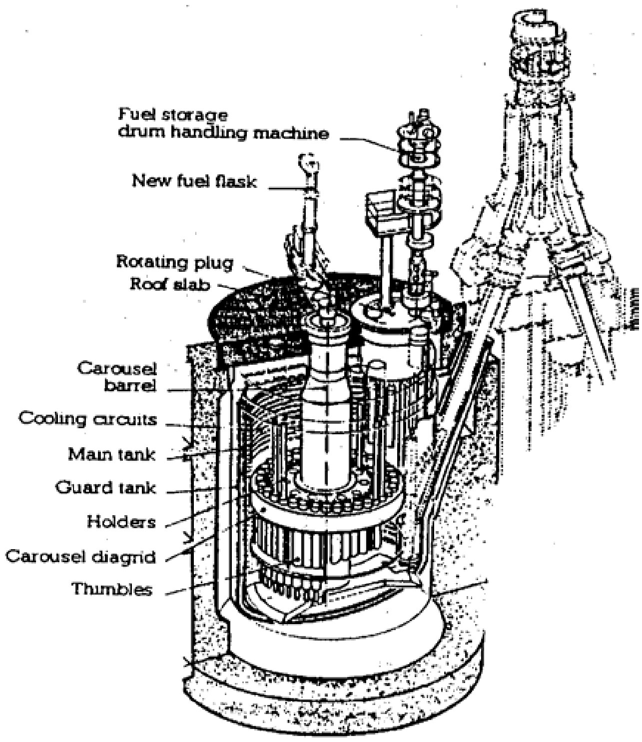


FIG. 18

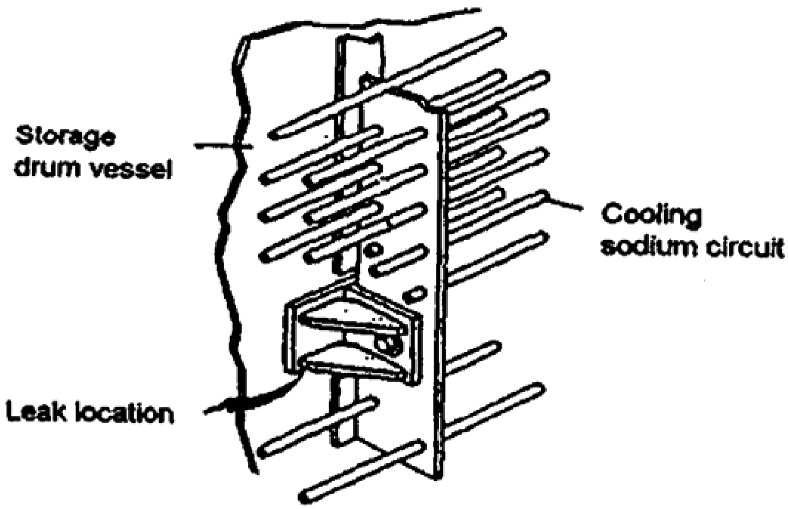


FIG. 19

FIGs 18, 19. SPX storage drum and leak location.

The drum played the role of a storage buffer between the reactor and the water storage pool; it allowed reducing loading and unloading time to a minimum, and thus the duration of reactor outages. The storage drum is part of the fuel handling line.

To facilitate handling of the fuel between the storage positions and to reach the required capacity of fuel assemblies, two-level storage was provided; the maximum residual power from fuel assemblies stored was 3 MW, which was removed through two redundant sodium loops (Figs 17 and 18).

The storage drum consists of a 9 m diameter and 13 m high vessel made of 15 D3 ferritic steel with a low molybdenum alloy. It contained about 800 m^3 (700 tons) of sodium at a nominal temperature of 200°C. This sodium was cooled or heated by sodium circuits circulating in a bundle of tubes fixed on the inner side of the vessel. The storage drum is equipped with a safety vessel. Both vessels were separated from each other by a 150 mm gap, and were located in a 2 m thick concrete pit.

In the initial plant project, it was envisaged to remove the fuel subassemblies to a reprocessing plant after a period of one year in the sodium-filled storage drum. It was also planned to renew 50% of the core after a half-cycle of 320 EFPD. The storage drum could receive 409 subassemblies with a decay heat of up to 28 kW. Removal of the subassemblies was carried out when the decay heat was less than 7.5 kW.

3.2.3.2. The incident

On 8 March 1987 a leak was detected in the space between the storage drum vessels by the leak detection system. Several methods were implemented to confirm the leak, and assess its importance. The balances for the levels of the storage drum and the storage tank showed that there were 20 m^3 of sodium between the vessels.

The leak was confirmed and the first investigations to locate the leak and size of the crack started. This work took place between 27 August and 9 September 1987. Infrared thermograph, xenon and helium detection in the inter-vessel space were used. The localization of the leak had been carried out during the fall of the sodium level as a result of emptying the storage drum in August 1987. The leak was found to be along one of the lower support-plates for the storage drum cooling circuit (Fig. 19).

At the time of the leak, the storage drum contained only virtually radiation-free subassemblies whose decay heat was negligible. The most serious risk, although very improbable, was, in the short term, a leak on the safety vessel made of the same material as the storage drum vessel, not knowing the cause of the leak. The unloading operations of the storage drum were finished on 31 July 1987. The fuel and fertile sub-assemblies were then, after approval, put back in the reactor, and the slightly irradiated dummy subassemblies were at this stage stored in special containers constructed for the purpose.

After emptying the sodium completely, and when the storage drum had cooled down, investigations consisting in particular of X-rays through the two drums showed up that other plates were affected by the same type of not-yet-penetrating fault. It became obvious that local repair would be impossible, and that it was necessary to anticipate human access into the drum. This meant an air-fill of the main drum and its cleaning.

3.2.3.3. Causes of the accident

The leak was caused by a horizontal crack around 60 cm long on the lower angle welding bead which secured a plate. After the drum sodium drainage and the first investigations, identical faults to those observed on the plate at the origin of the leak were found on similar plates. Numerous other observations made in-situ showed that cracks of similar type, but nevertheless less severe, existed not only in the vicinity of the plate-supports but also on the structure welds of the drum.

The reuse of the initial sodium drum after repair proved to be impossible and it was necessary to define replacement. When the drum was removed, it was found out that long (several meters) cracks had also formed in the constituting weld beads of the main vessel. Considerable effort was expended on researching into the cause of the fault, in order to determine the follow-up necessary (reuse or not of the other facilities in 15 D3 ferritic carbon steel grade), and to assess possible risk on other structures or components.

The destructive examination samples taken at the beginning of 1988 showed that cracking was very probably due to:

- The existence of start sites (micro-cracking) in zones of high hardness;
- Embrittlement by hydrogen, and
- The cracks developed disruptive zones under the influence of residual welding stresses close to the elastic limit of the material.

3.2.3.4. Dispositions chosen for installation repair

After weighing up a considerable number of solutions to solve the problem posed by the occurrence of the leak, it appeared from October 1987 that the choices were between two categories of solutions:

- The first solution consisted of reconstructing the two geometrically very similar drums, but in austenitic steel;
- The second solution consisted of abandoning the decay storage function of the drum (this function being therefore assured in the reactor itself), and maintaining only the function of subassembly transfer.

It was the last solution that was finally chosen, in March 1988, particularly because it allowed personnel to resume more rapidly operation in normal conditions. The new device was called the “fuel transfer station” (PTC). The implementation of this solution required dismantling the fuel storage drum. This was started by the conversion of the residual sodium into sodium carbonate through a controlled additional of water vapor and carbon dioxide gas.

This operation (began in August 1988) allowed placing the vessel in contact with atmosphere and were performed with pressure suits. So as to ensure compatibility of site-work with plant operation, a drum work zone (zone de travail barilla, ZTB) was created to isolate the repair area from the rest of the reactor building, not only from the point of view of ventilation but also detection, protection, fire risk, health physics, and handling operations. In July 1989, the dismantling was over.

The schedule for the realization of the PTC shows that it will be available at the beginning of 1992: the in-situ assembling of the new stainless steel chamber started in December 1989.

Reactor operation was not possible without a fuel subassembly discharge route during the PTC construction. Consequently a special flask was designed to transfer the subassemblies directly from the reactor to the fuel cave.

In 1982 the necessity for fast reactors in France was less acute and therefore there was less need for a dedicated reprocessing centre. This led NERSA to decide on the on-site construction of temporary storage for several spent cores. This was referred to as the fuel storage pool building (APEC). Then in 1988, when repair of the storage drum turned out to be impossible, NERSA chose to eliminate it altogether and replace it with a gas-filled transfer chamber.

This modification, which was made possible through the existence of the fuel storage pool building, in turn led to modification of the management mode of the core which was based on frequency one. The core was renewed entirely after a cycle of 2 to 3 years (640 EFPD). Replacing the subassemblies at a later stage required 7 to 8 months delay, including an initial period of 2 months, for decay of the first subassemblies to a level of 7.5 kW.

The APEC and the PTC make up two links of the handling line. Construction of the APEC covered the period 1984 to 1989, and that of the PTC lasted two years (1990/91). The PTC was a facility that implements a fuel removal process with no intermediate storage between the reactor core and the water filled pool. This facility was very different from the ferritic steel fuel storage drum.

The subassemblies were placed by the transfer machine in a sodium-filled container which is carried by the pot in the A-frame. In the argon-filled fuel transfer station under argon, the container carrying the subassembly is placed on a pivot arm which transfers it to the handling line. The pivot arm has a third position to receive the new subassemblies to be loaded.

After washing, the subassemblies are placed in a transfer shuttle which can receive three subassemblies. The APEC offers storage for approximately 1 700 subassemblies between the pool with a capacity of about 1 400 subassemblies and a hall which can house about 300 subassemblies in casks under gas atmosphere. Unloading capacity of the handling line is about four to five subassemblies per day.

3.2.4. Primary sodium contamination

After plant shutdown to permute diluents (7 September 1989 to April 1990) and shutdown for works as a result of detection of leak on the feed water purification circuit (28 April 1990 to 31 May 1990), the reactor was moved into critical mode and reached its nominal power on 11 June 1990. During all this period up to the temperature build-up (12 June 20% nominal power), sodium clean lines were monitored by the operating team on recording delivered by the plugging indicators using the usual operation methods: plugging temperature, general slope of the curve. Monitoring confirmed that there was a good level of cleanliness; in particular, the 110°C level lasted around six hours.

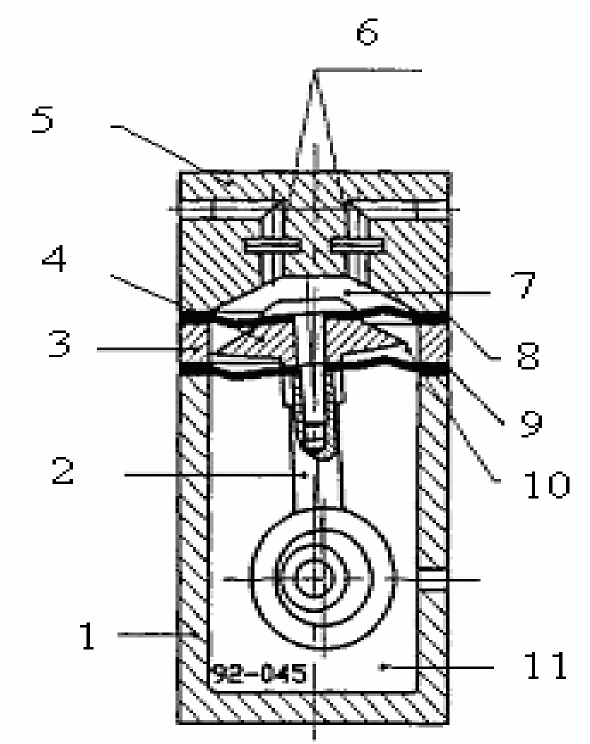
That phase of operation did not last very long. Already during startup, elevated plugging temperatures were found in the primary sodium system and continued to rise excessively. In normal operation, the plugging temperature must be under 120°C. The technical specification required: between 120 and 150°C to shut down within one month, above 150°C to shut down immediately. The criteria were established to limit the corrosion rate and to avoid impurity crystallization during fuel transfers and maintenance outages at 180°C.

On 10 June, during power escalation, the plugging temperature raised and stabilized at 140°C. This corresponded to a small and normal contamination of the sodium caused by maintenance during the outage. Normally this is a temporary situation and the plugging temperature decreases after a while. On 20 June, a second inflexion point appeared on the flow curve. This point corresponded to a plugging temperature of 180°C.

Indeed, an accurate analysis of the flow curve highlighted several inflexion points corresponding to several plugging temperatures. These plugging temperatures can be indicative of different types of impurities contained in the sodium. According to the chemists, the plugging temperature of 180°C could be disregarded because it was giving indications of hydride (and not oxide) that “would be quickly eliminated through the stainless steel intermediate heat exchanger tubes to the secondary system thanks to hydrogen permeability”. It was then decided to take into account the plugging temperature of 140°C only, and so to remain in power.

On 26 June, one of the two filter cartridges of the integrated primary purification system clogged. On 30 June, the second filter cartridge also clogged. As the purification system was unavailable, the unit was planned to be shut down on 3 July to replace the saturated cartridges, and indeed shutdown occurred automatically due to an electrical defect. After reactor fast shutdown, it was maintained isothermal at 250°C, ~ 40°C above the oxygen saturation temperature in the primary sodium when the rate of impurities was maximum (15 ppm oxygen) to avoid crystallization in the coldest parts of the reactor. Meanwhile, investigations were carried out to identify the origin of this contamination.

The argon taken from the primary circuit is then channeled by a compressor through a measurement chamber (Fig. 20).



1-compressor bode
2-main rod
3-intermediate ring
4-top of connecting rod
5-head
6-closures
7-compression chamber
8-neoprene working diagram
9-neoprene safety diagram
10-safety chamber
11-control unit

FIG. 20. Compressor of the SPX gas system.

Investigations of the reasons indicated that the membrane of one compressor had been defective for a long time already, and the argon cover gas therefore had been exposed to atmospheric air. About 600 L/h of an argon and air mixture entered the reactor argon circuit downstream of this circulation pump.

There was no system to measure the purity of the argon cover to provide an alarm for the operator in case of contamination of this neutral gas. The amount of sodium oxide produced was estimated to amount to 350–400 kg. Subsequent cleaning of the contaminated sodium took several months and was managed by the in-plant cold traps.

To remove these impurities personnel operated with the integrated primary cold traps, and the two new cartridges operated clogged in about 4 weeks retaining about 100 kg of oxide. After temperature increase in the primary circuit in order to dissolve residual oxides on the sodium surface and on cold metallic parts, we were free of sodium purification at the end of January of 1991 without clogging the second pair of new cartridges. The safety valve on the primary argon circuit, which was also polluted with oxides have also been replaced.

The other corrective actions concern:

— Review of the argon circuit design to avoid air intakes, and in particular setting up instrumentation on the compressors for detecting membrane failures;
— Installation of a chromatograph for periodic monitoring of the argon cover gas;
— Review of the technical specifications in order to give more accurate criteria, based on the duration of low temperature on the flat on the plugging temperature recorder and, on the unplugging temperature more representative of the saturation one;
— Estimation of induced risks of corrosion, plugging and aerosols.

Besides, this incident gave the opportunity of further knowledge on the chemistry of sodium (new tests on the Cadarache loop), on the behaviour of integrated cold traps and on the workings of plugging indicators.

3.2.5. Argon leak from the sealing bell of the intermediate heat exchangers

The argon supply of the bell gas cavity was carefully monitored by the operator. A significant increase in the frequency of bell re-inflation on one IHX was observed in December 1994.

The leak was plugged by installing a metal sleeve applied and held in position by permanent deformation (expansion by passing hydraulic pressure) using two crimping on either side of the leak on the inner side of the pipe (pipe co-expanded with the sleeve). The equipment used to install the sleeve was then left in the piping, and was subject periodic monitoring. Corrective action was taken and in the latter part of 1995 the plant was ready to operate.

The seal between the intermediate heat exchangers (IHX) and the inner vessel for the SPX was provided by a gas (argon) seal – sealing bell surrounding the IHX (Fig. 21).

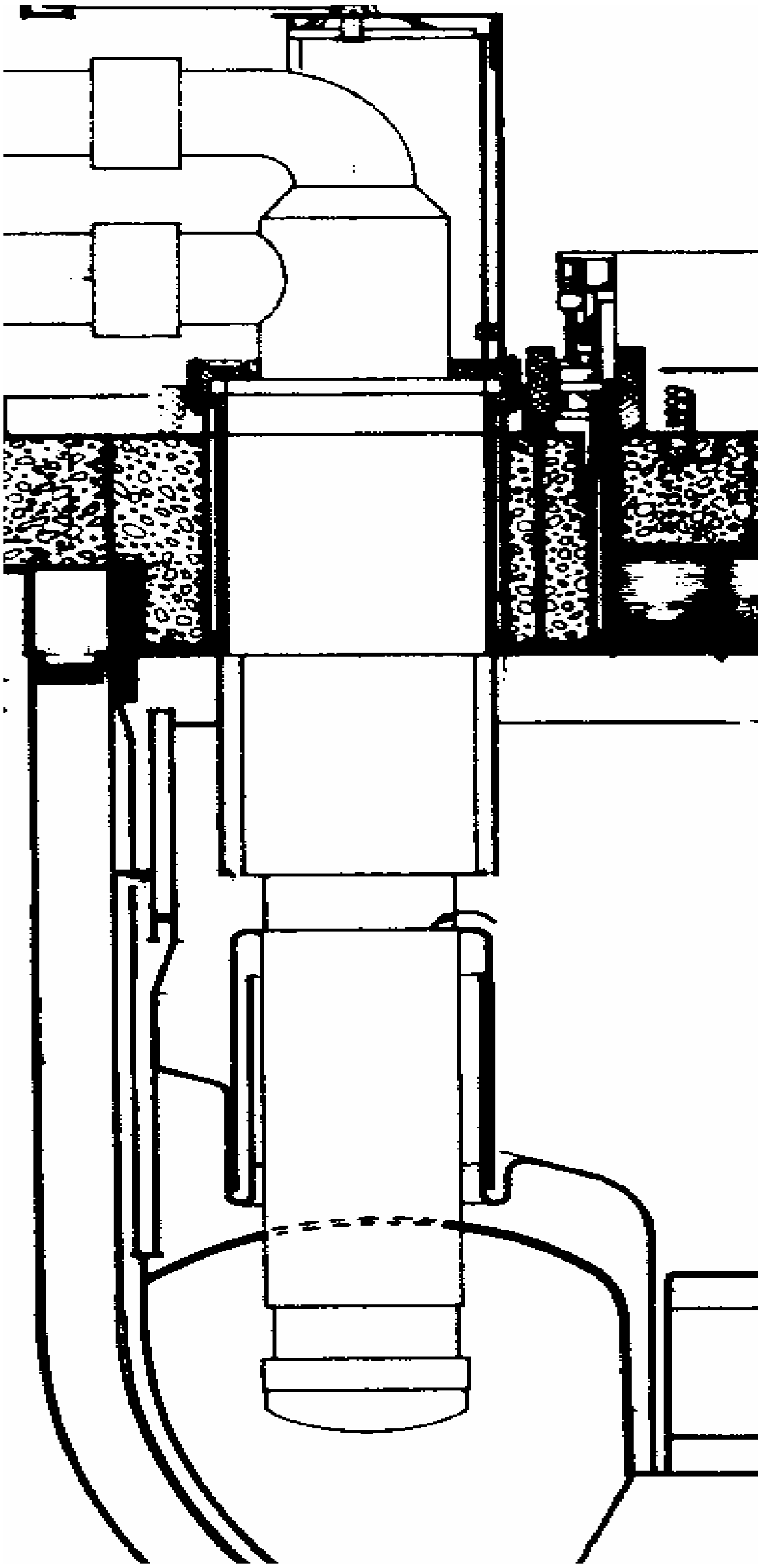


FIG. 21. Gas (argon) seal – sealing bell of the IHX.

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4. BN-600 AND BN-350 REACTORS

4.1. BN-600 POOL-TYPE LMFR

4.1.1. Commissioning and design features

The nuclear power plant (NPP) BN-600 has been operating since 1980 as the Beloyarsk-3 power plant. The NPP construction cost was ~ 312 million Rubles [1980] [approximately 620 million US$ (1980)]. The planned budget was exceeded by less than 5%.

First criticality was reached on 26 February 1980. The basic result of the physical startup in March 1980 (213 low (21%) enrichment fuel subassemblies (FSAs), 143 high (33%) enrichment FSAs and 13 permanent reactivity compensators) showed that the measured physical characteristics of the reactor were correspondent with the design values. Measurement of sodium flow through each FSA was carried out two times: before and after the power startup of the reactor.

The primary circuit hydraulics was investigated, at zero reactor power, both under steady-state conditions and simulated emergency conditions. All loops of secondary circuit were filled with sodium in February 1980. The investigation showed that the hydraulic resistance of the loops was two times below the design values. Power startup began on 5 April 1980. In mid of June power 40%, in mid of August 80% and on 18 December the reactor power reached the nominal power level 1470 MW(th) and 600 MW(e).

The design of the reactor is generally on the pool (integral) concept, with all the main primary circuit components placed in one vessel: the core, the radiation shielding, the main circulation pumps, the intermediate heat exchangers, and also the piping. The integral concept enables very compact reactor unit, which is reliable and safe as regards cooling of the core and confining radioactivity.

Pipelines with high temperature coolant, operating under stress are excluded, also are the cumbersome electric heating cables and the sealed concrete cells for location of the primary equipment. Less metal is used for the components, and the amount of construction work is greatly reduced. The surface area of load-bearing walls separating radioactive sodium from the external environment is largely reduced. Absolute leak-tightness of the main primary circuit pipes is not required, as leaks are confined within the reactor vessel.

The reactor vessel is of simple cylindrical shape, 12.8 m in diameter and 12.6 m high, with no nozzles below the sodium level. The low cover gas pressure in the reactor (~ 0.4 kg/cm^2) enables the large-sized reactor vessel to be made with a small wall thickness (30-40 mm). As experience has shown, this kind of vessel can be assembled on-site, from individual factory-produced parts, with minimal problems. It may be mentioned that the geometric dimensions of the BN-600 reactor vessel are almost the same as those of foreign pool type reactors-Phénix and PFR - although BN-600 plant power is some 2.5 times higher. This has been achieved as a result of using rationalized designs for the reactor and the equipment

The distinctive feature of the BN-600 reactor is bottom support of the reactor vessel which gives, in the designers' opinion, certain structural and technological advantages compared with alternative option of top-suspended reactors. Through a support ring welded at the point where the cylindrical wall joins the base, the vessel is seated on foundation roller supports.

Inside the vessel on the support ring is a rigid box-type support structure carrying the pressure chamber with headers and the core, the breeder blanket, the spent assembly store, and also the in-vessel radiation shielding, pumps and intermediate heat exchangers.

Thus all loads from the weight of the vessel, the in-vessel structures and the mass of sodium (~ 700 tons) are transferred to the foundation of the reactor well. The main advantage of this support layout is the load removal from the upper part of the reactor, which operates at the highest temperatures.

In the neck of the conical reactor cap there is a rotating plug with an eccentrically positioned rotating column in it, which carries the safety system drives and the two refuelling mechanisms. By combined rotation of plug and column, the refuelling mechanisms and grab system can be aligned with any recess in the core, the radial breeder or the in core storage. At the same time the rotating plug and column constitute a component part of the upper biological shielding of the reactor. Freezing liquid seals are used to prevent leaks.

To guarantee core cooling even in the event of hypothetical seal failure of the reactor vessel, the vessel itself and the adjoining pipe runs are enclosed in a sealed safety casing. Even complete filling of the space between vessel and casing will not cause a break in sodium circulation around the primary circuit. On the outside the safety casing is coated with thermal insulation. The whole reactor unit is contained in a concrete well covered from above by a protective plate.

Primary circuit components - the heat exchangers and pumps are installed in cylindrical support “sleeves” passing through corresponding necks in the reactor vessel cap. Connection of the “sleeves” to the vessel and the safety casing is effected using a bellows compensating the difference in temperature expansions (the relevant temperature difference < 100°C).

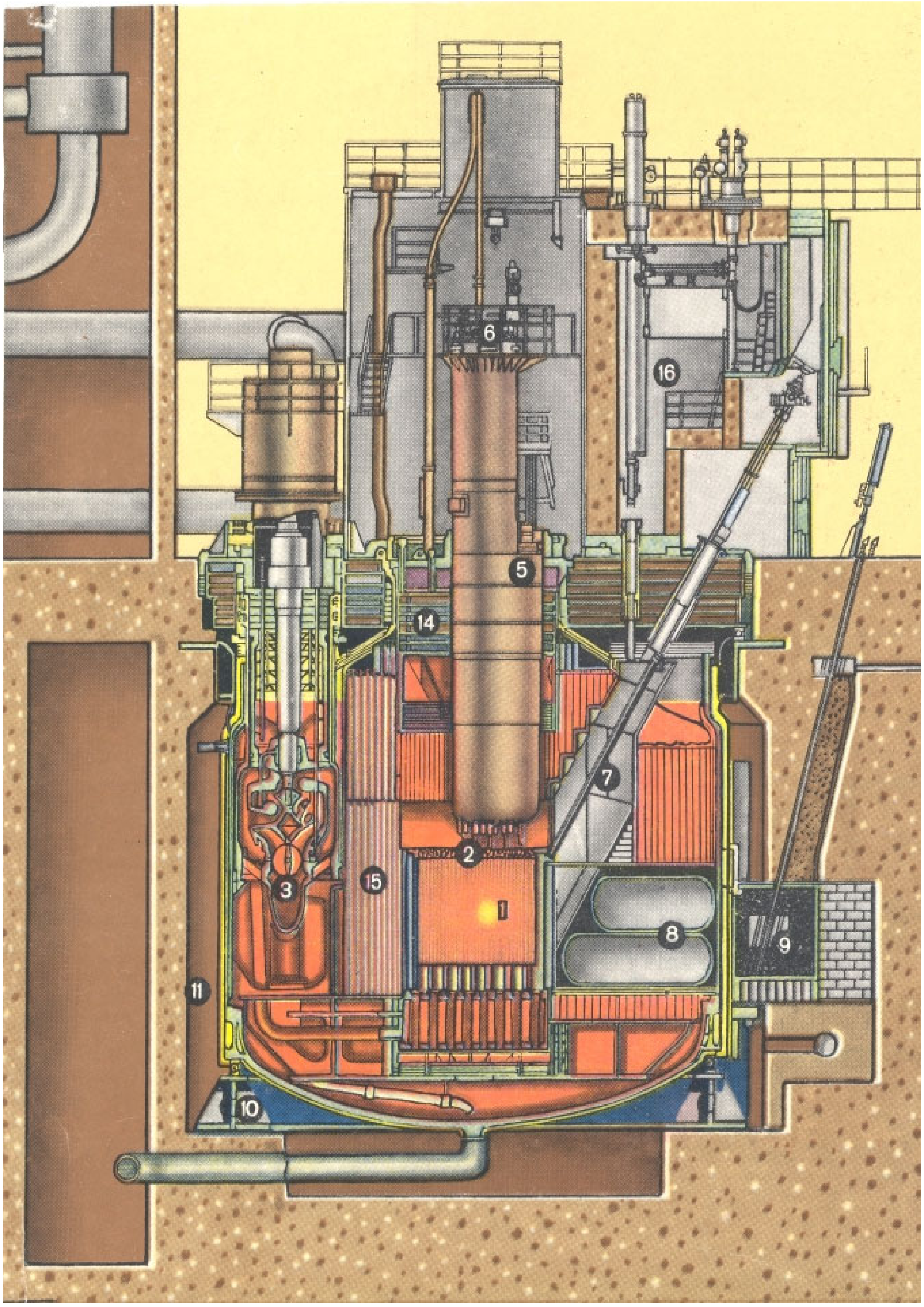
Heat removal from the core takes place via a three-circuit layout: there is sodium in the primary and secondary circuits, and water and steam in the tertiary circuit (Figs 1 and 2).

The primary heat exchangers are situated behind the radiation shielding, in an area of low neutron fluxes, so the secondary sodium reaching the steam generators has virtually no activity. The secondary circuit also provides shielding of the reactor against the consequences of possible faults in the steam generators. Coolant circulation takes place via the three parallel primary circuit loops, each of which incorporates two heat exchangers and one vertical immersion-type centrifugal pump with direct intake from the reactor tank.

The pumps with a delivery of about 10 000 m^3/h each, are situated in the “cold” part of the circuit, after the heat exchangers, which ensures low and stable temperature conditions for their operation. Sodium at a temperature of ~ 380°C is fed by the pumps to the pressure header, from which it is distributed around the assemblies of the core and radial breeder according to the maximum heat release in them. Part of the coolant goes to cool the reactor vessel (~ 1000 m^3/h), the spent assembly store and the in-tank shielding. Emerging from the assemblies at a mean temperature of 550°C, the sodium passes into the upper space of the reactor, from where it flows out via six intermediate heat exchangers.

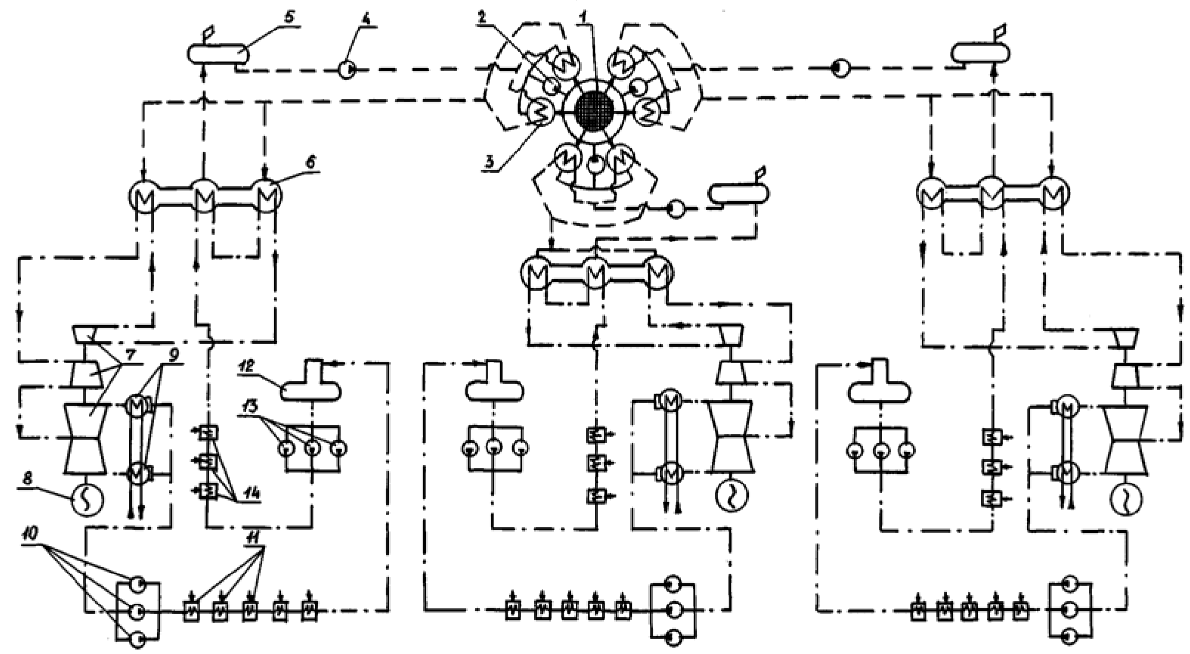
The secondary circuit sodium, at a temperature of 320°C, is fed down a central fall pipe into the lower space of the heat exchanger, from where it rises up through tubes, becoming heated to 520°C by the heat given off by the primary circuit coolant. The ascending movement of the coolant in the core of the reactor is chiefly selected on the basis of the conditions for the development of natural circulation.

There are no shut-off valves in the primary circuit. Shutting down one loop is carried out by forced closure of a non-return valve with the corresponding primary circuit pump halted. For the power plant to operate at intermediate power levels with parameters close to nominal, a controllable drive is used in the pumps, providing a smooth variation in the rotation speed (and the sodium flow rate) in the range 25–100% N nom.



1, 2-core, fuel assembly, 3-primary pump, 4-intermediate heat exchanger (IHX), 5-central column, 6-control rod drive mechanism, 7-loading-unloading elevators, 8-neutron channel, 9-neutronic measurement chambers, 10-reactor supports, 11-reactor vault, 14-rotating plug, 15 neutronic protection, 16-refuelling cell

FIG. 1. BN-600 reactor block.



1-core, 2-primary pump, 3-intermediate heat exchanger, 4-secondary pump, 5-buffer tank, 6-steam generator, 7-turbine, 8-generator, 9-condensers, 10-condensate pumps, 11-low pressure heaters, 12-deaerator, 13-feed electric pumps, 14-high pressure heaters

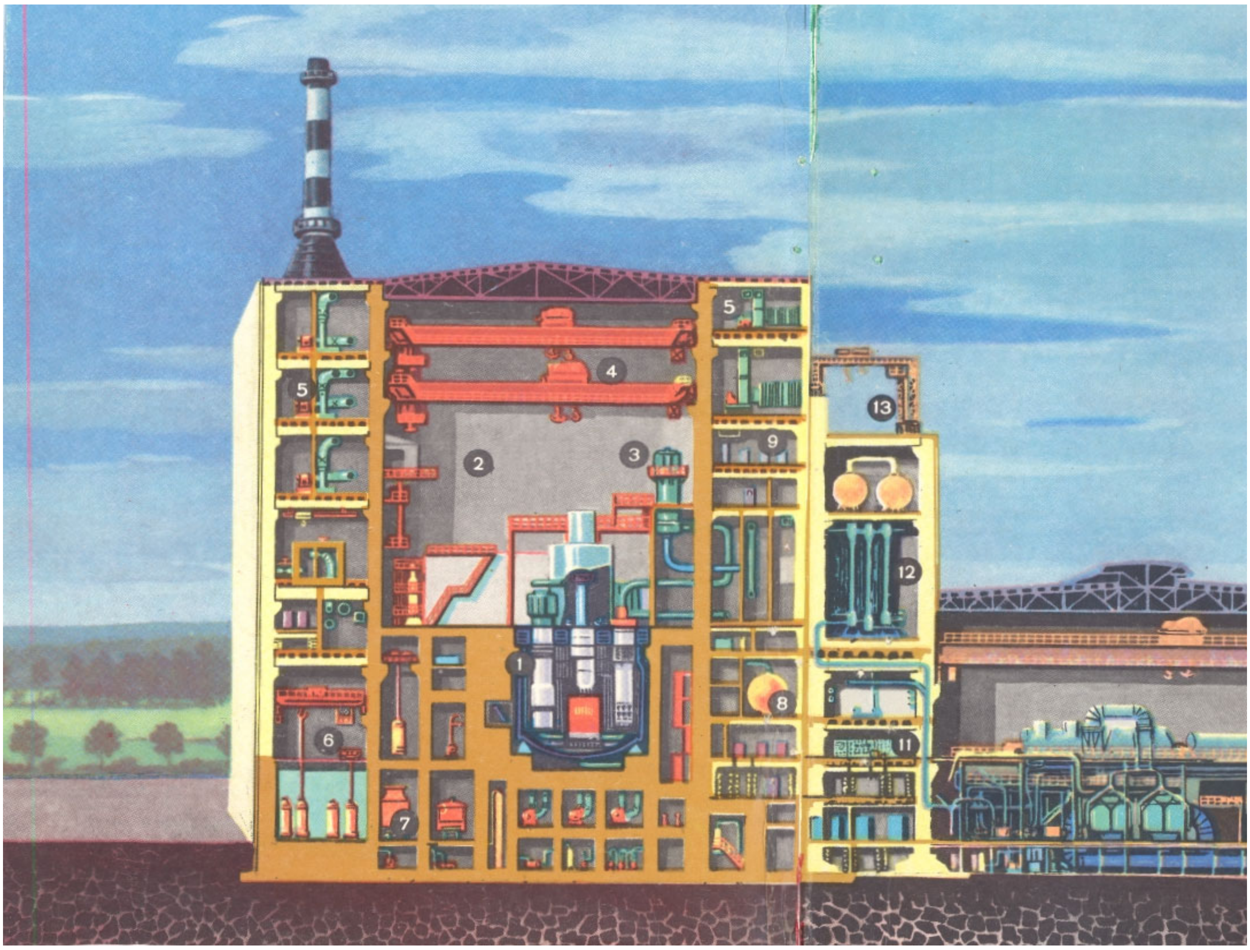
FIG. 2. Principle heat diagram of the power plant BN-600.

There are no shut-off valves in the primary circuit. Shutting down one loop is carried out by forced closure of a non-return valve with the corresponding primary circuit pump halted. For the power plant to operate at intermediate power levels with parameters close to nominal, a controllable drive is used in the pumps, providing a smooth variation in the rotation speed (and the sodium flow rate) in the range 25–100% N nom. The core is made up of a compact grouping of around 400 hexagonal assemblies with an “across-flats” size of 96 mm and a spacing interval of 98 mm. An assembly (subassembly) comprises 127 pins of 6.9 mm external diameter contained in a casing tube with a perforated spike at the bottom for a feed of coolant from the pressure header, while at the top there is a head for gripping by the refuelling mechanism during assembly transportation; 27 holes in the core are occupied by the control and safety system rods of the reactor.

The fuel charging of the core and the efficiency of the reactivity compensation system ensure continuous operation of the reactor for a period of some 150 days (at full power). The duration of a core run is around 450 days, so that at each refuelling 1/3 of the assemblies which have achieved the design burnup are replaced. Around the core assemblies are assemblies which are identical in external configuration, forming the radial breeder. The elements of these assemblies are filled with waste uranium dioxide. The axial breeders consist of the same material. These are situated in the same claddings as the pins. Behind the radial breeder there the positions for the in-pile store. It is used to cool assemblies removed from the core before they are removed from the reactor. The BN-600 reactor is designed to operate on both uranium and plutonium fuel, and a gradual transition to breeder mode by making up the reactor with plutonium bred if it is possible. Fuel subassembly refuelling takes place with the reactor shut down using a set of mechanisms comprising. Figure 3, as an example, shows of the loading/unloading system of the BN-600, which is in operation since 1980. The latter comprises:

- Three of an eccentric arrangement rotating plugs with two in-pile refuelling mechanisms (close and distant relative to the reactor core axis) installed on the small plug, which carry out replacing of assemblies inside the reactor;
- Two drums for new and spent fuel assemblies;
- A spent fuel-to-washing cell transfer mechanism;
- Fuel transfer and washing cells;
- Two inclined loading-unloading elevators, which transport the assemblies from the reactor to the transfer box of the handling and transport channel and back;
- An assembly transfer mechanism situated in the transfer box, which transfers assemblies from the elevators to the handling and transport channel and back.

The sodium level in the reactor vessel is such that transportation of spent assemblies inside the reactor takes place under a layer of sodium. This prevents any danger of inadmissible heating of fuel being moved under conditions where there is no forced coolant circulation. Subsequent cooling of spent assemblies in the in-pile storage with organized cooling enables the level of heat emission in the fuel to be reduced substantially, which simplifies subsequent handling of it in the external handling and transport channel. A rectangular shape was adopted for the BN-600 reactor building after comparing different versions (Fig 3 a–d).



1-reactor, 2-reactor hall, 3-secondary pump, 4-crane, 5-ventilation system, 6-water pool, 7-irradiated fuel transfer cask, 8-sodium storage tank, 9-electric heating control system, 10-turbine hall, 11-control and protective system, 12-steam generator (SG), 13-crane (for SG)

FIG. 3a. Nuclear island and turbine building layout-elevation.

FIG. 3b. The BN-600 power plant building.

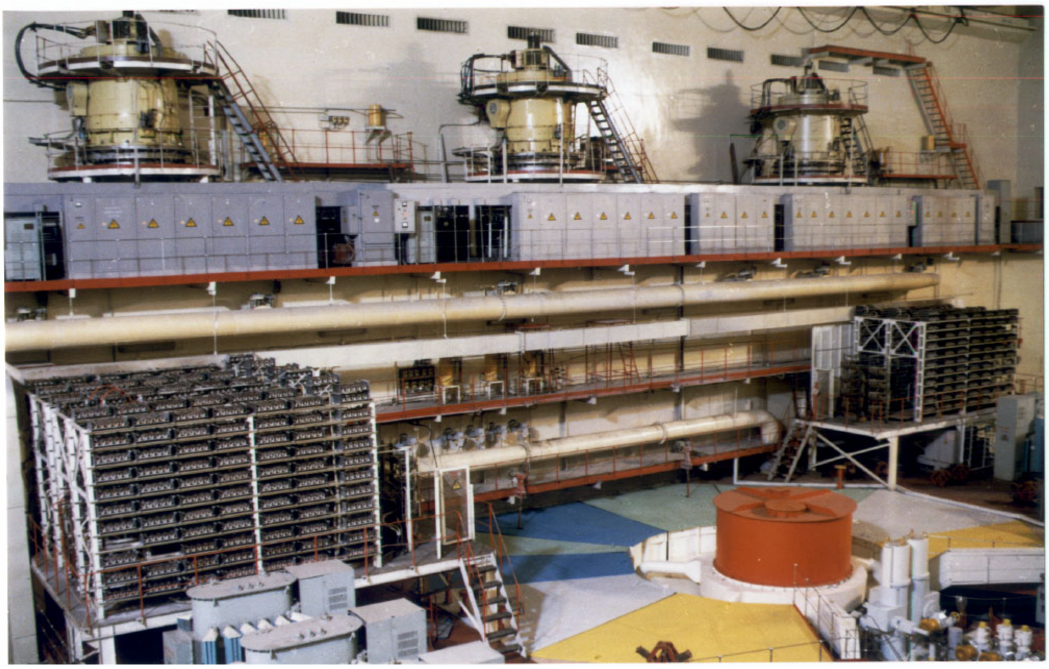


FIG. 3c. BN-600 reactor central hall
(on the top-secondary pumps, bottom-reactor protective dome (red).

FIG. 3d. Turbine hall of the BN-600 NPP.

4.1.2. Equipment tests[6]

A special attention was given to verify the correctness of the design principles used in the BN-600 and the functional qualities of the equipment. The primary circuit hydraulics tests were performed on a 1:6.8 scale reactor model with operating pumps (Fig. 4).

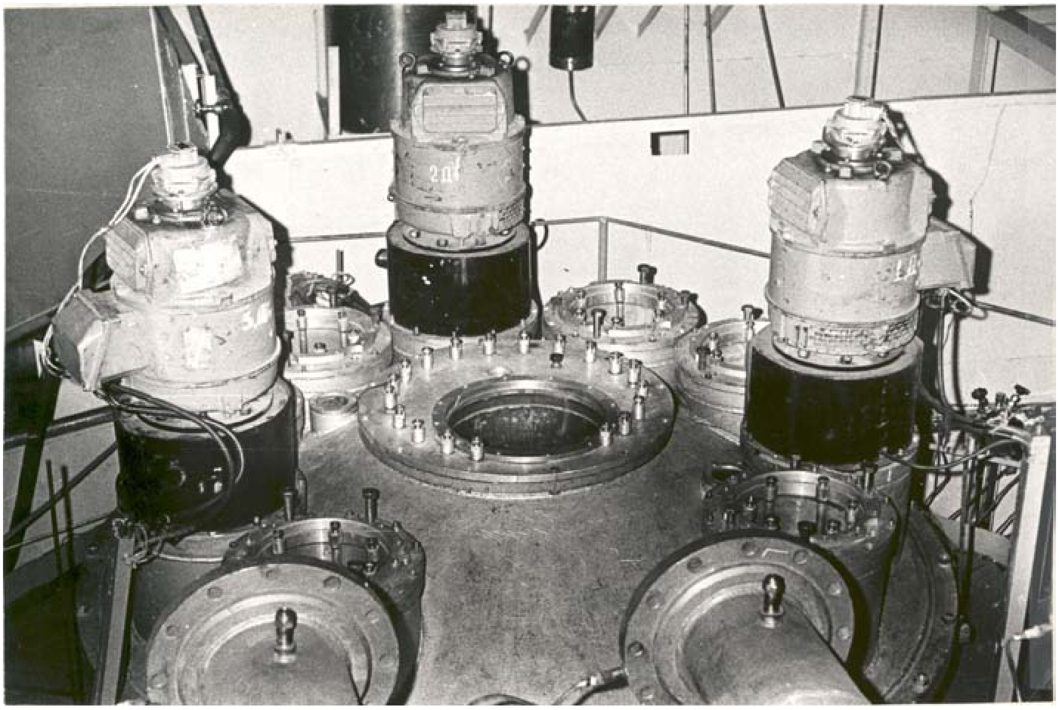


FIG. 4. Hydraulic model of the BN-600 reactor.

[6] For more detailed information see: LEIPUNSKII, A.I. et al, The BN-600 fast reactor; BUDOV, V.M., et al., A NPP BN-600 - the plant for the near future; Reports presented at the Nuclex-75, Basel, 1975.

The data produced in this way enabled checks and changes to be made with the results of calculation studies, on such basic issues as:

- The distribution of the coolant flow rate through the heat exchangers in various reactor operating modes with one loop not operating;
- The variation in the coolant level in the reactor vessel and the pump stand pipe under various reactor operating modes;
- The absence of any mechanical entrainment of gas by the coolant from the reactor vessel into the circuit, etc.

On a separate rig a study was performed of the coolant distribution through assemblies in the pressure headers. The influence on the flow rate through the assembly of the position of the holes in the assembly spikes was determined. Tests and final adjustments to the refuelling mechanisms were performed on sodium rigs with circulation of hot sodium. These tests particularly envisaged the performance of a number of studies in artificially created “accident” situations, such as failure of the control system, complete loss of power to mechanisms, high oxide content in the sodium, etc. Comprehensive checking of the refuelling system together with the control system was performed on a control assembly rig which comprised: metal structures simulating the vessel and reactor transfer cell, rotating plugs, central column with control and safety system drivers, ramps, mechanisms for refuelling and transfer of assemblies, dummy subassemblies, a simulated inlet plenum and electrical equipment and movement systems. On the rig, a number of subassembly refuelling operations were carried out, both in normal mode and in conditions simulating of all types of events. The latter included:

- Loading and unloading of deformed subassembly with a 10 mm bend, with subassembly rotated through 60, 120 and 180°;
- Loading and unloading of subassembly, with misalignment of axis of refuelling mechanism relative to subassembly head axis by 10 mm;
- Checking possibility of disconnection and removal of refuelling machine from jammed subassembly, raised to a height of 500 mm.

Under all these conditions the refuelling mechanisms were found to function satisfactorily. The refuelling mechanisms, elevators and fuel transfer mechanisms in the cells have been operated without any disturbances; above 40 reactor reloads have been carried out since the power unit BN-600 was put into operation. The total operating age in terms of double strokes of the in-reactor refuelling mechanism amounts to 36 100, for the elevators to 9 060 and for the ex-reactor fuel transfer mechanism to 29 200. These exceed the respective design values. Based on results of audits of the refuelling mechanisms, their operating life has been extended.

4.1.2.1. Sodium rig for testing of drivers and control and safety system rods

Experimental final design work and full life tests were performed on the control and safety rod drive and their rods using a special sodium rig at a temperature of 550-580°C (Fig. 5). The control and safety drive prototypes with absorber rods worked about 16 000 h on the rig, during which time the following operations were carried out:

- Automatic power regulation drive mechanism - 7 500 raising and lowering;
- Safety rod drive mechanism - 650 drops in fast scram mode and 600 raising and lowering in slow emergency protection mode;
- Shim rod drive mechanism - 2 600 raising and lowering.

These lengthy tests confirmed the reliability of all the control and safety rod drive mechanisms and the conformity between the main working characteristics and the design characteristics.

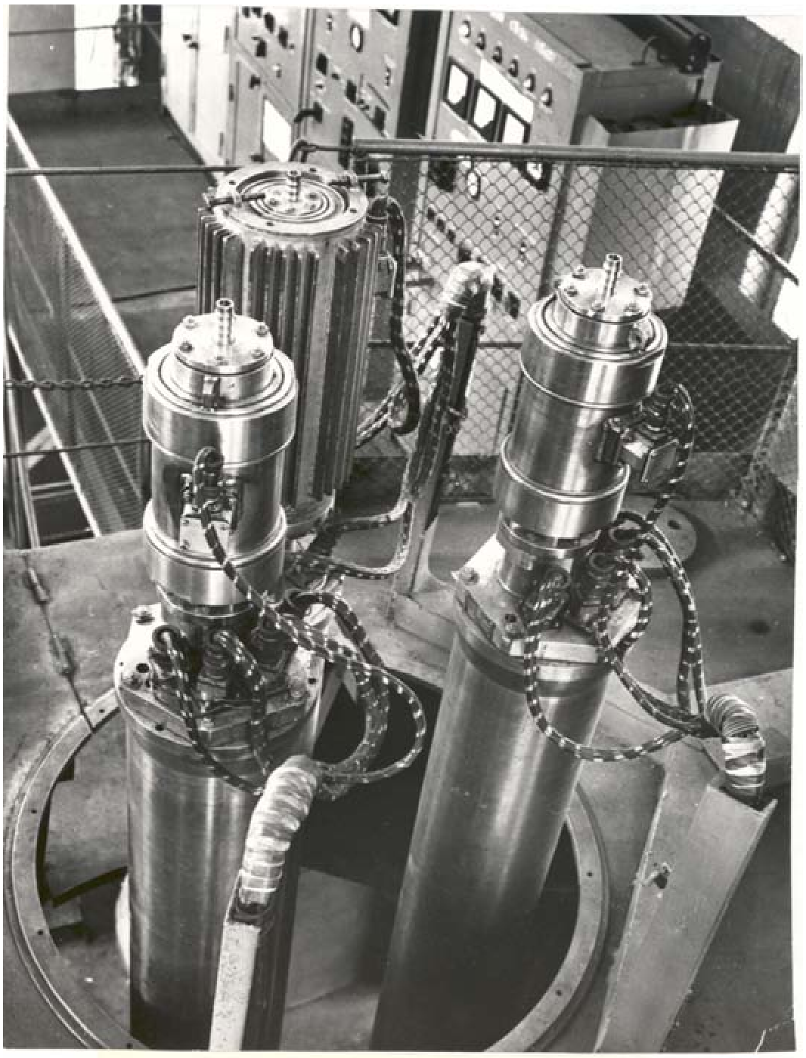
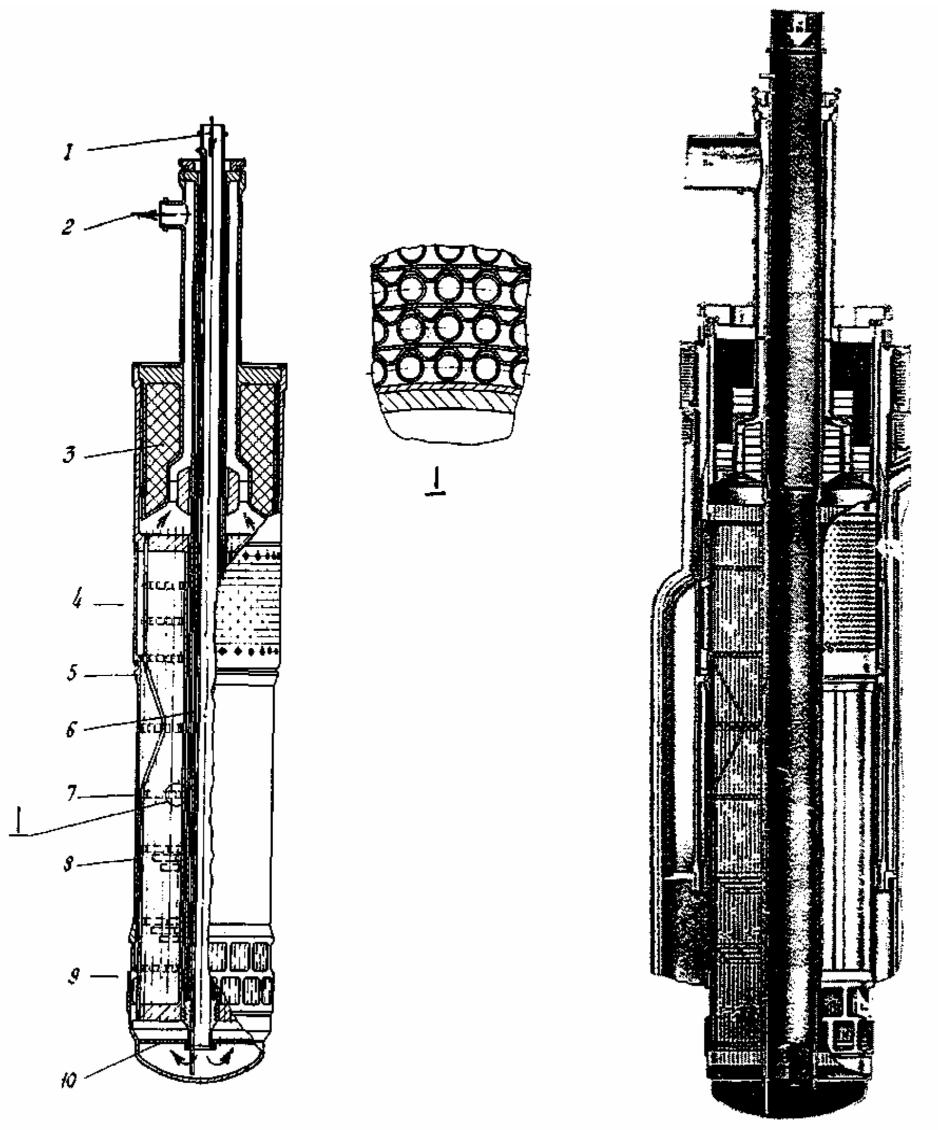


FIG. 5. Sodium rig for testing of drives, control and safety system rods for the BN-600.

4.1.2.2. Intermediate heat exchanger (IHX)

The IHX shell-and-tube sodium/sodium heat exchanger has 4 974 OD 16×1.4 mm tubes with expansion bends. In order to minimize the temperature difference between the bundle and the central tube, a double-tube arrangement was introduced. The annular gap is gas-filled. The perforated plate at the below part of the IHX was used to reduce the temperature difference between central tube and bundle. The distribution of the coolant flow rate through the heat exchangers in various reactor operating modes, including modes with one loop disconnected were performed on a 1:6.8 scale reactor unit model (Fig. 6). The study of IHX thermal hydraulics was performed on the mock-up tube bundles with the limited number of tube rows (one-, three-row bundle).



1, 2-secondary Na inlet/outlet; 3-shielding block; 4,9-primary Na inlet/outlet; 5-seeling element;
6-central downcomer tube; 7,8-heat exchange tube; 10-lattice

FIG. 6. BN-600 intermediate heat exchanger.

The low cycle fatigue and fracture investigations done on tube-tubesheet joint IHX nozzle model and IHX to IHX support shell joint. For the tube-tubesheet joint, two models with 19 and 37 tubes respectively were studied. The test results for heating rate 50-70 K/h hold period of 3 h at 550°C and cooling rate with of 5-6.5 K/s had not indicated any fatigue damage.

The tests on nozzle model with the heat rate 60-90 K/h hold at 580-600°C for 3 h and cooling in time 30-40 s up to 400°C revealed no crack initiations in the all surfaces of 2 test models. The calculated peak to peak stream values was 0.62% which can permit up to 3 000 cycles. The last aspect was on the creep fatigue damage on IHX support shell joint. The calculations showed that the margin on number of load cycles is less than the permissible value. Hence tests were conducted on the flat model, with 12Cr18 Ni steel.

The test temperature was 490°C. The calculation and experiments' comparison showed a good coincide. Accordingly crack initiation in the weld bead zone was observed after

1 100 cycles by acoustic emission method for the clearance of 0.5 mm between the shells. If this width is increased to 2 mm, cracks were seen only after 2 000 cycles. All six IHXs are operated since 1980 without any faults or troubles.

4.1.2.3. The pumps

The pumps underwent various model and full scale tests on rigs, where there was the possibility of simulating widely (bearing) varying operating conditions were carried out to guarantee the required high level of operational reliability. When the pumps were being designed, much thought was given to the hydraulic side, using computers.

Models of the pumps were created, and these were subjected to various hydraulic tests with water. Some of the pump units (shaft seal, bearings) were tested on special rigs, which enabled design changes to be made to them promptly and various designs to be tried out. Besides the usual testing of the functional properties of the pumps and their drives in all possible operating conditions, the full scale water tests also included checking of the pumps under extreme conditions, e.g. during operation under cavitation conditions, with non-regulation start-up, with delivery overload, etc. Rigs for testing pumps with hot sodium enabled the hydraulic and electro-mechanical characteristics of the pumps and their parts to be checked under various thermal conditions. Successful long term life tests on pumps confirmed the correctness of the designs selected for them.

The strength of the highly loaded reactor vessel parts was tested using models. In particular, a 1:10 scale model of the roof slab was made. The basic principle of the modeling work was equivalence of the stress level of the model to that of the “real-life” version. On a rig simulating a real pipe with normal pump, studies were performed of the vibration and vibration-strength characteristics.

Construction work and the manufacture and assembly of equipment for the BN-600 began in 1969. The reactor plant building was made ready for equipment assembly in 1973. Assembly of the main equipment commenced in 1974. By mid 1975 welding of the reactor vessel was practically complete. While the vessel was being assembled, the secondary circuit pumps and the component handling and storage equipment were also being put together. Assembly work was completed in 1979. The power station with BN-600 reactor was commissioned in April 1980.

4.1.3. Operating experience

Up to 1 January 2004 the reactor plant total on-power operation time amounted to 164 000 h, and about 88 000 GWh of electricity was generated. The power histogram and load factor for the unit operation are given in Figs 7 and 8.

It may be mentioned that the BN-600 is operating as a base load plant, with the reactor not participating in the regulation of load and frequency in the power system. The reactor power was therefore, nominal [600 MW(e)], 2/3 of nominal during operation without one of the heat transfer loops is not available, or zero during shutdown. The time spent by the BN-600 in each of these three states is characterized by the following data:

- Nominal (600MW(e)): 68%
- 2/3 power: 11%
- Zero power: 21%.

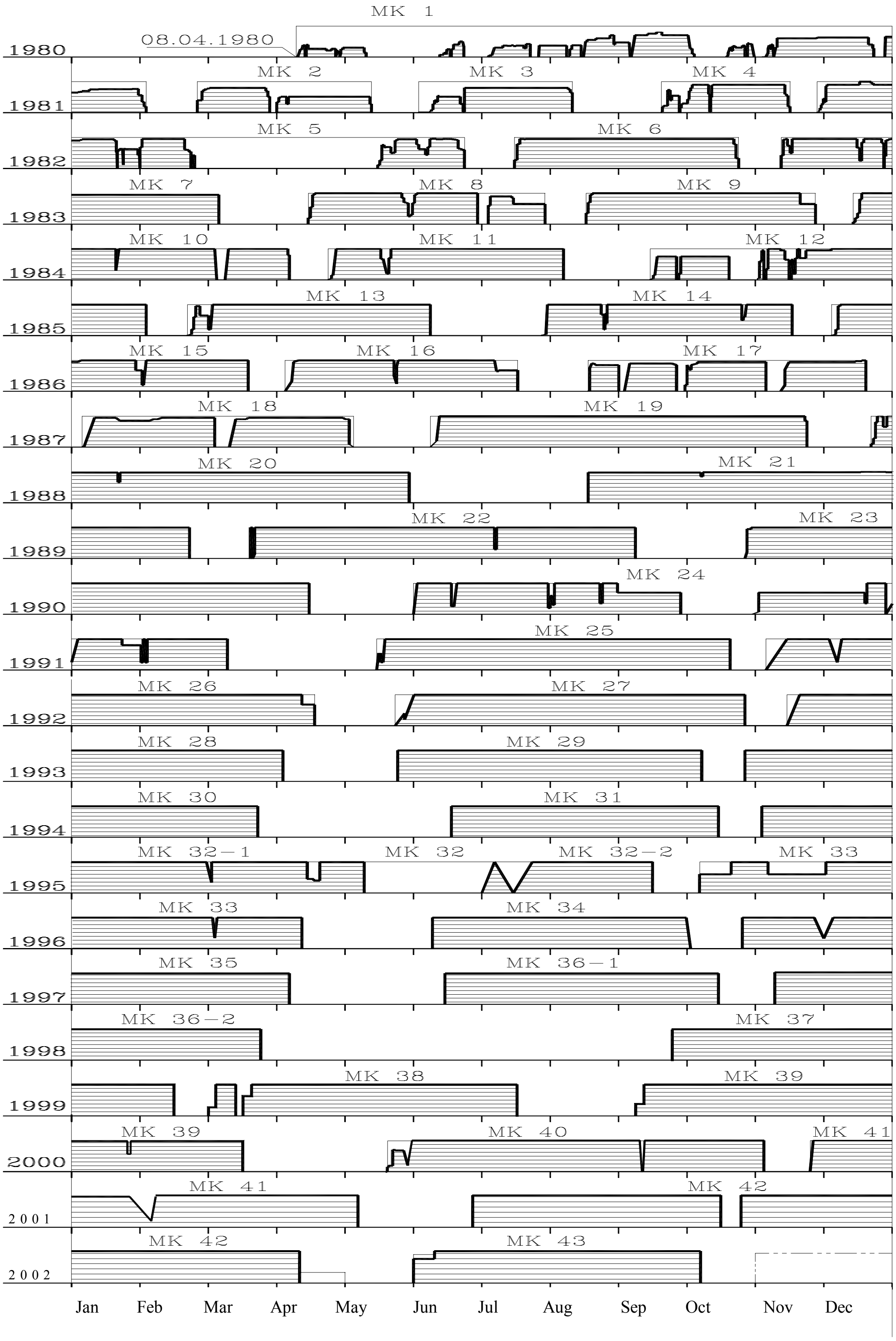


FIG. 7. BN-600 operating histogram (reactor is being shutdown two times per year for refuelling, 1998-the rotating plug repair).

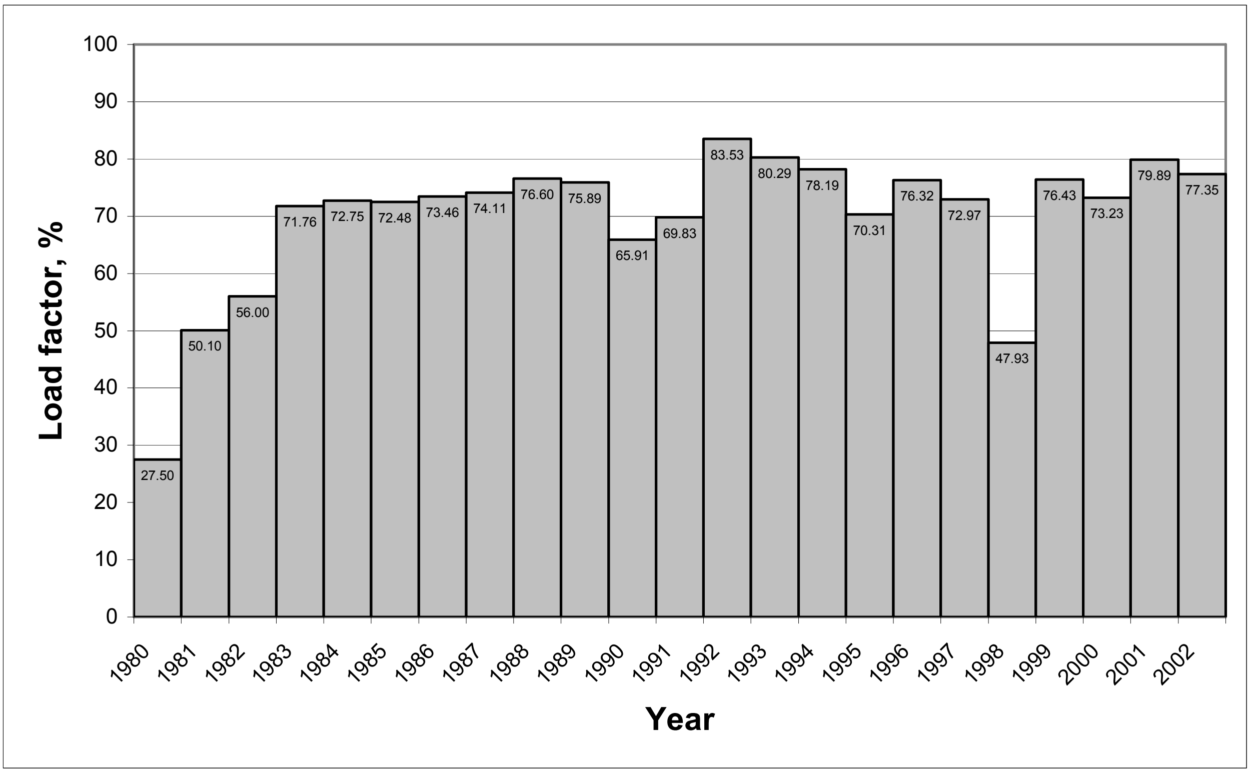


FIG. 8. BN-600 load factor (1998-the repair of the rotating plug).

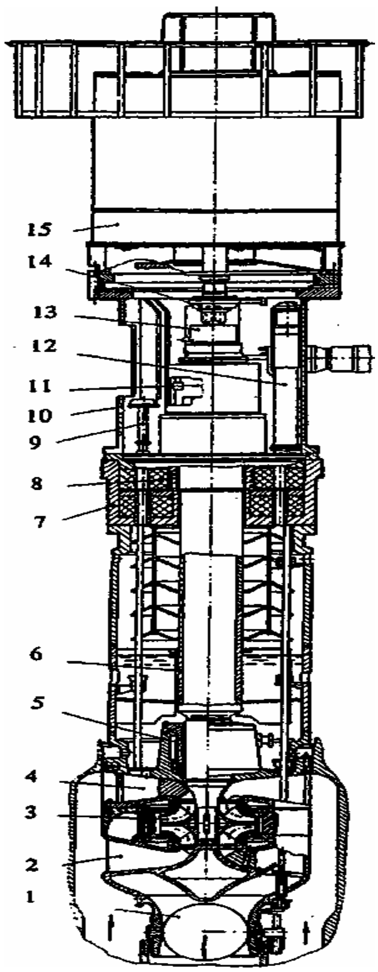
The unit load factor during initial years was as follows: 1980-27.5%; 1981-50.1%; 1982-56%. With experience, the load factor rose steadily, shutdowns of the unit are now mainly caused by routine preventive maintenance and refueling. The average load factor for the whole period of the reactor plant operation on full power equals to 70%, in 2001- 79.89%. The highest load factor (83.5%) was reached in 1992. As to reliability indicators the BN-600 power unit is among the best NPPs in Russian Federation. The BN-600 control system is so constructed that if any of the three heat-transfer loops are shut off the unit remains at 67% power. From 1982 on work was carried out by operational personnel to produce an algorithm for resumption of operation of a disconnected loop without reactor shutdown, with power reduced to around 20%.

There have been some 69 disconnections and connections of loops over nearly 22 years of operation. Analyses showed, if the two-loop operating mode had not been envisaged, load factor losses over the operating period which has elapsed would have been about 7%. This in fact took place during the early years of operation. Experience in operation of the BN-600 (and also of other three-loop fast reactors) showed that the need to work with two loops arises when equipment fails because of so-called “chronic faults” which require long-term repair (or replacement). Under BN-600 conditions, such faults were found in primary circuit pumps, steam generators and turbines in the initial phase of operation. After scrapping of the faulty steam generator modules and adjustments to the pumps, the BN-600 operated with a minimal number of loop disconnections. One may thus conclude that for serially-produced reactor installations with fully-developed equipment, and one turbine for reactor, and steam generators with high self power there is no special need to envisage stopping/switching of the loop without reactor shutdown, especially if this means additional capital cost and a safety margins decrease. The greatest losses associated with unplanned shutdowns (equipment and system failures) at the beginning stage of reactor operation have been connected with the primary circuit pumps.

Thus, in BN-350, there were disturbances in the pumps at start up after a long-term outage, caused by solidification of sodium in the gap between the pump shaft and its casing. There were also difficulties resulting from an increase in the sodium leakage flow from the impeller delivery due to temperature variations in transients, which prevented the sodium level rising or varying in the pump tank. During the initial period of operation of each of the reactors certain technical problems were solved. This concerned adjustment of the systems and components, bringing the reactor to nominal (rated) power, and feedback of operating experience.

4.1.3.1. Primary pumps

Some problems in the operation of the BN-600 primary pumps arose in the process of coming up to power in 1981 during a routine stage of increasing reactor power. When the pump rotation speed was increased, increased bearing vibration was noticed. It was found that the frequency-controlled motor coupled to the synchronous-rectifier drive induced torsional pulsations of 6–8% of the nominal torque. When the torque pulsations coincided with a natural frequency of shaft vibration resonances arose resulting in adverse consequences, such as cracks in the shafts and failure of the couplings (section 14 in Fig. 9). During 1982-83 strain measurements were made to evaluate the pump shaft stresses and prohibited zones in the shaft rotation frequency range were determined.



1-check valve, 2-lower scroll, 3-impeller, 4-upper scroll, 5-hydrostatic bearing, 6-shaft, 7-cover, 8-cooler, 9-level gage, 10-motor base, 11-face seal, 12-check valve drive, 13-radial-thrust bearing, 14-coupling, 15-motor

FIG. 9. BN-600 reactor primary pump.

In a number of cases the pump and the corresponding loop had to be disconnected, putting the reactor into the 2/3 power operating mode. During repairs to the shutdown pump, breakages were found in the half-coupling teeth and springs connecting the pump shaft to the electric motor shaft. Analysis of the damage to these parts led to the conclusion that they were not strong enough. These defects were eliminated by replacing the connecting couplings by stronger ones. During subsequent operation of the primary circuit pumps, areas with high amplitude of torsional vibrations of the pump shaft were found in the rotation speed range of 925-940 rpm. The dynamic deformation constant in the resonance zone did not exceed 20-25% of the static load on the pump.

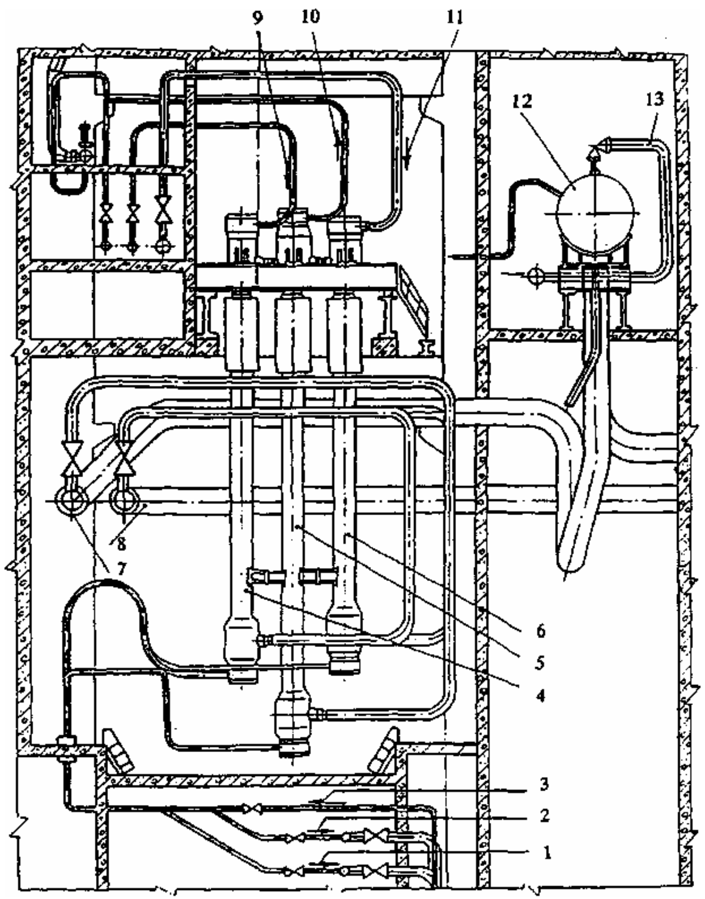
The torsional vibration frequency coincided with the pulsation frequency of the electromagnetic moment of the drive and the mechanical vibrations of the pump unit itself. All these phenomena were, however, observed in an area other than the working rotation speed of the pump. The removable parts of the primary circuit pumps have been replaced. When the unit is at nominal power, the pumps operate in unregulated mode, i.e. in squirrel-cage rotor mode. The use of advanced shafts and couplings and changing to a steady mode of pump operation after attaining the preset reactor power level have eliminated any failures of the reactor coolant pumps since 1985. There have been no problems with operation of the secondary circuit pumps. Long term operating experience has allowed the factors which limit the lifetime of the pumps to be identified. Cavitation-erosion wear of the impeller blades is now the single factor requiring periodical removal of the primary coolant pump internals to replace the impellers. The operators were not satisfied with the design lifetime of the face shaft seals, which was limited by degradation of the rubber sealing collars (lose of elasticity and cracking). The seals were replaced with spares, as a rule, during planned maintenance outages. The design of the seals currently in use has been modified significantly compared with the original design and has the following advantages:

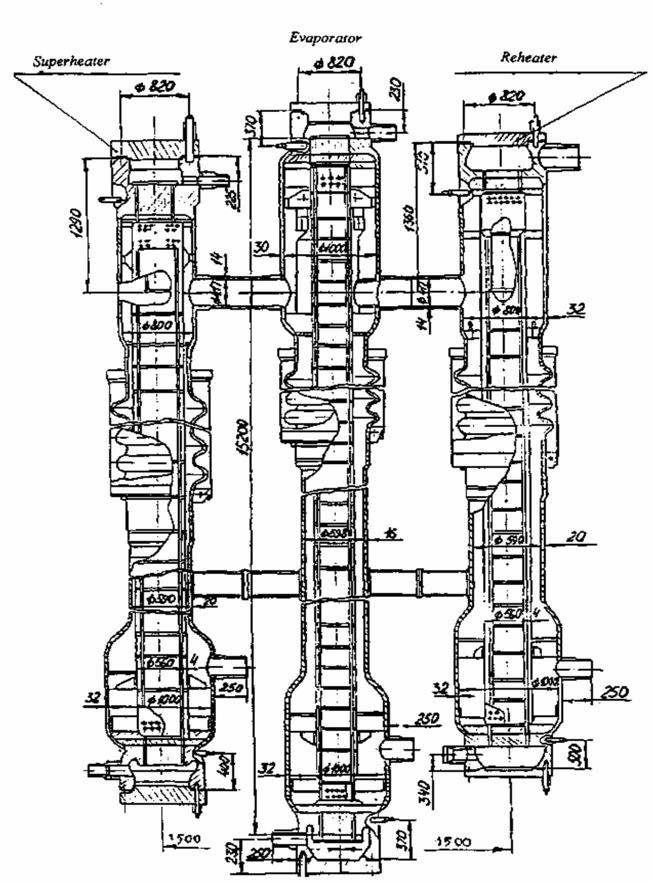
— Rubber sealing rings are used instead of collars - the possibility of misalignment of the sealing surfaces is reduced by improvement of the items which transfer the torque to the rotating sealing rings;
— A wear-resistant “graphite-to-graphite” sealing pair is used instead of “graphite-to-steel”;
— The oil-cooling surface in the seal is increased.

On the basis of operating experience and resulting from the improvements made in design the operating life for the pump internals has recently been increased significantly up to 50 000 for the primary pumps and to 100 000 for the secondary pumps.

4.1.3.2. Steam generators

In the earlier years of the development of fast reactors there was some incidence of leaks in the steam generators. Since leaks in sodium-heated steam generators are intolerable, however small they may be, whenever a leak occurs the unit in question has to be shut down and isolated for repair. This can be done without serious diminution of the power output only if there are a large number of separate units, any one of which can be isolated without reducing the total power significantly. The outstanding example of the advantages of the modular approach to steam generator design is afforded by the Russian BN-600 plant. It can be seen on Figs 10 and 11, and its main characteristics are shown in Table 1. This has three secondary sodium circuits, each with 8 separate steam generator modules, and each of these consists of separate evaporator, superheater and reheater sections, making a total of 72 separate heat exchangers. At least partly because of this the availability of BN-600 has been consistently high.





1-main steam removal, 2-reheated steam removal, 3-feed water supply, 4-superheater, 5-evaporator, 6-reheater, 7-sodium removal, 8-sodium supply, 9-steam to superheater, 10-steam removal from evaporators, 11-steam to reheater, 12-sodium expansion tank, 13-steam-sodiumreaction products dump

FIG. 10. BN-600 modular stem generator layout in compartment design line.

FIG. 11. BN-600 modular steam generator.

TABLE 1. MAIN CHARACTERISTICS OF THE BN-350 AND BN-600 SGs

Steam generator module	Tube material	Number of tubes per bundle	Tube diameter, mm	Tube spacing, mm	Sodium temp., ºC	Water/ steam pressure, bar	Water/ (steam) temp., ºC
BN-350 evaporator	2 1/4 Cr-1Mo	816	32×2 (33×3)	44	350-420	45	250-260
BN-600 evaporator	2 1/4 Cr 1Mo	349	16×2.5	28	320-450	140	220-340
BN-600 superheater	Cr18Ni9	239	16×2.5	33	450-520	135	350-505
BN-600 reheater	Cr18Ni9	235	25×2.5	33	450-520	28	505

Of the 12 cases of failure of the seal between circuits in the steam generator in only 2 cases was it necessary to shut down the corresponding heat-removal loop, but the reactor; the total duration of such shutdowns being around 400 h. In the other cases, the presence of valves enabled the sodium and water/steam section to be cut off with the heat removal loop in operation, virtually without reducing the reactor power. The modules which had suffered leaks between circuits were removed and replaced. Studies showed that the most likely cause of the water-into-sodium leak was manufacturing faults. This situation occurred in the superheater modules (7 leaks) and reheater modules (4 leaks) which are made of austenitic steel.

Studies revealed that the point of the original leak was in the area of the upper tube plates. The suggested cause was the development during operation of initial defects at the point of junction between tube and tube plate. In all the events irrespective of a leak size, no increase in pressure in the expansion tank up to the alarm setting has been observed and the operational power plant safety has remained within the limits. The precise design concept of a 'Large leak' event and the steam generator emergency protection system operation algorithms were validated by the analysis of the leak behaviour. In 10 events the failed modules have been replaced with new ones and in 2 events the failed modules have been restored and put into operation again. The restored evaporator module has accumulated 7 280 trouble-free running hours after the intervention and has been replaced with the new one for material testing. The restored reheater module has accumulated 53 854 running hours after the intervention and has been removed out of service on indications of water-sodium reaction having totally accumulated 68 366 running hours. Thus the design concept of the maintainability of the steam generator modules if they have been disconnected at the early phase of the 'Small leak' event development, has been also validated.

Analysis of the incidents that had occurred led to the following conclusions:

— The monitoring system worked satisfactorily under these conditions and in most cases enabled the time of the leak, the section it was in and even the module to be determined promptly;
— The confinement system also operated satisfactorily. It ensured a controlled dumping of argon-hydrogen mixture without the maximum design pressure being exceeded. At

the same time problems were encountered in implementing the algorithm for rapid creation of nitrogen counter-pressure on the tertiary circuit side following evaporation of the steam-and-water mixture from the affected module. Operating experience revealed the following possibilities:

— Continuing operation of the loop with the section disconnected without reducing the loop power;
— Disconnection of the affected section without shutting down the reactor, and even without disconnecting the loop;
— Connecting up a repaired loop without shutting down the reactor.

To give the modules better functional ability, the results of analysis of affected modules were used to develop and carry out a number of measures, particularly the installation of electric heating of the entry sections of steam lines of the superheaters and reheaters. Generally speaking, the sectional/modular design of the steam generator, fitted with the appropriate shut-off and stop gate valves, has proved acceptable under operating conditions, though relatively expensive.

4.1.3.3. Central rotating column

After 3–4 years of operation there has been some tendency for the rotation torque to increase because of deposition of sodium oxide. The cause has been identified and modification to correct the problem was done. During recent several years, growing increase of torque of the central rotating column (CRC) has been observed within some angle of its rotation. By 1997, it reached as high as about 60% of permissible value determined by the strength and capacity of its drive. In 1997, CRC bearing assembly was inspected by drilling special opening and using endoscope and some amount of sodium was detected there. This phenomenon was interpreted as a result of sodium vapor transfer from the reactor cover gas via the gap between the CRC and rotating plug (RP) and subsequent accumulation of sodium and its compositions in the CRC bearing assembly. Works on partial withdrawal of the CRC from the reactor vessel were scheduled in order to provide access to the bearing assembly for the sodium removal, and appropriate preparations were made. This task was unique, since no procedures of such a kind had ever been implemented on this type reactor in any country. During inter-maintenance interval, the program and the process of the CRC withdrawal from the reactor vessel were developed, and all necessary devices were designed and manufactured. CRC withdrawal confirmed that almost 100% of the bearing assembly surface was covered with sodium deposits filling the gaps between the bearing balls and cages. Besides, high density sodium deposits filled the gap between CRC and RP in the areas of seizure. Both cages and balls were withdrawn, sodium residues were removed from the bearing races and CRC/RP gaps, and new balls and cages were installed. Currently, obtained results and possible sources of this event are analyzed. The main conclusion is that owing to the comprehensive work, CRC availability has been restored and also experimental data of great importance for the fast reactors development and operation have been obtained.

4.1.3.4. Sodium leaks

In 20 cases the amount of poured sodium did not exceed 10 kg. In the rest 7 cases, the amount of sodium released were 30, 50, 100, 300, 300, 600, and 1000 L respectively. All leaks were detected in proper time by detection systems. All leaks took place in the auxiliary systems and caused only one reactor shut down. Table 2 gives the main characteristics of sodium leaks in BN-600 reactor. Over a total operation period (here means 24 years) 27 leaks into the environment of which 5 events involved radioactive sodium leaks have occurred. Fourteen events have involved passive burning of sodium.

TABLE 2. SODIUM LEAKS

System	Number	Quantity, L	No of Na burnings
Reactor	0	—	-
Intermediate heat exchanger	0	—	-
Storage drum	0	—	-
Primary auxiliary systems	5		-
—Gas purification	1	0.1	-
—Sodium purification system	4	0.3; 3; 0.2; 1000	1
Secondary circuit	18	—	-
—Main pipelines	0	—	-
—SG valve seals	4	1; 300; 30; 10	3
—SG leak detection system	1	2.0	1
—Drain and blow-off lines	10	0.2; 1; 10; 600; 300; 100; 0; 0; 1; 0000000000000000.0; 0.0;1.0	6
—Sodium storage	3	1.0; 0; 0	-
Sodium reception system	4	10.0; 50.0; 10.0; 0.0	3
TOTAL	27	~ 2 500	14

The main causes of sodium leakages were:

— Poor quality repair: 8 events;
— Latent defects of manufacturing and mounting: 6 events;
— Depletion of equipment lifetime due to inadequacy of the design: 7 events;
— Equipment design imperfections: 4 events;
— Human errors during operation: 2 events.

On 7 October 1993 the largest leak happened on the pipeline for removal sodium from the cold trap. The total amount of sodium escaped during the event was assessed to be approximately 800 kg. The first symptoms of the emergency were short circuits in the main and stand-by electric heating systems. Following actuation of the automatic alarm system by rising radioactivity in the exhaust ventilation air duct from the primary sodium purification system pipeline room, the supposed leaky section was isolated by the valves and the fire ventilation system was activated. These operated normally. Despite these measures a buildup of radioactive releases from the vent stack was registered as well as a deterioration of radiation conditions in the reactor building rooms, including the main control room. The release of radioactivity outside the plant was only 10 Ci. As a result, the primary sodium purification system was isolated completely and the reactor was shut down and put into the scheduled maintenance outage state a week earlier the assigned date. There was no overexposure of the plant personnel and no contamination on or off the site was registered.

All this gave grounds for classification of the event as an anomaly, i.e. level 1 to the International Nuclear Event Scale. After completion of the scheduled refuelling and the necessary repair the reactor was returned to power on 24 October 1993. The probable cause of a through crack in the joint between tubes with sodium flows of different temperatures could be high thermal stresses due to temperature cycling in combination with stresses caused by thermal expansion of the tubes. Despite the negligible radiological consequences it was concluded from analysis of the event that in the next BN type reactors external primary sodium pipework systems have to be excluded by arranging them inside the reactor vessel, or (if impossible) by providing complete jacketing in order to have two leak-tight protective barriers.

The largest leakage of secondary sodium happened in May 1994 in a drain pipeline (ID 48 mm). Approximately 600 L of sodium were lost, but only about thirty kilograms burned. The remaining sodium was retained and smothered with extinguishing powder in the catch system. In both leaks the protective systems were effective. The damage was not extensive and repairs were affected quickly. Of the 27 cases of leaks in only one case was it necessary to shut down the reactor;

4.1.4. Reactor core: design and operation

Due to the core fuel failures in the period from 1983 till 1987 the reactor was shut down six times for unplanned refuelling. Core design improvement became essential to provide operating reliability and safety of the reactor plant. Examination of the failed fuel revealed stress-induced corrosion of the annealed austenitic steel cladding as one of the main causes of early failure. The cladding was damaged mainly in the peripheral region of the core. It was due to the very unfavourable operating conditions for peripheral fuel assemblies. Because of reshuffling and rotation in the course of operation, the fuel rod linear heat rating and cladding temperature rose to 54 kW/m and 710°C respectively at the end of a fuel cycle. In the advanced core M design (the first core modification) the following changes had been implemented to improve conditions for fuel operation:

— The core active height was increased from 75 to 100 cm, decreasing the fuel rod maximum linear heat rating to 47.2 kW/m;
— Reshuffling and rotation of the fuel assemblies were eliminated;
— Swelling-proof cold-worked austenitic steel was used for the cladding and for the fuel assembly ducts.

By the end of 1987 the reactor core was completely assembled with advanced fuel subassemblies. Loss of cladding integrity events virtually terminated resulting in substantial reduction in fission product activity in the reactor gas plenum. The increase of caesium nuclide concentration in the primary system had also stopped.

In 1990–1992 the reactor core was changed to the advanced Ml design (the second core modification, Fig. 12). Ferritic steel was used in the new duct design and boron-modified cold-worked austenitic steel for cladding. Fuel burnup in the core has reached 10% h.a. with a fuel cycle length of 160 effective power days (efpd).

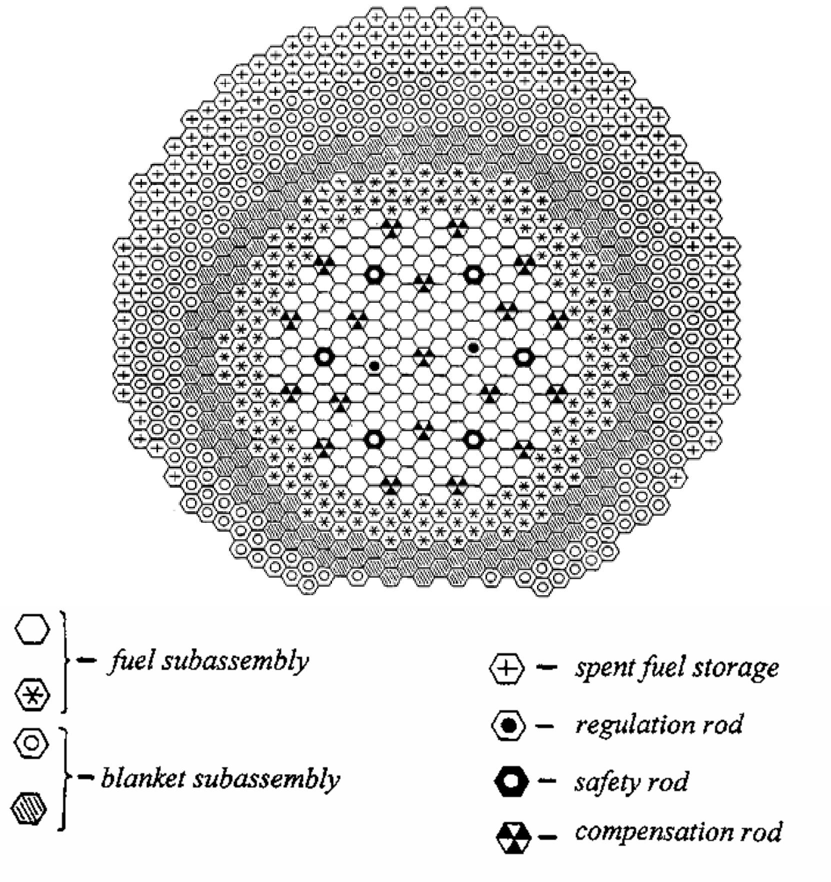


FIG. 12. BN-600 core.

Development of advanced radiation-resistant steels was (and is) the main problem in the attainment of higher fuel burn up. Considering the present status of this problem, a fuel burnup of 12% h.a. (90 dpa) is believed to be quite realistic for an advanced reactor core fuel subassemblies (Figs 13 and 14). This would give two options for the core refuelling pattern: to increase the refuelling interval from 160 to 190 efpd or to change to a refuelling pattern replacing 114 FAs at a refuelling interval of 145 efpd.

In either case the core reactivity margin will have to be increased, e.g. through expansion of the "medium" fuel enrichment zone at the expense of the adjacent FAs of the "low" enrichment zone. If the medium enrichment core zone were to be expanded to the limit of one FA row, the heat rating of the fuel rods would remain at an acceptable level ~ 48 kW/m.

The core of the first type was designed for quite high performance (original design, Figs 13 and 14, Table 3). However, as the initial operating experience showed, it turned out to be unreliable. Even during the first fuel cycles cladding loss of integrity events started, increased swelling of fuel assembly ducts and some control rod items was observed, as well as loss of ductility of the control rod guide tube material.

TABLE 3. BN-600 REACTOR CORE DESIGNS EVOLUTION

	Performances	Reactor core type		
		1	M	M1
1.	Reactor thermal output (max.), MW	1470	1470	1470
2.	Reactor core diameter, mm	2058	2058	2058
3.	Active core height, mm	750	1000	1030
4.	Axial blankets height, mm			
	Upper	400	300	300
	Lower	400	380	350
5.	Number of different fuel enrichment zones	2	3	3
6.	Fuel enrichment (U-235), %			
	LEZ	21	17	17
	MEZ	-	21	21
	HEZ	33	26	26
7.	Number of FAs in core zones			
	LEZ	209	136	136
	MEZ	-	94	94
	HEZ	160	139	139
8.	Core fuel cladding OD × wall thickness, mm	6.9×0.4	6.9×0.4	6.9×0.4
9.	Fuel rod length, mm	2400	2400	2400
10.	Fuel rod gas plenum length, mm	808	653	653
11.	Number of fuel rods in FA	127	127	127
12.	Duct width across flats × wall thickness, mm	96×2	96×2	96×2
13.	Core structural materials:			
	Cladding*	EI-847	EI-847	ChS-68
	Duct	Cr16Ni11Mo3	Cr16Ni11Mo3Ti	**
14.	Fuel rod maximum linear heat rating, kW/m	54.0	47.2	47.1
15.	Fuel rod cladding peak temperature, ºC	700	700	700
16.	Maximum fuel burnup, % h.a.	7.2	8.3	10
17.	Maximum radiation dose to cladding, dpa	43.5	53.9	75.0
18.	Fuel operating life, eff. d	200/300	300/495	480
19.	Core fuel cycle, eff. d	100	165	160
20.	Fuel inventory in core, kg	8260	11630	12090
21.	Average fuel burnup, MWd/kg U	42.5	44.5	60.0
22.	Temp./power reactivity effect, %, k/k	-1.4	-1.3	-1.3

* EI-847 - Cr16Ni15Mo3Nb - austenitic steel, ChS-68 - Cr16Ni15Mo2Mn2TiB - austenitic steel;
** Cr12MoBnVB-ferritic-martensitic-steel.

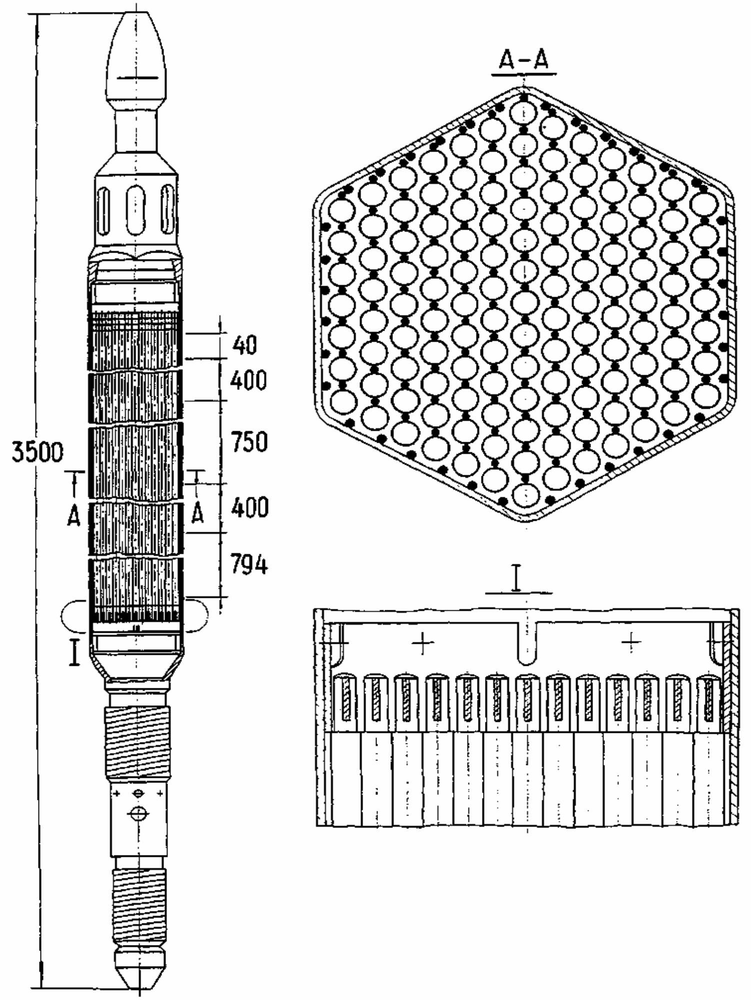


FIG. 13. BN-600 reactor core fuel subassembly (first type - the core active height 75 cm).

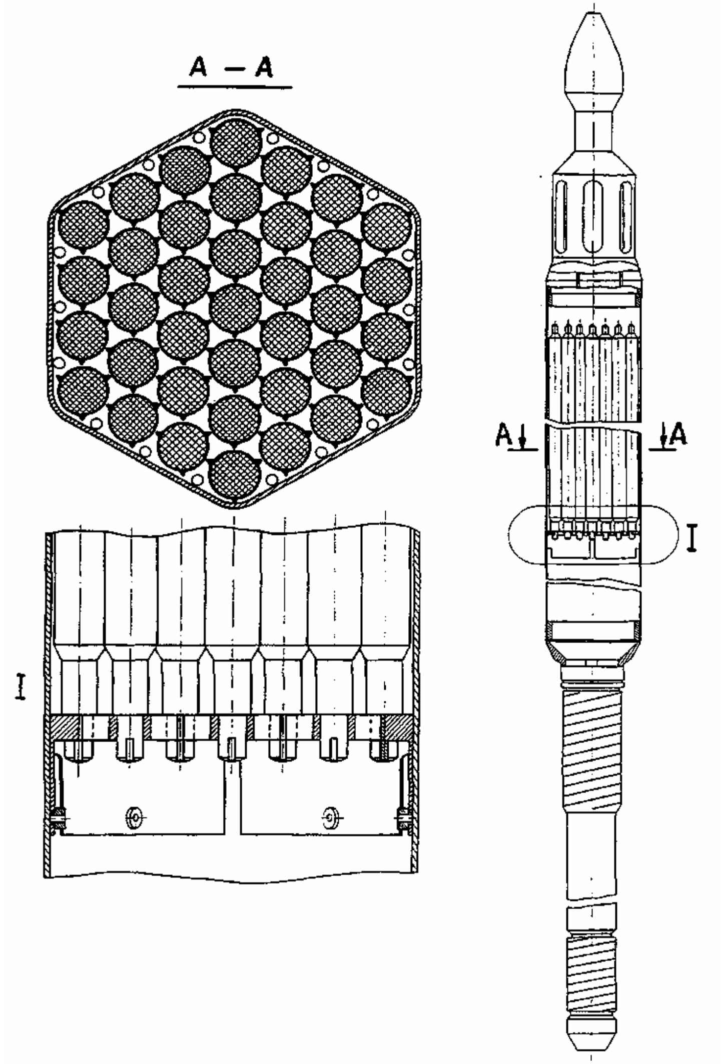


FIG. 14. BN-600 radial breeder subassembly.

4.2. BN-350 LOOP TYPE REACTOR PLANT

4.2.1. Design features

The first-in-the-world demonstration commercial liquid metal cooled fast breeder reactor BN 350 was designed and built by the former USSR, at present in the Republic of Kazakhstan. The BN-350 reactor plant is located near the city Aktau (former Shevchenko), Mangyshlak Region of Republic of Kazakhstan on the shore of the Caspian Sea (Mangyshlak peninsula). It was designed and constructed as a two-purpose pilot-industrial power plant (for electricity generation and heat production for seawater desalination) by organizations of the former Soviet Union under the supervision of the Ministry of Atomic Energy (MINATOM).

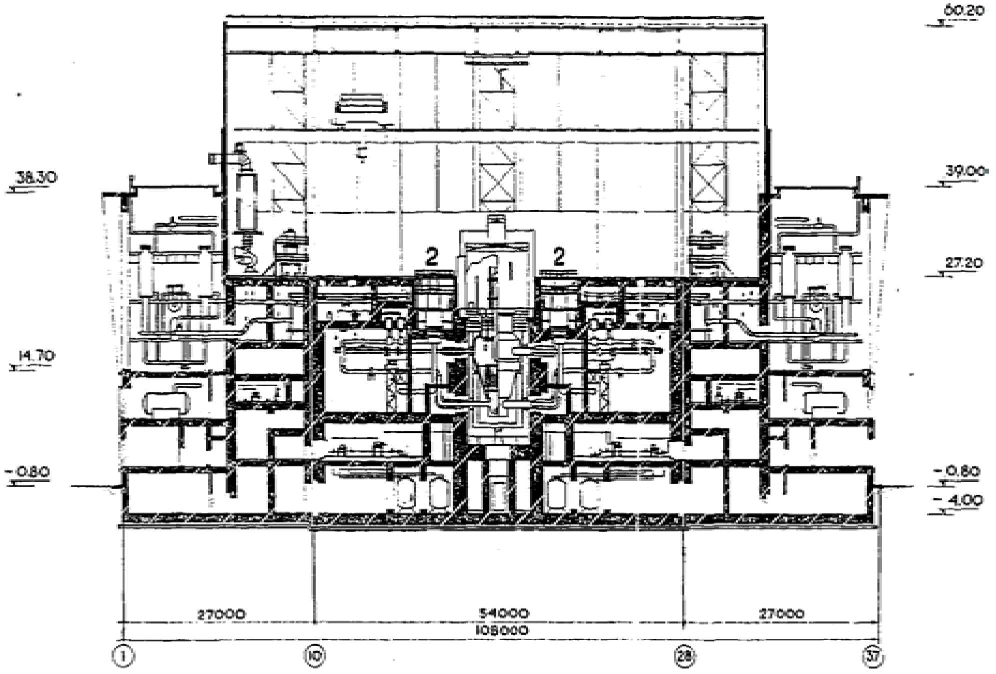
The BN-350 sodium cooled fast reactor has been operated by the Mangushlak Power Generation Company (MAEK) as a part of the industrial complex including the large capacity co-generation power plant and the distillate production works providing electricity and fresh water for Aktau and the adjoining industrial region, where natural fresh water resources were unavailable. A particularity of the plant was that the reactor for electricity production was physically separate from the desalination plant.

BN-350 has a dispersed (loop) arrangement of the primary circuit components, i.e. the primary sodium pumps, intermediate heat exchangers, and valves are housed in separate compartments (cells) and are connected with the reactor and by interconnected pipelines (Figs 15 and 16). The temperature expansions of the pipelines are accommodated by the bends. The reactor plant includes the following main components:

- Fast sodium cooled reactor, six primary loops;
- Six intermediate (secondary) loops;
- steam generators,
- Refuelling complex (integrated mechanical system);
- Primary and secondary sodium purification system;
- Automated process control system, including the reactor control and protection system (CPS);
- Diagnostic systems for monitoring the operating state of the safety related components and systems.

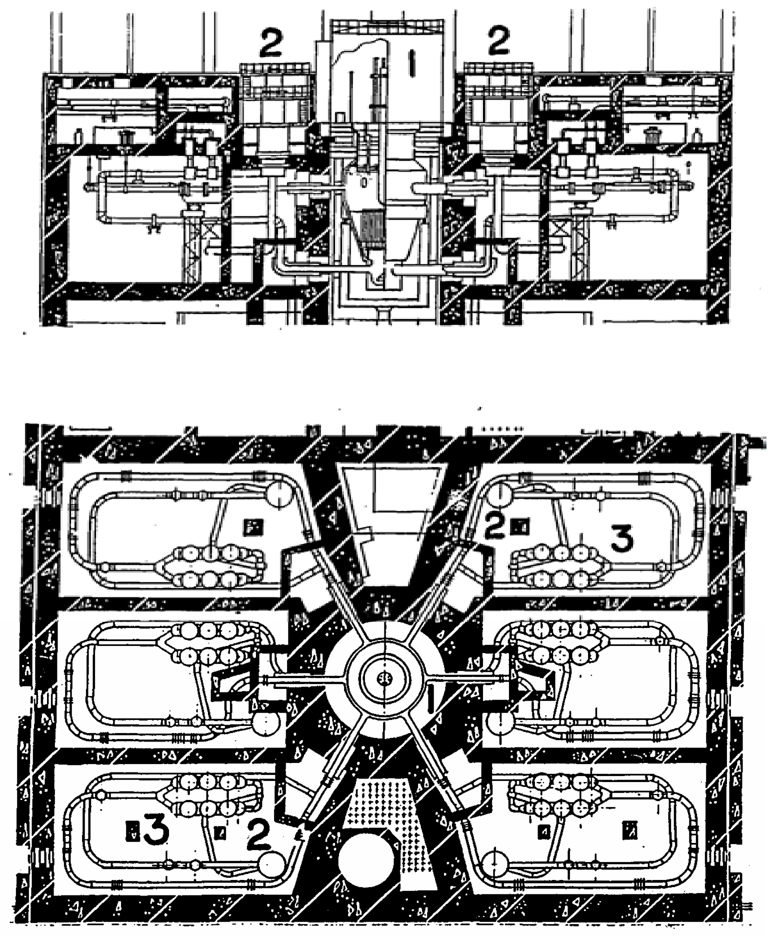
During power operation core heat removal and transport to the working medium (steam/water) are provided by a three-circuit flow scheme (Fig. 17). The primary circuit is composed of six intermediate heat exchangers (IHXs), six primary sodium pumps (PSPs), and sodium pipelines with gate and non-return valves. The pressure chamber with the core diagrid and the upper mixing plenum above the core are the common sections of primary sodium flow path.

The volume of the radioactive sodium in the primary circuit is equal to about 550 m^3, including ~ 150 m^3 in the reactor vessel, 170 m^3 in the pipelines of the primary six loops, 6 m^3 in six PSPs, 16 m^3 in six PSPs overflow tanks, 30 m^3 in five cold traps and ~ 130 m^3 in six IHXs.



1-reactor; 2-primary pump; 3-heat exchanger

FIG. 15. Vertical longitudinal section of BN-350 reactor.

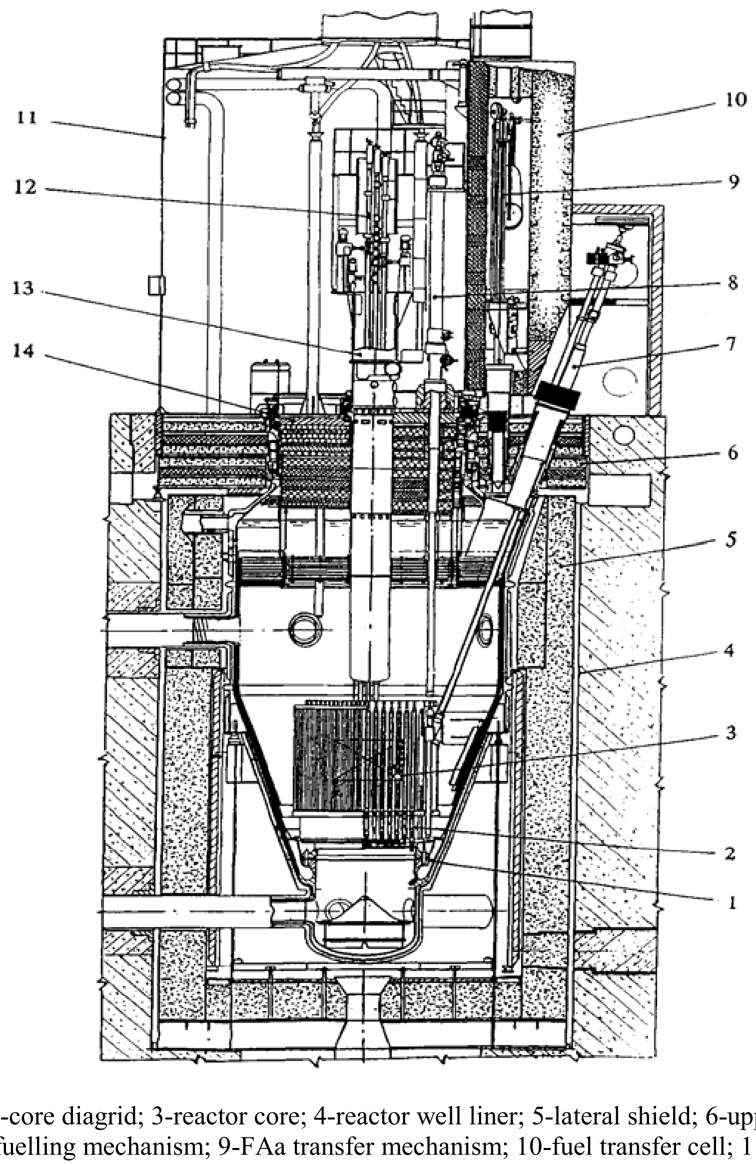


1-reactor; 2-primary pump; 3-heat exchanger

FIG. 16. BN-350 Nuclear island: vertical and horizontal section.

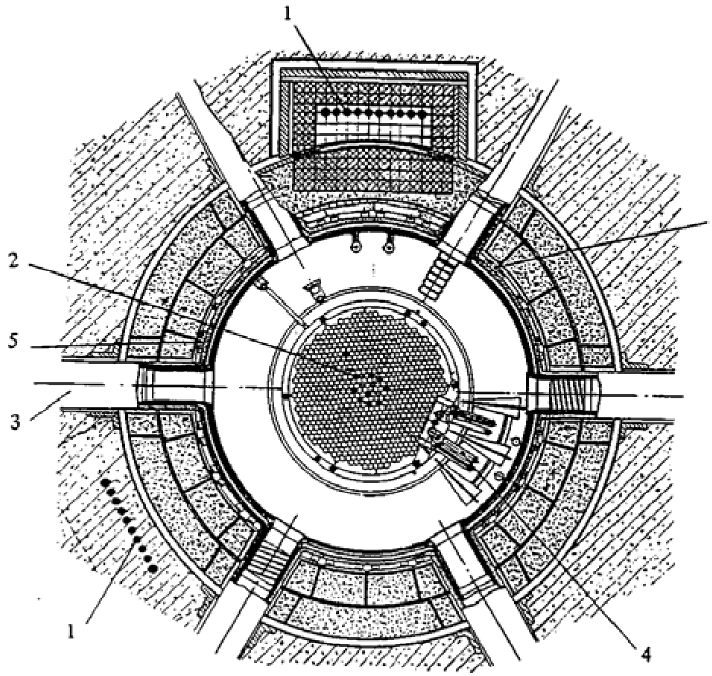
Sodium flow is distributed from the diagrid into the core and the radial blanket fuel assemblies. A portion of the primary sodium (250 t/h) is removed from the pressure chamber through throttles and utilized for cooling the reactor vessel and its outlet nozzles. There is a capability to isolate each primary loop from the reactor using two gate valves on the suction and pressure pipelines of the circuit. On the pressure pipeline of each loop downstream of the primary sodium pump (PSP) — a flap-type check valve is provided eliminating coolant backflow in the event of a PSP trip in one loop when the other PSPs are operative. Basic data for the BN-350 reactor plant are given in Table 4 for a nominal power of 750 MW.

The BN-350 reactor includes: the reactor vessel which contains the core diagrid with neutron reflector and a set of core and blanket fuel assemblies; the reactor refuelling system; the above core structure with CPS drive mechanisms and in-core instrumentation guides (Figs 17–19). The band and the core diagrid are made of Cr18Ni9 steel. Weight of the band with the shielding subassemblies is about 35 tons, weight of the core diagrid being about 17 tons.



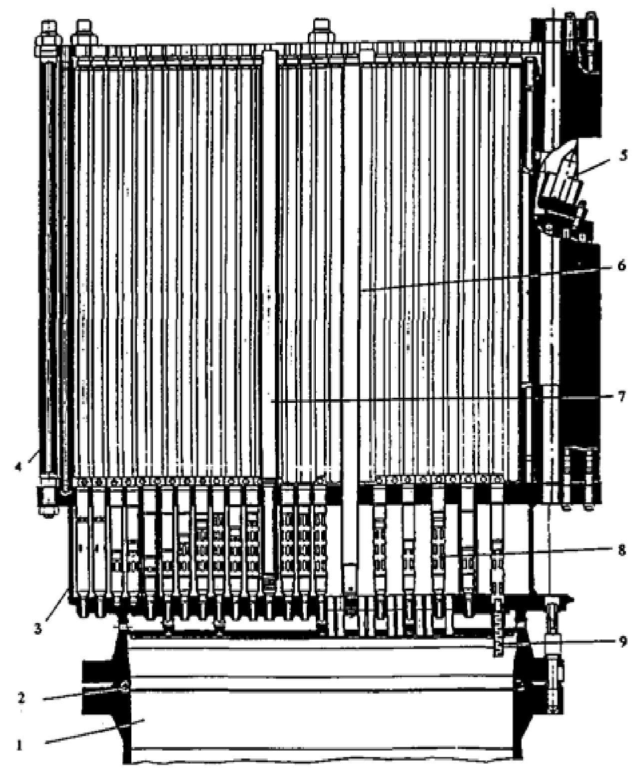
1-reactor vessel; 2-core diagrid; 3-reactor core; 4-reactor well liner; 5-lateral shield; 6-upper-stationary shield; 7-elevator, 8-refuelling mechanism; 9-FAa transfer mechanism; 10-fuel transfer cell; 11-protective dome; 12-control rod drive mechanism; 13-above core structure; 14-rotating plugs

FIG. 17. BN-350 reactor.



1-set of ionization chambers; 2-reactor core; 3-sodium outlet pipe;
4-elevator; 5-channels for additional ionization chambers

FIG. 18. Reactor plan view (cross section).



1-sodium pressure chamber; 2-sealing ring; 3-diagrid; 4-neutron reflector;
5-elevator lower support; 6-emergency protection rod sleeve;
7-reactivity compensation rod sleeve; 8, 9-throttling sleeves and devices

FIG. 19. Core diagrid.

TABLE 4. BN-350 BASIC OPERATING PARAMETERS

	Item	Value
1.	Reactor thermal output, MW	750
2.	Primary sodium temperature at reactor inlet/ outlet, ºC	288/437
3.	Sodium flow through reactor, t/h	141000
4.	Secondary coolant temperature at SG inlet/outlet, ºC	420/260
5.	Sodium flow in secondary loop, t/h	3400
6.	Number of operating loops (plus one-reserve)	5
7.	Main steam temperature / pressure, ºC/ MPa	405/4.5
8.	Steam flow, t/h	1070
9.	Maximum electric output of power unit, MW	125–150
10.	Distilled water output per day, t	100 000
11.	Maximum neutron flux in core, n-cm^{-2} s^{-1}	6×10^{15}

The diameter of the cylindrical part of the reactor vessel is 6 000 mm, and the wall thickness is 30 mm. In the middle section of the vessel there is a support belt by which the reactor is located on 16 roller bearings arranged on a support shell of 5 850 mm diameter transmitting the reactor weight load onto the foundations. The reactor vessel is enclosed in a guard vessel. Thermal shielding, main reactor vessel, reactor guard vessel and supporting ring are manufactured of Cr18Ni9 steel. Weight of thermal shielding of the reactor vessel is about 53.9 tons, and weight of the main vessel with the guard vessel and supporting ring is about 101.5 tons. The rotating plugs, ~ 2.5 m thick, have multilayer structure formed of steel sheets, graphite and thermal insulation. Under the plugs the shielding is located made of Cr18Ni9 steel sheets of total weight of ~ 94 tons.

Outside the reactor vessel (in the reactor cavity) there are 7 650 mm high 3-grade steel lateral shielding consisting of 160 mm thick plates, located inside iron ore filling shielding. Iron ore filling is placed in the 3-grade steel reinforcing cage, having 20 mm thick walls; total weight of the iron ore filling is ~ 1 303 tons. The next to it is the concrete covered with 20 mm thick 3-grade steel sheet lining (Figs 18 and 19). Under the vessel similar shielding structure is provided. The total weight of 3-grade steel shielding is ~ 344 tons.

4.2.2. General results of operation

The BN-350 plant history is as follows (Fig. 20):

- 1965–1971: construction period;
- 29 November, 1972 first criticality of the reactor;
- 16 July 1973: power startup of the reactor. The extended start up was due to loss of integrity events in four evaporators (detected by the appearance of hydrogen in the gas plenum) when the SGs were filled with water;
- end of 1973 – February 1975: SGs repair; 1973–1975: operation at power levels up to 300 MW(th);
- from March 1975: operation at 650-750 MW(th) for electrical power [~ 150 MW(e)] generation and sea water desalination (~ 100 000 tons of desalinated water per day);
- from January 1996 to June 1998: operation at 420 MW(th), 50 MW(e), producing 45 000 tons of distilled water per day;
- April 1999: final shutdown.

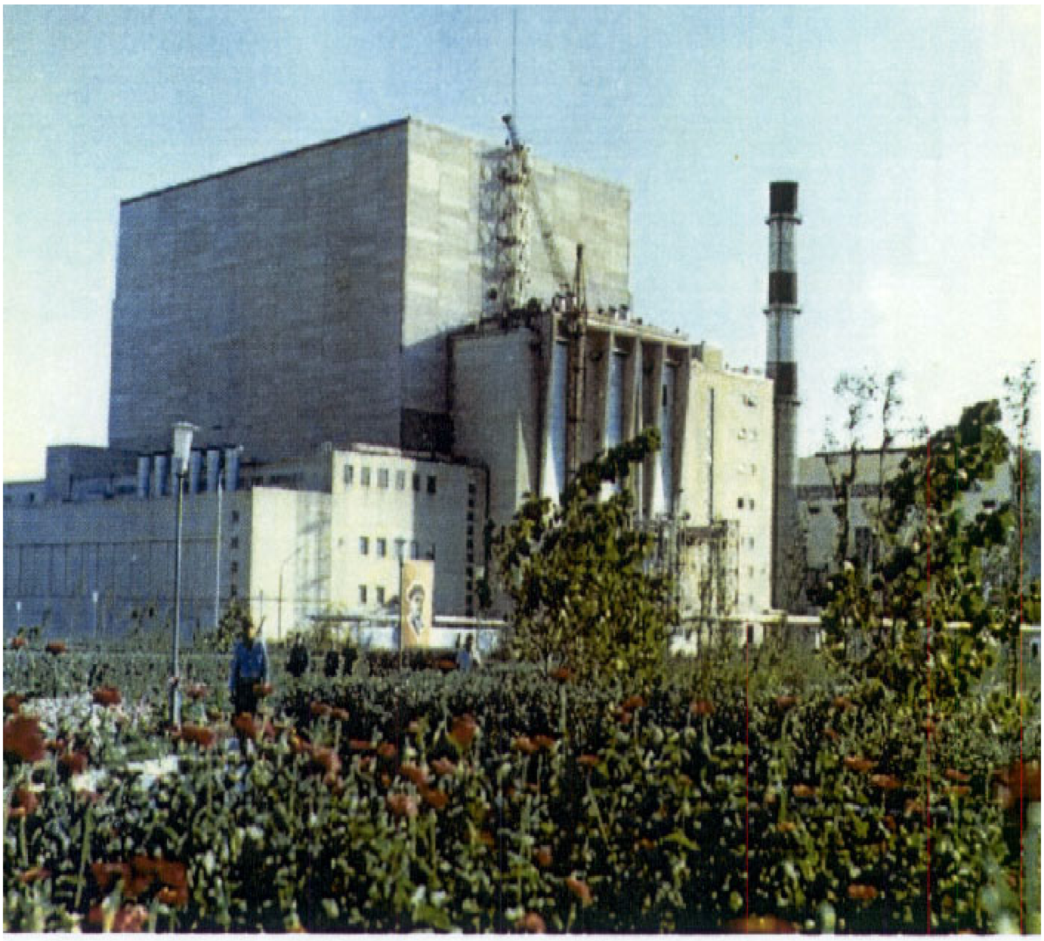


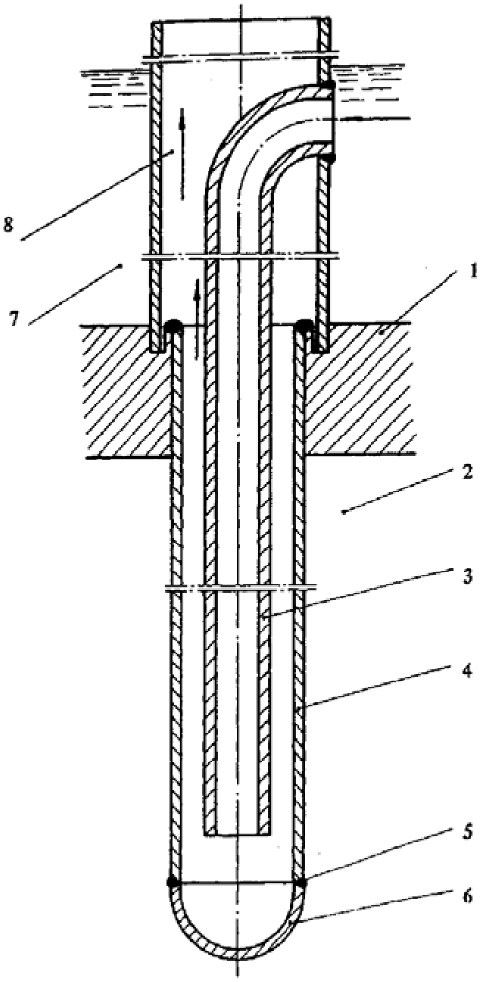
FIG. 20. BN-350-a freshwater source in the desert: overall survey.

For more than twenty five years, the operation of the power unit associated with the BN-350 reactor have promoted the exploration of the new industrial region of Kazakhstan which is rather rich in natural resources.

4.2.3. Steam generators and intermediate heat exchangers: operating experience

The BN-350 steam generators consist of two super-heaters with U-shaped tubes and two evaporators with re-entrant tubes inside which water flows under natural convection and partial evaporation conditions. Main data for the BN-350 SG (evaporator) is given in Table 1. In general, the major plant components had been tested prior to assembly, and had resulted in reliable operation in the main. However, it had been decided not to test the steam generators for reasons of expense. The initial period of reactor plant operation was characterized by unreliable operation of the SGs. Numerous loss of integrity events occurred in the re-entrant evaporator tubes.

Metallographic examination of a great number of re-entrant tubes showed the presence of microcracks in the tube-to-bottom weld joints (Fig. 21). Mechanical deformation of the tube bottoms during cold stamping were acknowledged as the most probable cause of the microcracks. Growth of the microcracks could occur under the effect of internal stresses arising during welding the bottoms to the tubes and under cyclic thermal loads during evaporator operation.



1-tube sheet; 2-sodium; 3-downcomer tube; 4-heated outer tube; 5-lower weld seam (cracks);
6-bottom of Field tube; 7-boiler water; 8-steam-water mixture outlet

FIG. 21. BN-350 evaporator re-entrant tube.

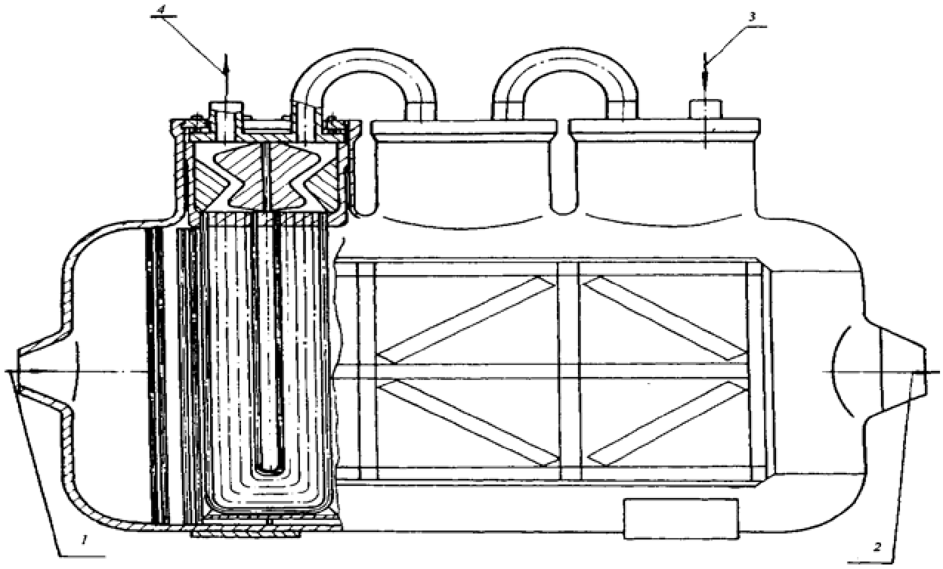
After repair of the re-entrant evaporators when outer tubes of 32×2 mm (OD×wall thickness) were replaced by 33×3 mm tubes with machined bottoms, reactor plant operation was continued with five loops at thermal power of 650 MW. The superheaters had caused no problems, but when some of the peripheral tubes were found to be vibrating they were plugged as a precautionary measure. The availability of stand-by components (six loops) and the ability to operate with different numbers of loops (from three to five) provided for stable operation of the reactor plant and production of electricity and fresh water. As a result, repair and replacement work on the steam generators (as necessary) were carried out without shutting down the reactor.

One of the repaired evaporators failed later due to a large leak of water to sodium. It was dismounted and replaced with the micromodular SG “Nadjozhnost-1” of Czech fabrication (1980). Another SG of the same type “Nadjozhnost-2” was put into operation in 1982 instead of one of the re-entrant SGs which had been operated reliably since the reactor start up. In general, the plant operation demonstrated the reliability and high operating performances of the re-entrant tube SG design taking into account particularly their behavior in water to sodium leak events.

The thermal-hydraulic disadvantage of the design was revealed by natural convection flow instability during emergency residual heat removal. The use of “Nadjozhnost”-type steam generator eliminates this disadvantage and in addition allows the removal of heat from the SG by air flowing outside the modules under forced or natural convection conditions. In January 1989, both these SGs failed due to stress-induced corrosion on the steam-water side of the tubes promoted by non-uniform thermal-hydraulic conditions in the tube bundles. After repair the SGs were put again into operation in 1993.

The reactor was designed for thermal output of ~ 1000 MW, but in the early periods of operation its power level was limited by unsatisfactory operation of the steam generators. To give of less demanding conditions for the operation of evaporators with re-entrant tubes a maximum power of 130 MW(th) was specified for each SG. For the Czech-produced SGs “Nadjozhnost” the allowable power level was set somewhat higher — at 200 MW(e). In addition, from an experiment carried out in 1976 on plant emergency cooling with loss-of-normal power it was found that the available capacity of the steam/water system under these conditions restricted reactor to 750 MW(th). This power level was not exceeded during subsequent power operation periods, taking into account the power limitations imposed by the SGs as well. During its operating life the reactor has operated at various power levels.

The average load factor in respect to allowed power levels was 85%. Reduction in the load factor was due to outages for refuelling and planned maintenance of the equipment. The reactor was normally shutdown for scheduled refuelling and maintenance two or three times per year with outage durations of 20-30 days. Horizontal tube-and-shell IHXs with three modules connected in series, made of U-shaped tubes were used in the BN-350 (Fig. 22).



1, 2-primary Na inlet/outlet, 3, 4-secondary Na inlet/outlet

FIG. 22. BN -350 intermediate heat exchanger.

The IHX of each loop consists of two sections connected in parallel both for primary and secondary coolant flows. The IHX is located in a suction loop upstream of the primary coolant pump, while in the secondary circuit it is in a pressure loop downstream of the circulating pump. The IHX tube bundles can be removed if necessary and replaced with new ones. The most stressed units in the IHX are the fixing joints for the tube module covers and for the frame which stiffens the flat walls of the IHX body. Measurements of temperatures and stresses in various items of the IHX were carried out during reactor plant operation. On this basis requirements were formulated to limit the rate of the IHX heating-up in steps of 10%

specified power with delays of 5–10 h in each step. By 1998 the IHXs have operated more than 180 000 h at various power levels without any disturbances and failures.

Radiation doses to the BN-350 personnel have been associated mainly with repair and maintenance activities for sodium components in the primary circuit boxes. Radiation conditions in the rooms (except the reactor well) depended substantially on the state of fuel rods in the core. In the initial period of the reactor plant operation when the first type core was used numerous cladding failures caused a significant rise of fission products activity in the primary system. In 1979 the gamma radiation dose rate on the surface of the sodium equipment at reactor shutdown reached 8.90 μSv/s, 80% of which was attributed to caesium nuclides. Radiological conditions were improved significantly after completion (in 1979) of the core change over to the second type fuel design. This was provided not only by reduction in the number of failed fuel rods, but also by clean up of caesium from the primary coolant using a special trap with a graphite absorber. Since 1984 the gamma radiation dose rate in the primary circuit rooms during outages has not exceed of 1.5 μSv/s.

Due to the loop design of the reactor primary system the reactor well is virtually inaccessible even at outages because of high radiation-induced activity of the reactor vessel structures. Transport of gaseous fission products into the upper part of the control rod drive mechanisms and to the oil system of the reactor coolant pumps turned out to be one of the reactor operating peculiarities, which degraded radiological conditions in the reactor servicing zones. The major source of the reactor radiological impact to the environment was gaseous discharges from the equipment air cooling system and from the reactor plant rooms through the vent stack. Improvements in the fuel design and the associated reduction in the number of failed fuel rods in the core (which became quite rare in later operation periods) resulted in radioactivity of the plant discharges to the atmosphere being determined by the radiation-induced activation of air in the reactor well cooling system. Daily release of gaseous nuclides was 0.55-0.74 TBq while that for aerosols was 1.9×10^{-6} TBq. Observations over many years of radioactivity of the flora and fauna, and radiological conditions in the local populated areas and in the sanitary restricted zone around the NPP, showed that those characteristics affected by natural and man-induced radiation sources corresponded to background radiation levels.

4.2.4. Reactor core and refuelling equipment: design and operating experience

The reactor central part contains a set of core and blanket fuel assemblies (FAs) guide sleeves for CPS rods and neutron reflectors installed in the diagrid (Figs 23 and 24). The diagrid is attached to the sodium pressure chamber. The internal plenum of the diagrid is divided into two chambers: high and low pressure. The sodium from the high-pressure chamber is distributed for cooling the core FAs, control rods and FAs of the inner blanket. From the low pressure chamber the coolant goes for cooling FAs of the outer blanket. Core FAs transferred to the internal store around the outer blanket periphery are cooled by natural convection of coolant in the reactor vessel. During the initial period of the reactor operation until 1979, when the first design core (fuel rod of 6.1 mm OD) was used, large number of fuel failures (loss of clad integrity events) occurred. These fuel failures caused a significant increase in fission fragment activity in the primary circuit and consequently resulted in deterioration of radiation conditions in the reactor plant cells. Therefore the second design of core fuel assembly was developed with fuel rods of 6.9 mm OD. This advanced core provided for increased fuel burnup and more reliable operation of the fuel rods, mainly due to the following improvements:

— The gas plenum height in the fuel rod was increased at the expense of integration in one clad tube (6.9×0.4 mm) of core and axial blankets material (fissile and fertile) and reduction of the lower blanket height;
— The fuel assembly duct material Cr18Ni10Ti (austenitic steel) was replaced by stabilized austenitic steel Cr16Ni11Mo3 in a heat-treated state;
— The coolant pressure in the middle plane inside the duct was diminished by approximately 35% resulting in a decrease in duct deformation by radiation-induced creep;
— The power distribution over the core radius was flattened by the incorporation of a medium fuel enrichment (21%) zone between the existing core zones with "low" (17%) and "high" (26%) enrichment of fuel, resulting in a decrease of the fuel rod specific heat rating.

The measures mentioned above and reactor operation under power conditions resulted in a significant reduction in the number of defective fuel rods in the core. Further increase in fuel burnup was achieved through utilization of the ferritic steel EP-450 for ducts and improved austenitic steel in cold worked state for cladding.

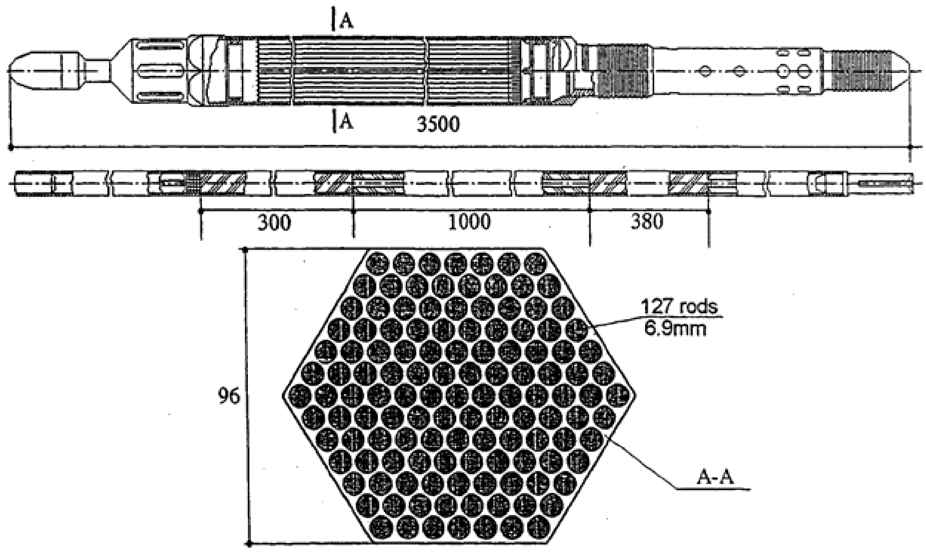


FIG. 23. BN-350 fuel subassembly and rod.

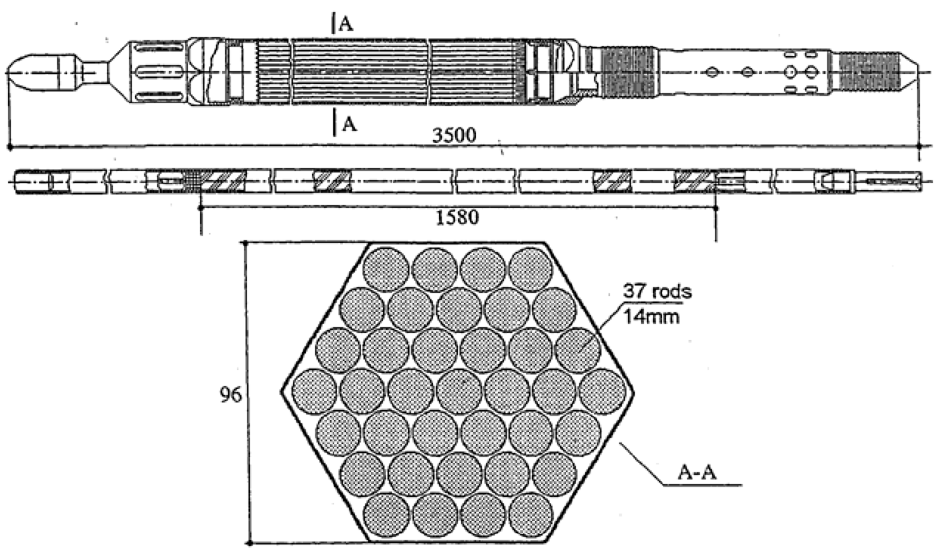


FIG. 24. BN-350 blanket subassembly and rod.

The BN-350 refuelling system consists of two components:

— Complex of in-reactor refuelling mechanisms (Fig. 25);
— Complex of out-of-reactor refuelling systems (Figs 26 and 27a,b).

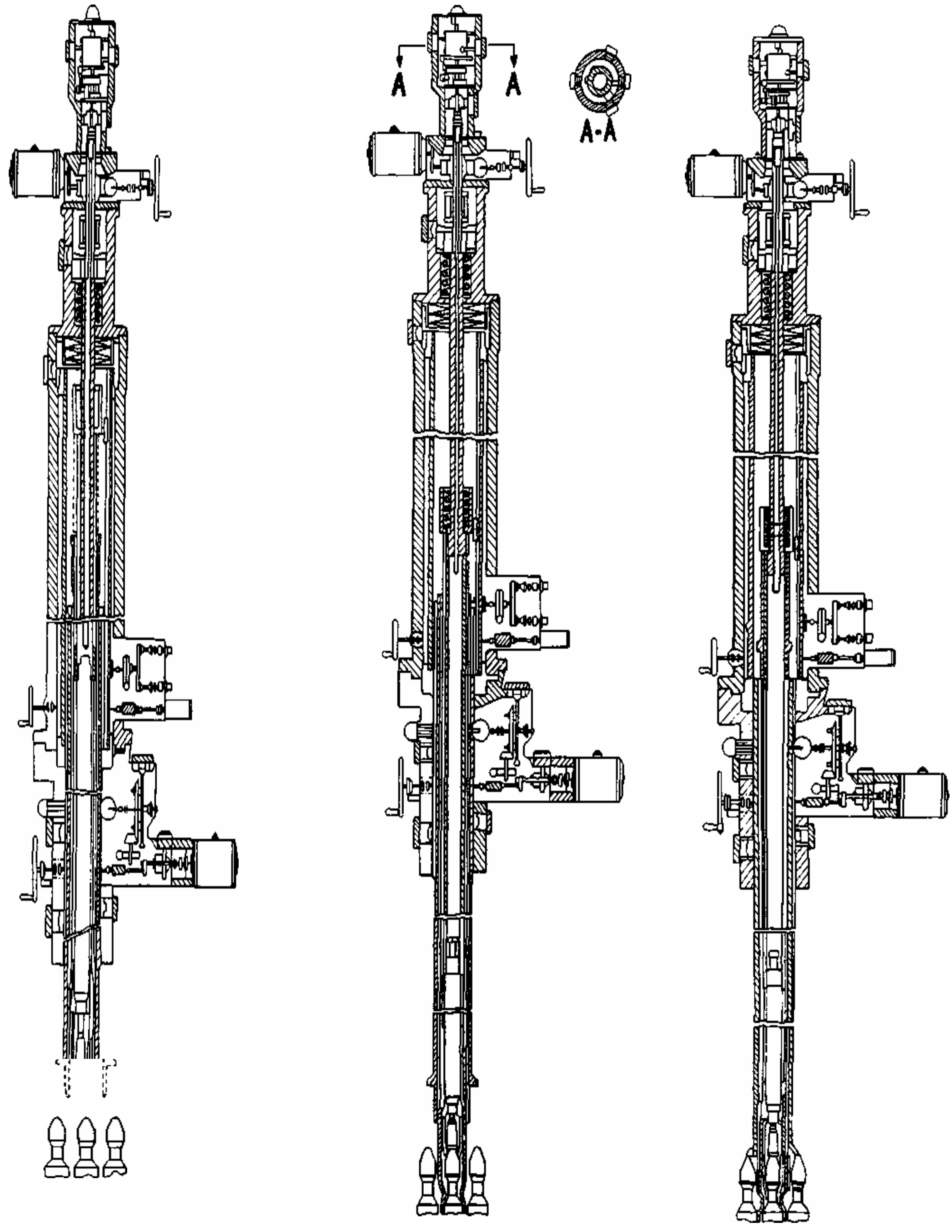


FIG. 25. BN-350 refuelling mechanism-mode of operation.

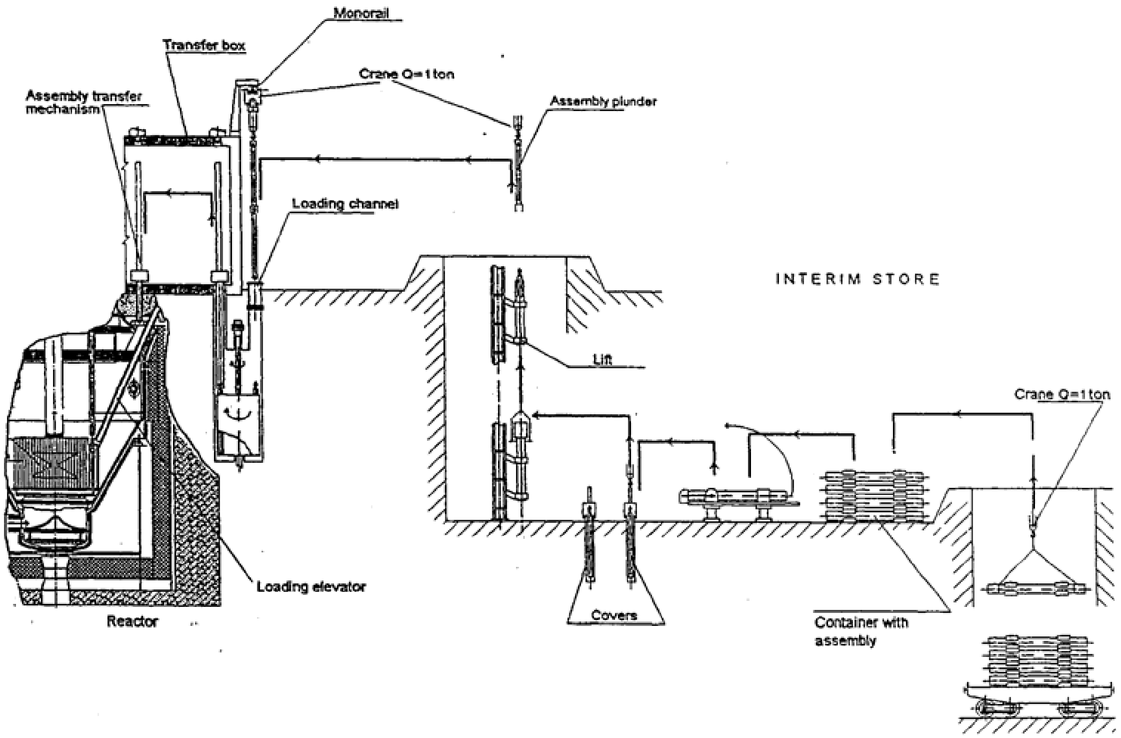


FIG. 26. Fresh subassemblies loading principle.

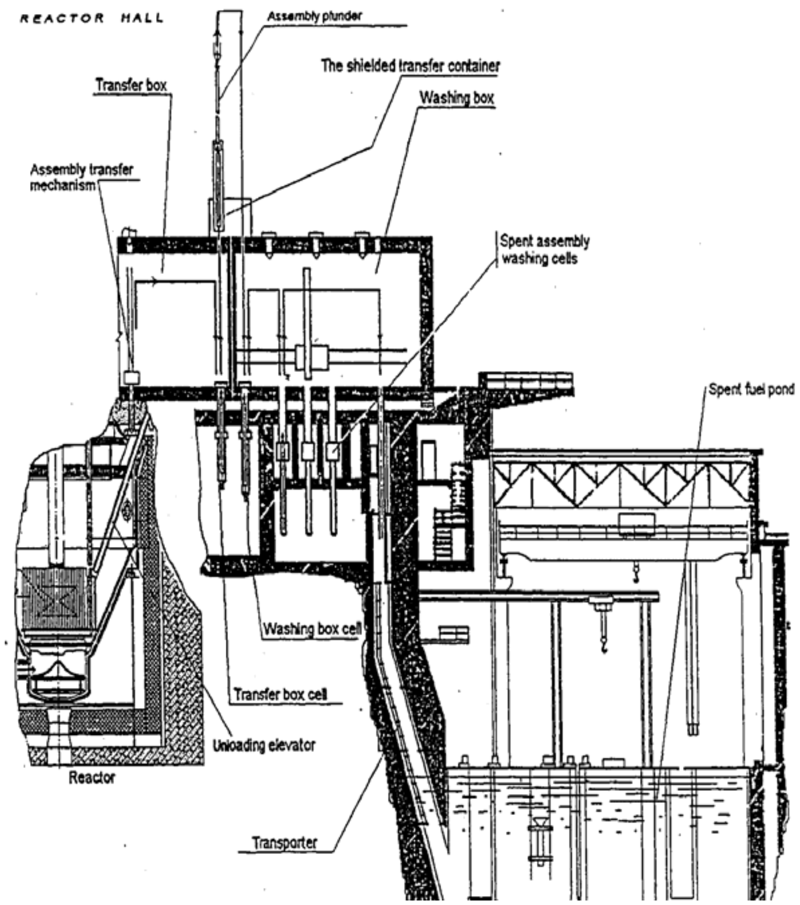


FIG. 27a. Spent subassemblies unloading principle.

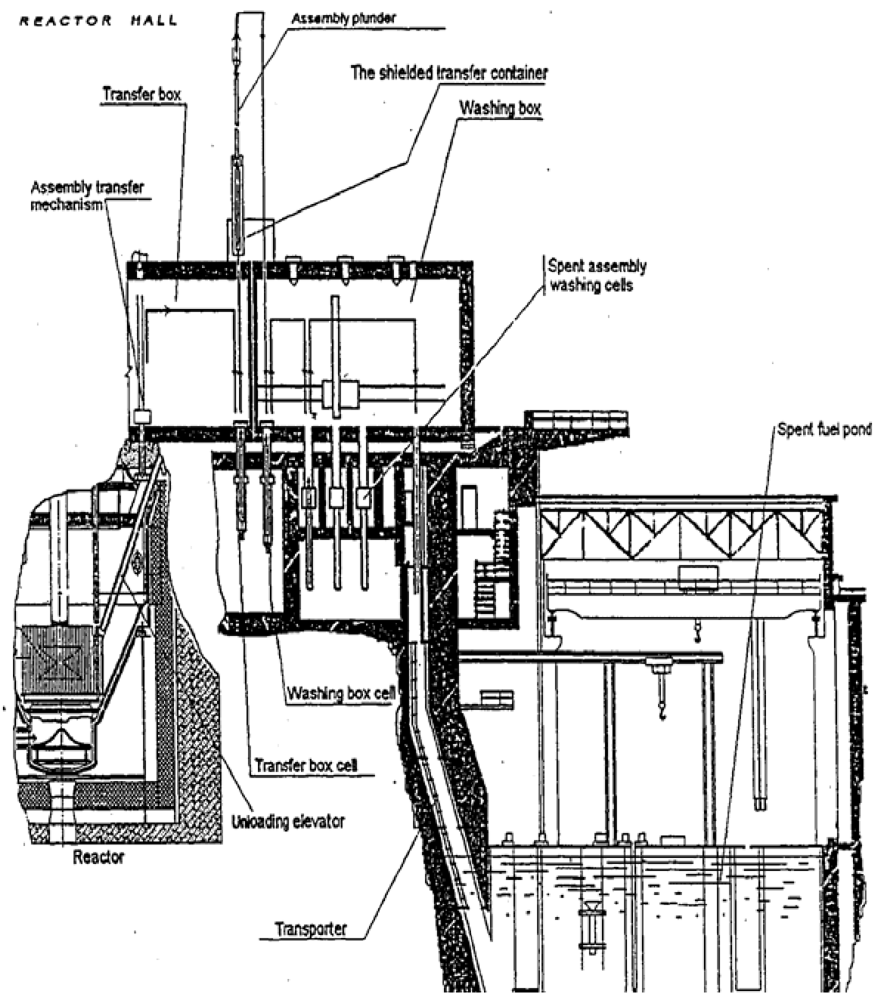


FIG. 27b. Spent subassemblies unloading principle.

The design parameters of BN-350 core and blanket assemblies are presented in Tables 5 and 6, respectively. The first complex is a constituent of the reactor and includes: two rotating plugs (large and small), refuelling mechanisms and loading-unloading elevators. This complex provides for loading and unloading of the main core items and their rearrangement inside the reactor. The rotating plugs are disposed eccentrically to each other. On the small rotating plug, the refuelling mechanisms are mounted eccentrically. By rotating the plugs the refuelling mechanism is guided to any position in the core, radial blanket and in-vessel store, or to any control rod. The refuelling mechanism simultaneously with reshuffling of FAs rotates them for alignment with the hexagonal cells in the core. The complex of out-of-reactor refuelling systems includes:

— Fresh FAs drums;
— Fuel transfer cell transfer mechanism;
— Spent fuel drum;
— Washing cell fuel transfer mechanism;
— Loading-unloading elevator plug lifting mechanism.

TABLE 5. DESIGN AND PARAMETERS OF BN-350 DRIVER ASSEMBLIES

Parameter	Assembly type		
	Type I	Type II	Type III
Years of usage	1974-1980	1978-1990	1987-1998
Hexagonal duct material	12-18Cr-10Ni-Ti	08-16Cr-11Ni-3Mo	12-13Cr-2Mo-Nb-V-B
Duct flat-to-flat distance × wall thickness, mm	96×2	96×2	96×2
Number of fuel elements	169	127	127
FE length, mm	1140	1790	2440
FE diam. × wall thickness, mm	6.1×0.4	6.9×0.4	6.9×0.4
Cladding material	16Cr-15Ni-3Mo-Nb	16Cr-15Ni-3Mo-Nb	16Cr-15Ni-2Mo-2Mn-Ti-V-B
Fuel column height, mm	1060	1060	1000
Gas plenum length, mm	25	250	720
Burnup, %	5.6-5.8	13	13
Dose (displacements per atom)	50-60	85	90

TABLE 6. DESIGN PARAMETERS OF BN-350 BLANKET ASSEMBLIES

Parameter	Assembly type	
	Type I	Modernized
Years of usage	1974-1992	1992 onwards
Hexagonal duct material	12-18Cr-10Ni-Ti	08-16Cr-11Ni-3Mo
Flat-to-flat distance × wall thickness, mm	96×2	96×2
Number of fuel elements (FEs) per assembly	37	37
FE column height, mm	2492	2492
FE Cladding diameter × wall thickness, mm	14.0×0.4	14.0×0.4
Cladding material	16Cr-15Ni-3Mo-Nb	16Cr-15Ni-3Mo-Nb
Fuel pellet outer/inner diameter, mm	13.0/0.0	13.0/0.0
Fuel length, mm	2410	1580
Gas plenum length, mm	18	18
UO_2 enrichment (^{235}U, %)	0.4	0.4

Spent FAs and other items are transported into sockets in the washing cell and then transferred into the spent fuel water pool for decay. The fuel transfer cell with its transfer mechanism and the washing cell adjoin the reactor. New fuel assemblies are loaded into the reactor from the new fuel storage drum located beneath the refueling cell. New fuel loading operations have not posed any difficulties during the reactor operating life. The design intent was that irradiated fuel assemblies were to be transported from the fuel transfer cell to the washing cell through the spent fuel storage drum located in a tank filled with sodium-potassium alloy beneath the junction between the cells.

In the BN-350 reactor, there was an incident (1976) causing the fuel storage drum failure. The cover (plug) of the drum included concrete filler. In the course of heating, vapor of crystallization water from the concrete penetrated into sodium-potassium alloy, filling the drum, and, upon interaction with this alloy, chilled the drum. BN-350 personnel developed special technology to dissolve the formed conglomeration using water-oil emulsion, and as a result of its implementation, fuel subassemblies stored in the drum were set free and placed in the water pool. A special lead-shielded flask was designed and manufactured for transporting spent fuel assemblies from the transfer cell to the washing cell.

It was noticed during operation that increased forces were required for rotation of the shield plugs. Probable causes were sodium vapor condensation in gaps or nonuniform heating of the hydraulic seals. Plug operation became normal after increasing the Pb-Bi eutectic working temperature to 200°C and temporary interruption of the plug air cooling system in the process of heating up the hydraulic seals. The refuelling mechanisms, elevators and fuel transfer mechanisms in the cells have been operated without significant disturbances. Minor disturbances were remedied through replacement and modernization of individual items.

To improve the reliability of the mechanisms a system to control installation and withdrawal forces was incorporated. During the entire period of reactor operation (till October 1995) 56 planned refuelling cycles have been fulfilled. The time spent for one fuel assembly replacement was one hour. The total number of loading-unloading operating cycles of the elevators and fuel transfer mechanisms was ~ 3 200. Minor disturbances were remedied through replacement and modernization of individual items. To improve the reliability of the mechanisms a system to control installation and withdrawal forces was incorporated.

The first-in-the-world demonstration commercial liquid metal cooled fast breeder reactor BN-350 was designed and built by the former USSR in the Republic of Kazakhstan for an estimated 20 year life-time, which was reached in 1993. Considering the importance of the reactor for the region, both as a source of desalinated water and electricity, it was the intention of the Kazakhstan Government to extend the useful life of the reactor and continue to operate it as long as safety conditions permitted.

In 1995, as a result of its limited experience in decommissioning, and its limited financial resources, the Republic of Kazakhstan requested IAEA's assistance in improving safety for the extension of the life of the reactor and in preparation of the proper technological procedures for the eventual decommissioning. To this end, the IAEA organized a mission to review the safety status of the reactor and a Consultancy (CS) on "Harmonization of international assistance to ensure stable operation during the remaining life-time and the development of an effective decommissioning programme for the BN-350 fast reactor" in October 1996.

A composite programme for the prolongation of operation, and safety assurance of the BN-350 plant, which included modernization and replacement of some reactor systems, was developed by design organizations of the Russian MINATOM and the Kazakh Atomic Energy Committee (KAEC). Some features to improve reactor safety were installed; but financial difficulties delayed full implementation. Taking into account the large expenditures needed for fulfilment of the programme, a Government decision to shut down the BN-350 plant was taken in April 1999: that is, after ~ 25 years of reliably supplying fresh water and energy to Aktau (former Shevchenko) and the adjoining region, a large city and industrial centre in the desert, that had quickly grown following construction of the reactor.

4.2.5. BN-350 electromagnetic pumps

Recently the investigations on electromagnetic pumps in fast reactor secondary circuits have been in progress. In the former USSR this investigation was carried out as applied to the BN-350 plant. Electromagnetic pumps (EMP) are widely used for liquid-metal coolant pumping over the booster circuits of fast reactors. A certain positive operating experience was provided concerning the main circuits of the research reactors BR-10, EBR-II and the secondary circuit of the BOR-60 reactor plant. The electromagnetic pumps fabrication and operation experience for the flow rate up to ~ 1000 m^3/hr provides a possibility of activities on designing higher-power pump with the flow rates of several thousands m^3/hr.

The development of these pumps was supported by a system of theoretical calculations and experimental and design studies performed in D.V. Efremov's NIIEFA Institute in St. Petersburg (Russian Federation). A cylindrical induction pump with the metal flow turn through 180° was adopted for operation in the main fast reactor circuits. This design provides the dismounting and assembling of the inductor related to its repair or replacement without circuit loss-of-tightness.

The secondary and primary circuit EMP with the loop configuration are set in the circuits pipelines through, inlet and outlet connections and do not need dipping into sodium. The inductor coil cooling-down is substantially simplified due to this. The pump of this design-TSLIN-3/3500 was developed and then fabricated for 3500 m^3/h flow sodium pumping, under the pressure 0.3 MPa and the temperature 350°C in one of the loops of the BN-350 reactor secondary circuit. Its key design parameters are: number of poles - 12; pole pitch - 0.156 m; channel height - 26 mm, average diameter of channel – 0.95 m, overall dimensions: height - 5 m, diameter - 1.8 m. The total test data were conducted at a liquid-metal rig at the sodium temperature 300°C in 1986; processing was completed in 1987. The nominal operating conditions are specified by the following parameters:

— Flow rate Q = 3600 m^3/h
— Developed pressure P = 0.3 Mpa
— Intake power ~ 1 MW
— Voltage V = 650 V
— Current I ≈ 3000 A at f = 50 ha (c.p.s)
— Cos φ = 0.3

The experimental values are fairly close to the calculated ones (the calculation error is as high as 9%, for the efficiency at the same flow rates - 3.5% absolute).The thermal tests of the pump have shown, that the maximum temperature of coil measured by thermocouples on the coil surface is 280°C, and the average temperature of coil measured in the resistance method is ~ 240°C. The corresponding calculated temperatures are ~ 295 and ~ 260°C, respectively, that very close to the experimental ones. The mentioned operating temperatures of coil with the utilization of material brought to a commercial level provide the required lifetime with a high degree of reliability. In addition, a series of tests were performed to confirm inherent requirements of operation in fast reactors. In the course of these tests the excessive inlet pressure (top section of the channel) changed from 1.1 to 0.35 kg/cm^2 at the flow rate 3750 m^3/hr (the rate in the active section of the channel is ~ 13.5 m/s). In this case no deviations in the pump parameters, vibration values were observed. In. the whole it was established, that the pump satisfies the requirements on non-cavitational operation in the nominal and partial operating conditions. The general view of the fabricated pump is represented below. Its key design parameters are: number of poles - 12; pole pitch - 0.156 m; channel height - 26 mm, average diameter of channel - 0.95 m, overall dimensions: height -5m, diameter -1.8 m.

4.3. POOL VS. LOOP TYPE DESIGN: SOME KNOWLEDGE GAINED FROM CONSTRUCTIONS AND OPERATIONS

Some experts believe that future sodium cooled fast reactors will have a pool type arrangement of the radioactive circuit. It was chosen now in all countries dealing with these reactor types except for Japan. The advantages of the pool type design become especially apparent at present when safety requirements for all nuclear power plants become more and more stringent. The Russian Federation is the only country where relatively large power loop type (BN-350) and pool type (BN-600) reactors have been designed and successfully operated (BN-350 in Kazakhstan) for a long time.

Experience has shown that a loop type design of the primary circuit is simpler and faster in construction. However, this design type has some inherent specific difficulties in solving safety problems. The main of them are assured preclusion of coolant leakage from the reactor at large ruptures of the primary circuit and high reliability tightness of equipment and piping cells to preclude radioactive sodium burning at small leaks. The problem of reliable protection of both the reactor and personnel at radioactive coolant leaks is one of the most important for the loop type design. As a result of long-time, almost 25-years operation of the BN-350 loop-type reactor in Kazakhstan interesting information was obtained which was published in technical magazines and discussed at international conferences and meetings. For example, on the basis of the BN-350 experience one can comment upon the maintainability of loop-type designs. It was believed that due to localization of specified parts of valves the reduction of coolant activity during reactor on-power operation can be attained.

However, operating experience has shown that in large power reactors two combined “hazardous” factors are acting: an increase of the piping diameter and an increased distance of valves along the pipe from the reactor. So, in the BN-350 the valves in the cells are situated near the “dead” supports of piping, the pump and heat exchanger, i.e. nearly 40 m long non-separated piping is between the reactor and the valve. Such layout is explained by arrangement considerations and is related with large weight of valves, considerable vertical displacements of reactor nozzles and “maintainability” requirements. After the reactor commissioning, the assumes were confirmed of a vigorous natural convection to be developed in 600 mm diameter piping connecting the reactor and the heat exchanger that caused heat and radioactive sodium transport from the reactor along the pipe. Moreover, this convection proved to spread also over the intermediate heat exchanger and was developed with a small reverse flow in the loop.

Special studies have shown the stable natural convection to develop in 2-3 hours after shutting the valve. Good reproducibility and agreement (within a measurement error) of temperature variation curves for various loops were indicating of an objective and non random character of the process. These results are confirmed by measurements of background in the cells of shut off loops at bringing the reactor to power. Due to convection, the value of the radiation flux in cells with cut-off loops differs little from those in operation. The value of convection will grow with increased coolant temperature rise in the reactor and the pipe diameter. This phenomenon characteristic of large diameter piping with horizontally arranged expansion bends somewhat decreases the advantages of the loop type layout from the viewpoint of carrying out repairs at the loop with the reactor in operation. The fabrication and installation experience of the BN-350 and BN-600 reactor vessels

One of the main elements of the reactor is its vessel. The specific conditions of its operation and the need to ensure its high reliability and fault-free operation for a long period of time place stringent requirements upon the quality of the material, welding, manufacturing precision and vessel monitoring methods. Therefore, from the economic and technical point of view it is most advisable to build the vessel completely at the factory. This version is possible for vessels only on condition that they are transported by a waterway which connects the manufacturing site and the vessel installation site that is not always desirable from the viewpoint of the plant construction site selection.

The vessels of reactors such as the BN-350 (H = 13 000 mm and D = 6 000 mm) and the BN-600 (H = 13 000 mm and D = 12 900 mm) were divided into large blocks in order to transport them by rail to the site. In the process the vessels were divided into elements which not only satisfied the requirements of railroad packaging, but also made it possible to properly assemble and weld them on site with minimum expenses. Thus, at transporting by rail, a loop type vessel has no special advantages over a pool type vessel due to the necessity of dividing them into installation blocks. However, the fact that the dimensions of the BN-600 vessel were increased by more than twice in comparison to the BN-350 made it necessary to divide it into a considerably larger number of installation blocks, and this somewhat complicated the technology and increased the vessel installation time.

On the basis of analyzing the experience of manufacturing, assembling, welding and installing the equipment of the primary circuit and reactor vessel of the BN-350, as well as the BN-600 reactor vessel, we can draw several conclusions:

(1) In manufacturing the assemblies of the vessel at the factory with their subsequent assembling and welding on site, it is absolutely necessary for the manufacturing industries to carry out a complete trial assembly of the vessel for checking all joints and geometric parameters, especially in connection with the fact that sometimes there are deviations from the drawing for the two layouts under consideration manufacturing is a unified process;
(2) The manufacture, assembly, welding and installation of the vessel requires the use of special equipment which should be designed to be used in assembling and welding, both at the factory and during installation on site; this ensures greater precision in the assembled structure and greatly reduces the expense for vessel manufacture;
(3) The primary circuit equipment (pumps, heat exchangers, fuel handling system, etc.) should be delivered for installation completely assembled after undergoing the necessary series of checks in the shops and being subjected to easily removable preservation, the installation operations must be kept to a minimum and be as simple as possible, and they must not require internal adjustment of the equipment. In general, the experience of the on-site fabrication of the BN-350 and BN-600 vessels and of their long time operation is quite satisfactory;
(4) The reactor plant operating experience has demonstrated quite acceptable maintainability of the pool design of the LMFR.

There have been no additional difficulties in repairing primary pumps on BN-600 compared with BN-350. Repairs of the reactor equipment items and sodium systems, including such complicated operations as replacement of the reactor coolant pumps, rearrangement and jacketing of primary sodium systems, extraction of distorted items from the reactor core, etc. had virtually no influence on the duration of planned outages, which were defined, as a rule, by inspections and preventive maintenance of the steam-water system components.

There have been six leaks in the austenitic stainless steel 316 Phénix intermediate heat exchangers (IHXs). The decision was taken to remove all 6 IHXs and modify them. The removal and modification one-by-one of the Phénix IHXs was perhaps the most difficult potential maintenance task foreseen on pool-type fast reactors. Its accomplishment, with the minimum of reactor downtime, represents a conspicuous success and demonstrates the importance of foreseeing events at the design stage, making adequate provision of equipment and space for the repair, planning and learning from previous operation. In spite of the restricted number of the pool type fast reactors under operation there is a great diversity of their design incarnation. Two ways of the pool type reactor vessel fixing are known: suspension from the upper plate and bottom supporting. There exist different ways of fixing pumps, IHX's and core. The important part of the pool type reactor is the inner vessel, which separates cold and hot coolant inside the vessel. A number of other alternative solutions are known. Below considered those which have a considerable effect on the reactor economic and engineering characteristics.

One of them is the way of reactor vessel fixing. The pool type PFR, Phénix and Super-Phénix reactor vessels are known to have an upper suspension and only two reactors, operating BN-600 and BN-800 that is under construction, have their vessels supported in the bottom. The vessel bottom support has some advantages which were discussed at the international conferences and meetings. The preliminary opinion was that this type of support is preferable for the middle power reactors (with the vessel diameter of about 12–15 m). The design versions using the bottom support have been considered in other countries, e.g. in the U.S. and Japan.

The Bechtel Corporation’s making use of the bottom support significantly enhances seismic stability and allows designers to reduce wall thickness of the reactor main and guard vessels. This corporation has evaluated cost savings as a result of using bottom support versus top suspension of the reactor vessel. It is necessary to state that the way of the vessel fixing directly influence the reactor cover plate design and fixing of the in-vessel components. Once again the European countries and Russian Federation designing demonstration commercial reactors encountered the problem of selecting the optimal way of the vessel being fixed and the way of smoothing out the load in the in-pile components. Thus the experience gained should be analyzed. The different ways of fixing reactor vessels and in-pile devices carried out in demonstration reactors are characterized below.

4.3.1. BN-600

It is characterized by fixing all components: the core and neutron shield, pumps, heat exchangers — on the low supporting ring and rotating plugs — on the reactor roof. Through special supports the load is transferred to the foundation. The supporting ring with all the equipment is installed and the upper part of the reactor operates at different temperatures. Therefore such a design reveals a complex problem of matching thermal expansions of pumps and heat exchangers to those of the reactor vessel and that of sealing vessel penetrations. The BN-600 reactor (cross section of pump and IHX) is shown in Fig. 28.

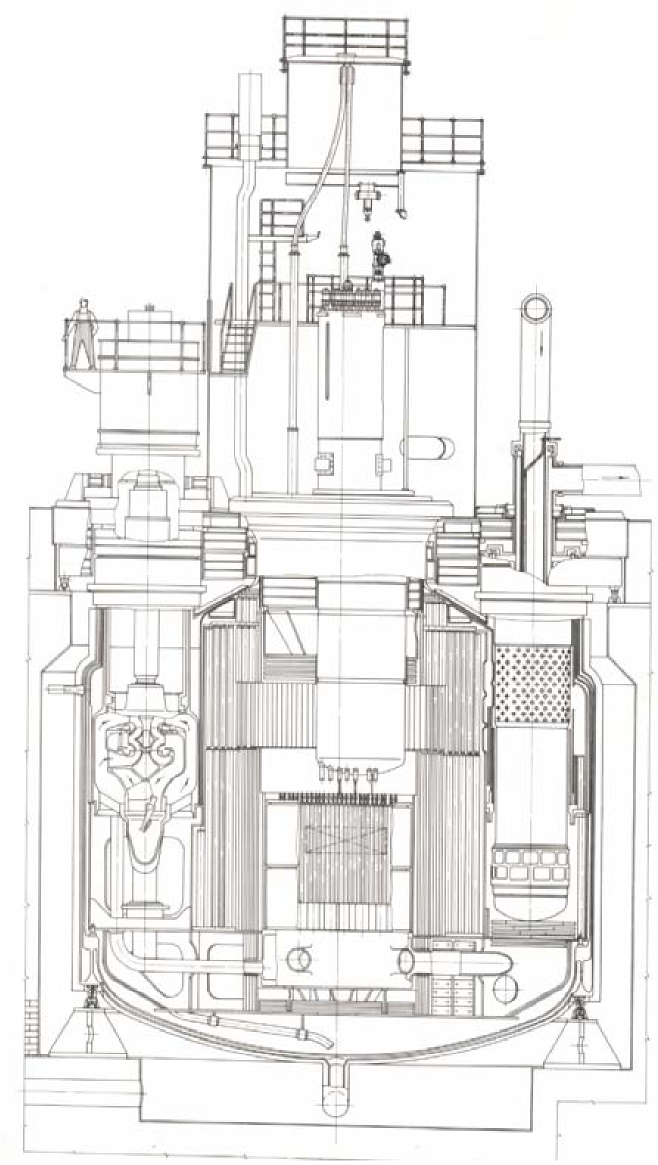


FIG. 28. The BN-600 reactor (cross section of pump and IHX).

In the BN-600 reactor these problems are known to have been solved by means of the bellows used in the cylinders where pumps and heat exchangers are installed. If necessary, the bellows can be changed. Cooling the vessel and especially its upper part decreases the difference in lateral displacements of the reactor vessel head and supporting ring, where pumps and heat-exchangers are installed, thus improving the bellows operation conditions.

Supporting structures on which pumps, heat exchangers and pressure chamber are fixed operate at the same temperature, so there is no need in using compensators between them, being a great advantage of this design. This way of vessel fixing has confirmed a good performance during BN-600 operation. As some negative peculiarities insignificantly impeding its operation we can point out the following:

- Position of the central column (with control rods) and the core depends on the vessel head temperature;
- Significant vertical displacements of the upper part of IHXs and secondary pipes.

4.3.2. Phénix

The vessel of the Phénix reactor is suspended from the upper plate on which the equipment is installed. In this design, similar to the BN-600 reactor, the reactor vessel cover plate is under high temperature conditions but in this case it is absolutely unloaded. The bearing plate operates under low-temperature conditions because between the plate and the reactor cover there is an insulation layer and besides this zone is cooled.

The pumps and heat exchangers are location movable sliding supports, sealing of the pump and heat exchanger penetrations is carried out by means of the bellows. A peculiar feature of this reactor top design is a massive cover plate having significant thermal inertia. It restraints a power growth rate or the reactor starting up and may be a cause of some significant thermal stresses occurring in the junction of the vessel cylindrical part and the horizontal cover. The sodium in the main vessel is separated into two zones:

— Hot pool at the core outlet where the hot sodium flows into the intermediate heat exchangers;
— Cold pool taken from a peripheral annular space between the primary tank and the wall of the main reactor vessel, which contains the three main circuit circulating pumps and six heat exchangers, suspended from the upper slab.

One of the main problems concerning the top support reactor design is to create acceptable operational temperature conditions for the roof, from which the vessel is suspended, and on which the equipment is installed (pumps, heat-exchangers, rotating plugs, etc.). For thermal shielding of the roof internal surface in the PFR reactor multi layer steel foil insulation was used, which turned out to be complicated and expensive. The roof was a very critical item in the construction sequence of the top suspension designs. That is why an another design the roof design is again considered, e.g. a simple solid ferritic steel roof of 0.85 m thick was adopted for EFR plant. Phénix: top support reactor vessel is shown in Fig. 29.

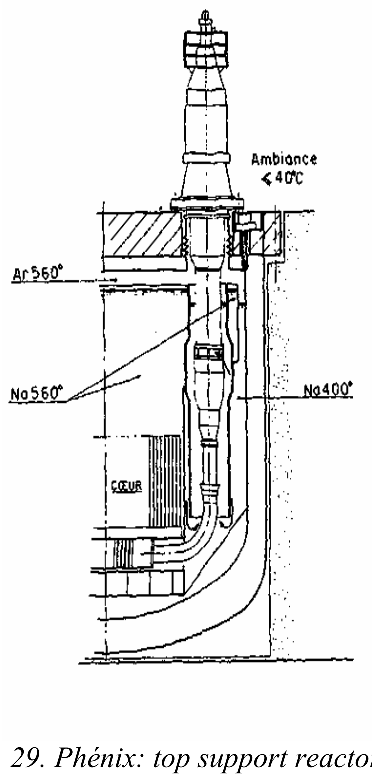


FIG. 29. Phénix: top support reactor vessel.

4.3.3. PFR

The equipment is stationary fixed on the reactor roof from which the reactor vessel is suspended. This design requires a complex thermal expansion compensation system of skirts, inside which tube bundles of heat exchangers are inserted. For this purpose the PFR reactor has bellows operating at high temperature and being inmaintainable. Compensation of pressure pipelines thermal expansions was achieved by means of the bends. In view of that several pipelines connect the pump to the pressure plenum. In the BN-800 reactor the bottom support is used as well. But the supporting ring is of a much smaller diameter, the load transfer set-up from the supporting ring to the vessel supporting assembly is changed, to enhance seismic stability roller bearings are replaced with the annular welded skirt. Elevation through PFR primary circuit is shown in Fig. 30.

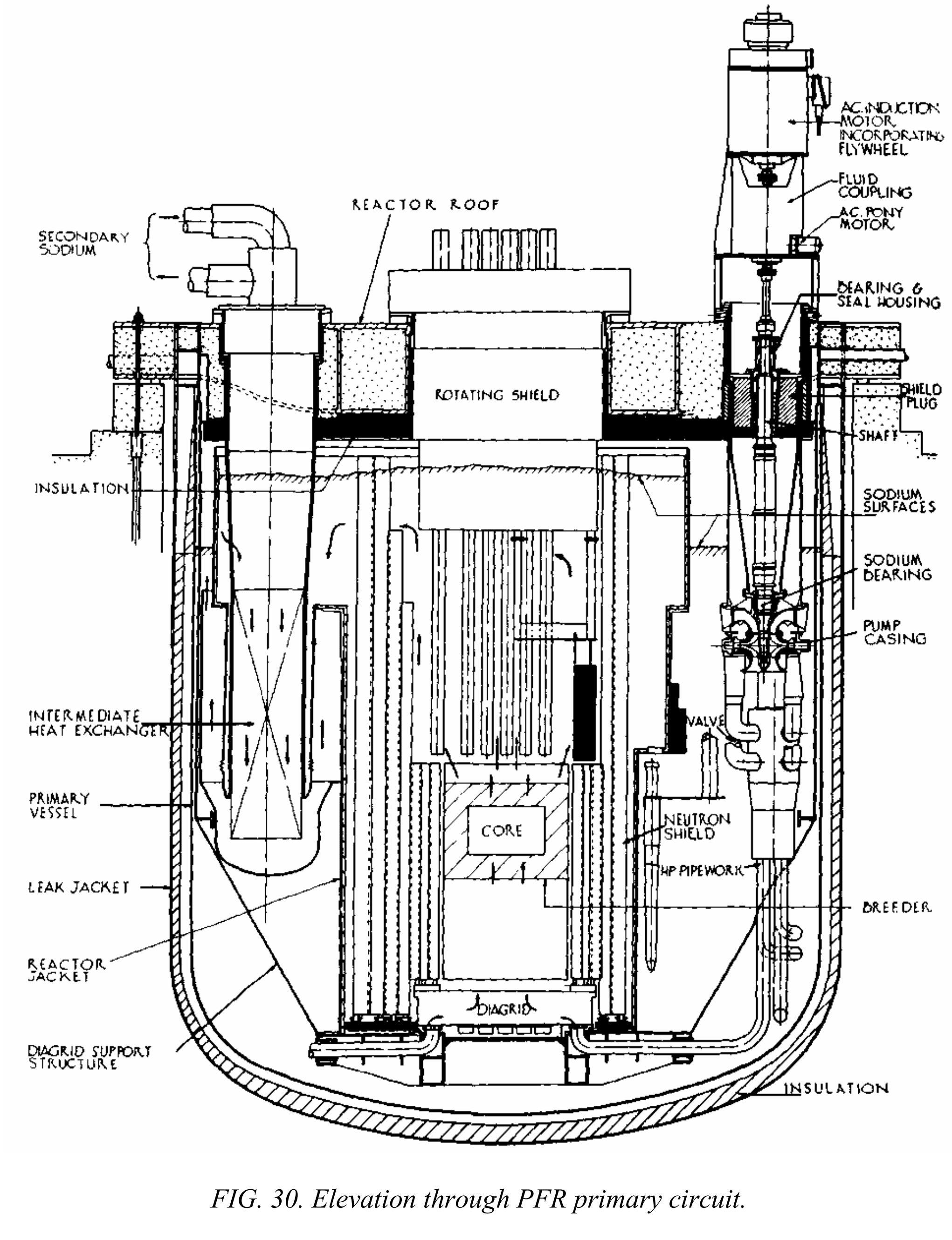


FIG. 30. Elevation through PFR primary circuit.

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5. TECHNOLOGIES FOR SODIUM AND REACTOR COMPONENTS MANAGEMENT AFTER SHUT DOWN OF THE BN-350 TYPE LMFR

5.1. INTRODUCTION AND SUMMARY

Two consultancy meetings on Decommissioning of the Kazakh BN-350 power plant (Vienna, Austria, October 1996; Obninsk, Russian Federation, February 1998) were convened by the IAEA. These meetings brought together a group of experts from France, Russian Federation, Kazakhstan, the UK, and the USA to exchange information on, and to review current technical knowledge and experience in the management of radioactive coolant and reactor components following closing of liquid metal fast reactors (LMFR), as well as their design features relevant for decommissioning procedures.

In response to the Kazakh Atomic Energy Committee (KAEC) and the necessity to preserve the knowledge in LMFR decommissioning, the IAEA collected and summarized of the technical information presented at the above mentioned meetings[7]. This chapter provides general and detailed information on activation characteristics of the primary coolant, reactor and components; treatment and disposal of the spent sodium; removal of the residual sodium deposits and decontamination of the BN-350 reactor – a typical loop type LMFR.

When decommissioning the BN-350 reactor, there arises the problem of the utilization of primary sodium with a specific activity of ~ 300 MBq/kg and of secondary sodium with a specific activity of ~ 0.5 Bq/kg. In principle, sodium can be used in the nuclear industry for newly constructed LMFRs, and in the chemical, metallurgical, engineering industries as well. In the former case, some limitations both on radioactive and non-radioactive impurities are imposed upon sodium. The rules applying to the transport of radioactive materials limit the radiation rate on the container surface to 0.55 Sv/s and at a distance of 1 m from it to 0.028 Sv/s. On the basis of calculations performed it has been shown that when using tank cars for sodium transportation these requirements are met by sodium with a specific activities of $^{22}Na \leq 6\times10^5$ Bq/kg and $^{137}Cs \leq 3.7\times10^5$ Bq/kg. Taking into account the initial sodium activity of these isotopes of 3×10^7 Bq/kg and 2.5×10^8 Bq/kg, respectively, the BN-350 primary sodium needs to have the caesium nuclides reduced by a factor of ~ 700; to reduce Na^{22} activity, decay storage duration of ~ 15 years is required. Another approach to coolant activity limitation is connected with non-admission of NPP personnel over-irradiation in the course of equipment maintenance at start-up and adjustment activities on a new reactor with reused coolant. Calculations have shown that these requirements are met by sodium with Na^{22} activity $\leq 4.5\times10^6$ Bq/kg and with ^{137}Cs activity $\leq 2.5\times10^6$ Bq/kg. To meet these less stringent requirements, the spent sodium needs to have the activity from caesium nuclides reduced by a factor of 100, and must be stored for 6 years to reach the required ^{22}Na level.

For primary sodium utilization in the chemical industry, more rigid requirements are posed: reduction of caesium nuclides by a factor of 20 000 is required. For inactive impurities the following requirements are posed: carbon ≤ 30 ppm (particles per million), oxygen ≤ 10 ppm, nitrogen ≤ 10 ppm, potassium ≤ 200/1 000 ppm (primary/secondary circuit, respectively), and hydrogen < 0.5 ppm. The spent coolant from BN-350 satisfies these impurities, requirements, except for potassium that is 400 ppm in the primary circuit and 2 000 ppm in the secondary

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[7]E.R. Adam (UK AEA) and G. Heusener (FZK, Germany) served as Chair persons of the Consultants Meetings; L. Kochetkov, A. Tsykunov, A. Karpov, V.P. Matveev, Yu. Nalimov, E. Popov, V. Shirin, and V. Yarovisyn (all from the IPPE, Russian Federation) made relative calculations and provided written contributions at the Obninsk Meeting.

circuit. Cleaning of the potassium impurities is possible by means of a single distillation at 250ºC or of dilution with pure sodium.

The sodium from the BN-350 secondary circuits can be used in a new LMFR without purification.

Overall only a small amount of so-called "sub-standard sodium", ~ 20 tons of BN-350 radioactive waste sodium coolant, is unsuited for re-use, and after reprocessing must be dispatched for burial. It includes sodium strongly polluted with the products of interaction with air or water, with fire extinguishing means, and sodium removed from cold traps and cesium traps. The analysis of the methods for processing this sodium for safe burial has revealed that the methods of waste sodium processing by means of sodium injection into water or alkali developed in France and the UK cannot be used for the processing of the "sub-standard sodium". The development of a new safe technology is required for which the experimental tests should be initiated. In order to identify the best procedure to process the "sub-standard sodium" it is recommended to test the following four methods of sodium processing: burning in air, gas-phase (O_2 or CO_2), solid-phase (metal oxides), and liquid-phase (with water under vacuum). The experimental studies performed at the Institute of Physics and Power Engineering (IPPE) on a laboratory scale confirmed the feasibility of these methods for the processing of the BN-350 "sub-standard sodium" radioactive waste. Semi-industrial tests will allow the development of the technology to be used at the BN-350 reactor.

IPPE has carried out the experimental substantiation of R&D works on new technology development aiming at rendering harmless medium level liquid radioactive waste (LRW). This new technology has been developed based on the absorption extraction of caesium and strontium radionuclides, and the formation of a new type of cement, namely geocement, by means of radioactive sorbent immobilization into the mineral-like alkaline and earth-alkaline hydroaluminosilicates. Studies have been pursued on the following:

— Principal absorbtion laws of caesium and strontium radionuclides by the natural zeolite-clinoptilolite both under steady state and transient conditions from low and medium LRWs, and
— The immobilization of the sorbent, containing ^{137}Cs, ^{134}Cs and ^{90}Sr into the geocement stones.

It has been shown in the studies that clinoptilolite, modified by sodium ions, is an effective sorbent for the extraction of ^{137}Cs, ^{134}Cs and ^{90} Sr radionuclides from the medium LRW. In addition, the clinoptiolite has been manifested to be capable, provided that the prescribed parameters of the process are observed, of assuring high quality LRW cleaned from cesium and strontium (cleaning degree up to 99.98%, distribution coefficient up to 1×10^6 mL/g) and the transfer of medium LRW to the non-radioactive category with respect to ^{137}Cs, ^{134}Cs and ^{90}Sr content, with their concentration lower than that for ^{90}Sr permissible in water. In the process of immobilization of the sorbent containing cesium and strontium radionuclides by grouting under normal humidity conditions, samples of geocement were obtained with the sorbent content up to 40 mass.%, high moisture resistance 10^6-10^7 g/cm^2 day value (close to that for the glass-like materials), and high compressive strength (10-25 MPa).

The process has been developed on the basis of accessible, low price natural materials, such as clinoptilolite as a sorbent, and metallurgical slag, clays and solutions of compositions of alkaline elements as components of the binding system. The stability of the slag-alkaline geocement, obtained as a result of hydration solidification of the binding system - the analogues of the rock forming minerals, provides readiness for environmentally safe disposal or long-term storage. The proposed method permits to convert cesium and strontium containing radionuclides LRW of magnitude into the solid form with more than two orders of magnitude reduction in volume, and the consumption of the clinoptilolite being relatively low (2-3 kg/m^3). After the cesium and strontium radionuclides are extracted from the LRW salt solutions, the radioactivity of these solutions is decreased down to zero or to an insignificant level (if some other radionuclides are present), thus simplifying considerably their further management.

The total volume of the remaining in the circuits after draining the sodium from the primary and secondary circuits and the NaK drained from the cold traps has been estimated to be ~ 100 tons. Considerations have been given how to remove this remaining sodium, and ensure remaining volumes allowing the safe conduction of washing and decontamination operations of the circuits and equipment (i.e. 150 L in the reactor vessel, and 15 L in the cold trap). The main method for the residual sodium removal in the Russian Federation is the usage of the steam-gaseous procedure. This procedure has been successfully employed three times in washing of the BR-10 reactor primary circuit. Preliminarily, for the primary and secondary circuits it is recommended to carry out vacuum distillation of sodium. For the decontamination of the primary circuit there are solutions and procedures that have proved their effectiveness at carrying out decontamination of equipment:

0.5% $KMnO_4$ at 70ºC; 5% HNO_3 + 1% $H_2C_2O_4$ at 70ºC.

It is recommended to perform the cleaning of the cold traps using the technology developed by IPPE that had been already employed at reprocessing of the BN-350 sodium from the secondary circuit cold traps.

A considerable amount of data on the evaluation of accumulated activity in BN-350 structural materials as well as in the reactor cavity has been produced using the relevant computer code. Steel structure activity is dependent on the level of impurities, e.g. manganese, europium, and cobalt, decay times of which are ~ 10 years, ~ 100 years, and ~ 150 years, respectively. Calcium dominates in concrete for 100 years, and in graphite ^{14}C dominates for 10-12 years. Therefore radioactive waste in future reactors can be reduced by reducing or eliminating certain impurities, e.g. replacing nickel with manganese in standard steels, reducing the amount of cobalt to 0.01 ppm, and reducing niobium and europium content.

After assessing the conditions at the start of decommissioning works, and the measures that would have to be taken for the reduction of radiation levels in empty pipes from the reactor vessel and for the replacement of safety and monitoring equipment, it was clear that it would not be possible to disassemble the primary circuit without robotics in 100 years, although limited access to structures should be possible in 50 years. Limited access could be gained in 10 years, provided chemical decontamination would be carried out on the primary circuit.

The meetings participants pointed out:

(1) For sodium processing, the goal is to minimize burial disposal and maximize reuse of sodium, which will be pursued grounded on the following:
   - To obtain the required purity levels, existing techniques should be refined and, if needed, new technologies, developed;
   - Storage options for reusable sodium should be investigated, including the use of the dump tank, barrels, etc.;

- Usage for sodium other than for current and future LMFRs should be investigated;
- Continue to evaluate disposal possibilities, comparing different options with different technologies. Costs and schedules associated with the deployment of these options need to be established;
- BN-350 has a NaK/mineral oil mixture. Technologies need to be identified/developed for reprocessing this mixture. Applications for use of the clean NaK need to be considered;
- Identify and/or develop a technology for dealing with liquid waste. The goal in dealing with liquid waste is waste volume minimization and final waste form stabilization.

(2) The coolant activity reduction can be achieved by the following means:
- For cesium isotopes, by means of purification with the use of carbon-graphite filters;
- For tritium, hydrogen and oxygen, by means of purification with the use of the cold filter traps;
- For corrosion products, by means of purification with the use of the getter traps.

(3) Two technologies have been developed for disposal of large quantities of both primary and secondary sodium coolant:
- NOAH technology developed and tested by Commissariat l'Énergie Atomique (France). In this process, aqueous caustic soda is produced, which can be sent to the appropriate effluent treatment plants; and
- Two-stage technology involving caustic process and carbonate process steps developed by ANL-West (USA). In the carbonate process, the sodium hydroxide is reacted with carbon dioxide to form sodium carbonate. A dry powder, similar in consistency to baking soda, is a waste form acceptable for burial.

(4) Under certain circumstances (e.g. closed geometries, sodium puddles), the use of alcohol to clean components or to destroy sodium can be dangerous. This is proven for ethylcarbitol, but at present it can not be excluded completely for other alcohols. It is therefore strongly recommended to perform appropriate studies prior to the use of alcohol together with sodium.

5.2. PRE-SHUT DOWN RADIOLOGICAL CHARACTERISTICS OF THE BN-350 COOLANT AND EQUIPMENT

By the end of life-in-service of the BN-350 reactor the following main radioactive waste will be formed:

- Fuel assemblies (FAs), blanket assemblies (BAs) and control rods;
- Reactor;
- Primary sodium;
- Pipelines and equipment of the primary circuit in the rooms, adjacent to the reactor vessel cavity;
- Pipelines and equipment of the air cooling system of the reactor vessel cavity and primary circuit cells;
- Secondary sodium;
- Cover gas system of the primary sodium tanks;

- Drainage tanks - primary circuit pressure compensators;
- System of lubrication of the main sodium pumps;
- Structural elements of the reactor vessel cavity;
- Route of the spent fuel movement, spent fuel intermediate storage tank and their cooling systems; washing systems of the fuel subassemblies, control rods and equipment.

Below data are presented on the characteristics of the main radioactive waste of the BN-350 reactor facility from the point of view of the problem of its decommissioning.

5.2.1. Primary sodium activity

During reactor operation on power the high activity of the primary coolant is determined by ^{24}Na ($T_{1/2}$ = 15 h). By the end of the reactor life after decay of ^{24}Na, the specific activity of the main radionuclides in sodium, determining radiological conditions for the sodium handling systems, is as follows:

^{22}Na ($T_{1/2}$ = 2.6 years) – 30 MBq/kg
^{137}Cs ($T_{1/2}$ = 30 years) – 250 MBq/kg
^{134}Cs ($T_{1/2}$ = 2.2 years) – 30 MBq/kg
^{22}Mn ($T_{1/2}$ = 2.6 days) – 0.3 MBq/kg
^{65}Zn ($T_{1/2}$ = 245 days) – 0.2 MBq/kg
^{60}Co ($T_{1/2}$ = 5.27 years) – 0.1 MBq/kg
Plutonium – 1 kBq/kg

Tritium ($T_{1/2}$ = 12.3 years) – 10 MBq/kg

5.2.2. Activated equipment of the primary circuit in the area adjacent to the reactor vessel cavity

The radiological conditions when handling the primary circuit equipment in the rooms adjacent to the reactor vessel cavity are mainly determined by three components:

- Activity of the radionuclides in sodium;
- Deposits of radioactive products on internal surfaces of the equipment;
- Activation of structural elements of the equipment and cells.

5.2.2.1. Radiological situation in the primary circuit cells (Na remains in the circuit)

In this case the radiological conditions in the cells will be determined by the radionuclides contained in sodium, first of all by ^{22}Na, ^{137}Cs and ^{134}Cs. For specific activities presented in Section 8.2.1, the expected calculated dose rate on the surface of thermal insulation 10 days after the reactor shutdown will be as follows:

- For ∅ 600 mm pipeline - 3.5 μSv/s, including 0.9 μSv/s caused by ^{22}Na; 1.9 μSv/s caused by ^{137}Cs and 0.7 μSv/s caused by ^{134}Cs ;
- For ∅ 500 mm pipeline - 3.1 μSv/s, including 0.8 μSv/s caused by ^{22}Na; 1.7 μSv/s caused by ^{137}Cs and 0.6 μSv/s caused by ^{134}Cs ;
- For the intermediate heat exchanger (IHX) - 1.2 μSv/s, including 0.4 μSv/s caused by ^{22}Na; 0.6 μSv/s caused by ^{137}Cs nd 0.2 μSv/s caused by ^{134}Cs

Mentioned values correlate satisfactorily with the results of measurements made in the cells.

The experimental data on the dose rates of gamma-radiation in the cells of six loops of the BN-350 are shown in Table 1. Significant dose rates (up to 10^2 μSv/s) on the primary sodium pump (PSP) overflow pipeline are caused by the increased deposits of cesium radionuclides on the surfaces adjacent to the cover gas/sodium interface of the pipelines and components.

TABLE 1. DOSE RATES OF GAMMA-RADIATION IN THE CELLS OF THE BN-350 PRIMARY CIRCUIT, μSv/s (MEASUREMENTS MADE IN 1995)

Measurement points	Primary circuit cells					
	318/1	318/2	318/3	318/4	318/5	318/6
Cell entrance area	0.40	0.50	0.45	-	0.36	0.80
Center of cell, 1 m above the floor	0.58	0.70	0.65	-	0.55	0.90
IHX - PSP ∅ 600 mm piping (reference point)	2.80	2.20	2.50	-	1.70	2.90
∅ 500 mm “cold” piping (PSP - reactor)	-	-	-	-	1.80	2.80
∅ 600 mm“ hot” piping (reactor - IHX)	-	-	-	-	1.40	2.50
PSP sodium overflow dump tank	-		-	-	3.80	3.80
PSP sodium overflow dump pipeline	-		-	-	-	60-160
PSP	-	-	-	-	-	3.80
IHX	-	-	-	-	0.92	1.00

Note: dose rates were measured on the surface of thermal insulation of pipes and equipment.

In Table 2 data are given showing dynamics of change of gamma-radiation dose rates in the reference point (IHX-PSP pipeline) for the long period of reactor operation (after the reactor shutdown and Na^{22} decay).

TABLE 2. DOSE RATES IN THE REFERENCE POINTS OF THE PRIMARY CIRCUIT CELLS, μSv/s

Year	Primary circuit cells					
	318/1	318/2	318/3	318/4	318/5	318/6
1974	-	0.04	0.04	-	-	-
1976	0.90	-	-	-	0.67	0.78
1977	-	-	3.00*	-	-	-
1978	7.50	10.00	10.00	5.00	6.80	5.00
1979	9.40	7.30	6.00	6.00	5.40	5.50
1981	3.00	3.00	3.20	2.00	1.90	1.80
1993	-	-	-	-	2.90	2.80
1995	2.80	2.20	2.50	-	1.70	2.90

* average value over all boxes

5.2.2.2. Deposits of radioactive products on the inner surfaces of the equipment

The radiological situation during dismantling of the equipment, such as heat exchangers, gate valves and pumps of the primary circuit, located beyond the biological shielding of the reactor, so that they practically have no induced activity, will be determined by the surface contamination. After the coolant is drained, different sorbed chemical elements and their combinations, containing long-lived radionuclides (~ 100 g/m^2), which determine the radiation situation during dismantling of the equipment, remain on the surface in addition to the sodium film.

On the basis of data, obtained as a result of experiments made during the BN-350 reactor operation, estimations of surface contamination of the primary circuit equipment at the moment of the reactor shutdown were carried out. The appropriate data are presented in Table 3.

TABLE 3. SURFACE CONTAMINATION OF THE PRIMARY CIRCUIT EQUIPMENT

Source of activity		Specific surface activity, MBq/m^2	Total surface activity of the equipment, TBq
^{22}Na		3.0	0.03
^{137}Cs	On the surface being in contact with Na	500	5.0
^{137}Cs	On the surface above Na level	5 000	0.9
^{134}Cs	On the surface being in contact with Na	50.0	0.5
^{134}Cs	On the surface above Na level	500	0.09
^{54}Mn	Hot Na area	1 500	13.0
^{54}Mn	Cold Na area	750	1.1
^{60}Co	-	100	1.0

It was supposed, that the surface activity of ^{137}Cs and ^{134}Cs fission products is the same for all elements of the primary circuit being in contact with sodium. For the structural elements, located in the gas cavities above the sodium surface, specific surface activity of ^{137}Cs and ^{134}Cs is about 10 times higher, than that for the surfaces, which are in contact with sodium.

Except for the radionuclides specified in Table 3, ^{125}Sb and ^{94}Nb isotopes may be present in amounts of up to 5.3 and 0.8 MBq/m^2, respectively, on the surface of the equipment of the primary circuit by the end of the reactor life. Gamma-radiation dose rates on the surface of thermal insulation of ∅ 500 and ∅ 600 mm pipelines after the coolant drainage, evaluated for the values of the surface contamination of the primary circuit equipment given in Table 3, can make:

- For ∅ 500 mm pipeline - 0.24 μSv/s;
- For ∅ 600 mm pipeline - 0.36 μSv/s (the IHX surface is 0.9 μSv/s).

The presented values could be even exceeded because of the increased deposit rate of radionuclides in gaps, dead-ends ("pockets") of sodium piping, and also on some sections of gas systems.

5.2.2.3. Induced activity of structural materials in the primary circuit cells

The in-pile experimental studies have shown that the activation of the main sodium equipment in the primary circuit cells would be low. It was shown that the activity of stainless steel caused by ^{60}Co at the time of reactor dismantling would not exceed 1 Bq/g of steel in the cells, adjacent to the reactor.

The only exception could be the steel of a limited part of sodium pipelines of the primary circuit, passing through the rooms, which are intermediate between biological shielding of the reactor vessel cavity and the cells containing the main primary sodium components. The activation will mainly take place in the area of an outlet of these pipelines from the openings in the biological shielding of the reactor vessel cavity, and also in the area of the piping inlet into the shielding. By the end of the reactor life, specific activity of steel caused by ^{60}Co would be ~ 5 Bq/g for the pipelines sections most remote from the biological shielding openings. The maximum specific activity of steel caused by the above indicated radionuclide in the area of the openings may be 50-500 Bq/g. Length of the pipelines in this sections is ~ 1 m. The dose rate of gamma-radiation on these sections caused only by the induced activity of ^{60}Co can reach ~ 3×10^{-2} μSv/s.

5.2.3. Activity of the radionuclides in the primary cold traps

In the cold traps (CT) of the primary circuit significant amount of long lived radionuclides will be accumulated by the end of the reactor life. In Table 4 the results of estimations of the content of radionuclides in all five CTs are given.

TABLE 4. ACTIVITY OF THE MAIN RADIONUCLIDES IN ALL COLD TRAPS OF THE PRIMARY CIRCUIT

Radionuclides	Activity, Bq
^{137}Cs	2.2×10^{13}
^{134}Cs	2.2×10^{12}
^{22}Na	3.3×10^{11}
^{54}Mn	1.5×10^{12}
Tritium	1.7×10^{15}

The estimation was made using results of measurements of gamma-radiation rate patterns in the cells where CT are located. The dose rate of gamma-radiation on the surface of thermal insulation of CT after the reactor shutdown and ^{24}Na decay is expected to be at the level of ~ 4.00 μSv/s.

5.2.4. Activity of air cooling system of reactor vessel cavity and the primary circuit cells

On the BN-350 reactor closed circulation circuit is provided for the cooling of the biological shielding of the reactor cavity. Air is used as a coolant. During planned shutdowns after decay of ^{41}Ar dose rates of gamma-radiation in the rooms of this system were analysed, and also gamma-spectral analysis of wet smears, taken from the external surfaces of heat exchangers and floor and from the internal surfaces of air ducts was carried out. The dose rates on the surface of the equipment of this system were:

- On air ducts: 5×10^{-3} μSv/s;
- On the floor and heat exchangers: 6×10^{-3}-2.5×10^{-2} μSv/s.

In Table 5 specific activity values, obtained as a result of gamma-spectrometer measurements of smears are presented.

TABLE 5. MEASURED VALUES OF SPECIFIC ACTIVITIES OF SURFACE CONTAMINATION

Radionuclides	Activity, MBq/m^2
^{137}Cs	0.5-2.5
^{134}Cs	0.1
^{125}Sb	0.06
^{60}Co	0.03-0.10

On the basis of these specific activities the dose rates of gamma radiation on the surface of the equipment were calculated. The total calculated dose rate was at the level of 1×10^{3}-3×10^{-3} μSv/s. The difference between evaluated and measured dose rates can be explained by that using smear technique makes it possible to determine only removable activity, but not the total amount.

On the basis of presented data it is possible to make a conclusion that the equipment of the system for air cooling of the reactor vessel cavity will be related to the category of solid radioactive wastes. On the basis of radiological measurements it will be possible to attribute also the equipment of air cooling system of the cell of one primary sodium loop (no. 3 loop) to the category of solid radioactive waste. The dose rate of gamma-radiation on the equipment of this system is 5.3×10^{-3} μSv/s. The radioactive contamination of this system mainly caused by ^{137}Cs has resulted from earlier repair work for replacement of the drainage valve in the cell of no. 3 sodium loop.

5.2.5. Activity of the equipment of the secondary circuit

In the secondary circuit of the BN-350 reactor practically only one long lived radionuclide tritium ($T_{1/2}$ = 12.3 years) will be present by the end of the reactor life. The specific activity of tritium in sodium by the time of shutdown will be 0.5 MBq/kg. The activity of tritium in the total volume of the secondary sodium will be equal to 2.4×10^{11} Bq, ~ 1.7×10^{14} Bq of tritium being contained in the cold traps.

5.3. PURIFICATION AND UTILIZATION OF SPENT SODIUM COOLANT OF PRIMARY AND SECONDARY CIRCUITS

Once decommissioning of the BN-350 reactor has commenced there will be the problem of disposing of the spent sodium coolant. About 500 tons of sodium will remain in the primary circuit of the reactor, which after decay of Na^{24} will be in the category of intermediate activity radioactive waste (~ 300 MBq/kg). In the secondary circuit ~ 480 tons of sodium remains with low specific activity (~ 0.5 MBq/kg). The calculated content of the main impurities in the primary and secondary sodium of the BN-350 reactor are presented in Table 6.

TABLE 6. THE CONTENTS OF THE MAIN IMPURITIES IN SPENT SODIUM COOLANT OF THE BN-350 REACTOR

	Impurities	Impurities concentration	
		I circuit	II circuit
1.	^{22}Na, MBq/kg	30	absent
2.	^{137}Cs, MBq/kg	250	absent
3.	^{134}Cs, MBq/kg	30	absent
4.	Tritium, MBq/kg	10	0.5
5.	α-nuclides, kBq/kg	1	absent
6.	Manganese-54, MBq/kg	0.3	absent
7.	Carbon, ppm	27	26
8.	Oxygen, ppm	2	2
9.	Nitrogen, ppm	3	3
10.	Chlorine, ppm	9	9
11.	Potassium, ppm	400	2 000
12.	Iron, ppm	3	2
13.	Hydrogen, ppm	-	0.2

The spent coolant of the BN-350 reactor could be used, first of all, in the nuclear industry for newly constructed fast reactors with sodium coolant. It could also be used in other industries as well (e.g. chemistry, metallurgy and mechanical engineering). In the chemical industry metal sodium is used widely for production of both inorganic (Na_2O_2, $NaNH_2$, NaCN, NaH) and organic ($(C_2H_5)_4Pb$, synthetic washing means, dyes, plastics, etc.) compositions. In the metallurgy and mechanical engineering sodium is used for production of high-strength cast irons, steels and various alloys. As it can be seen from Table 6, the principal contribution to the activity of sodium coolant of the primary circuit is made by $^{137(134)}Cs$ and ^{22}Na nuclides. In case of secondary circuit use of the spent sodium coolant certain recommendations on its radioactivity and impurities content will be made. This section is devoted to the analysis of these recommendations.

5.3.1. Requirements to the coolant quality for its reuse in newly constructed LMFR

5.3.1.1. Radioactive impurities content

The permissible levels of activity of the coolant for its use on the new reactor will be limited, on the one hand, by the requirements to the transportation of radioactive substances [1], and on the other hand by the requirement of non-admittance of the NPP staffing overexposure during operation with the equipment at the stage of sodium receiving, purification and realization of initial testing activity on the new NPP.

5.3.1.1.1. Transportation

When transporting outside the NPP site, the level of gamma-radiation on a surface of the transport shielded container and at 1 m distance from the container should not exceed 0.55 μSv/s and 0.028 μSv/s respectively. If transportation is made within the site, these values are 2.8 and 0.14 μSv/s, respectively. In order to evaluate the dose rates of gamma-radiation during transportation of spent primary sodium it was assumed, that transportation would be carried out in the 32 m^3 (25 tons of sodium) capacity railway cistern, the activity of the coolant being defined by three nuclides:

^{22}Na ($T_{1/2}$ = 2.6 years; a = 3.0×10^7 Bq/kg)

^{137}Cs ($T_{1/2}$ = 30 years; a = 2.5×10^8 Bq/kg)

^{134}Cs ($T_{1/2}$ = 2.1 years; a = 3.0×10^7 Bq/kg)

The calculations were carried out for two moments of time: just after the reactor shutdown and 6 years after shutdown [2, 3].

From Table 7 can be seen, that the total dose rates of gamma-radiation on a surface of the cistern and at 1 m distance exceed permissible levels, stipulated by the rules of radioactive substances transport even 6 years after the reactor shutdown [1]. Thus the maximum levels of radiation will be observed on a side surface of thermal insulation at the center of the tank (4.0 μSv/s on a surface and 2.02 μSv/s at 1 m distance from thermal insulation surface).

TABLE 7. GAMMA-RADIATION DOSE RATES FROM THE TRANSPORT RAILWAY TANK (length = 10.5 m; diameter = 2.2 m; wall thickness δ_{Fe} = 12 mm; insulation thickness δ_{is} = 250 mm; sodium mass = 25 tons)

Points of measurement		Dose rate, μSv/s			
		^{22}Na	^{137}Cs	^{134}Cs	Total
$Đ_1$	τ = 0	0.97	2.29	1.14	4.40
	τ = 6 years	0.19	1.94	0.18	2.31
$Đ_2$	τ = 0	1.10	2.64	1.41	5.15
	τ = 6 years	0.22	2.29	0.22	2.73
$Đ_3$	τ = 0	0.66	1.67	0.70	3.03
	τ = 6 years	0.13	1.50	0.11	1.74
$Đ_4$	τ = 0	1.76	3.96	2.11	7.83
	τ = 6 years	0.35	3.34	0.31	4.00
$Đ_5$	τ = 0	1.06	1.89	1.23	4.18
	τ = 6 years	0.22	1.63	0.18	2.03

$Đ_1$ – filling pipe flange; $Đ_2$ – on the surface of insulation (top of the cistern);
$Đ_3$ – at 1 m distance from point $Đ_2$;
$Đ_4$ – on a surface of insulation (sideways on center of the cistern);
$Đ_5$ – at 1m distance from point $Đ_4$.

In order to decrease radiation rate down to permissible level (0.55 and 0.028 μSv/s, respectively) it is necessary to lower specific activity of the coolant down to the following values:

- on ^{22}Na: $\leq 6\times10^5$ Bq/kg;
- on ^{137}Cs: $\leq 3.7\times10^5$ Bq/kg.

The estimations are carried out using the above mentioned technique [2, 3]. In this case 1:0.1 ratio of the contributions in dose rate made by isotopes of ^{22}Na and ^{137}Cs isotopes is assumed beforehand, taking into account, that the spent coolant can be cleaned from caesium activity (as it will be shown below), while it is impossible to separate ^{22}Na from the total mass of the coolant; the decrease of ^{22}Na activity is possible by radioactive decay during storage.

5.3.1.1.2. A new NPP start-up

For the sake of estimation of permissible levels of the coolant activity on the preliminary stage of works the following operations were considered:

— Delivery of sodium from the railway cisterns to the sodium dump tanks;
— Purification of sodium from impurities in the sodium dump tanks;
— Filling main circuits with sodium;
— Run-in of the main and auxiliary sodium systems equipment;
— Refuelling works on the reactor.

For calculations of restrictions of coolant activity with regard to maintenance and repair of the equipment in the NPP start-up period it was assumed, that the working time of the personnel does not exceed two hours (based on the experience of realization of these work), and that the dose to the worker obtained during this time is equivalent to the weekly dose (1.0×10^{-3} Sv). Thus the radiological situation will be defined mainly by ^{137}Cs. After 6 years storage and ~ 100 times reduction of caesium nuclides content the radiological situation will be defined by ^{22}Na (Table 8, row τ‘). As it can be seen from, dose rate in the vicinity of the surface of the equipment (0.5 m) after 6 years of sodium storage is estimated by 0.15-0.31 μSv/s, which enables work on maintenance and repair of the equipment to be carried out for 1-2 hours (Table 8).

TABLE 8. GAMMA-RADIATION DOSE RATES FROM THE EQUIPMENT IN SERVICE

Name of equipment	Features of equipment	Points of measurement	Time storage, years	Dose rate, μSv/s			
				^{22}Na	^{137}Cs	^{137}Cs	Sum
Sodium railway cistern	length=10.5 m diameter=2.2 m	on the surface of thermal insulation	$\tau=0$ $\tau=6$ $\tau'=6$	1.76 0.35 0.35	3.96 3.34 0.035	2.11 0.31 -	7.83 4.00 0.38
	$\delta_{Fe}=12$ mm $\delta_{isol.}=25$ m	at 0.5 m distance	$\tau=0$ $\tau=6$ $\tau'=6$	1.40 0.28 0.28	2.95 2.55 0.028	1.62 0.24	5.97 3.07 0.31
Cold trap of primary circuit	height=7 m diameter=1.2 m	on the trap surface	$\tau=0$ $\tau=6$ $\tau'=6$	0.97 0.19 0.19	2.02 1.76 0.018	1.10 0.17 -	4.09 2.12 0.19
	$\delta_{Fe}=25$ m $\delta_{isol}=30$ mm	at 0.5 m distance	$\tau=0$ $\tau=6$ $\tau'=6$	0.69 0.14 0.14	1.58 1.40 0.014	0.86 0.13 -	3.13 1.67 0.15
Lateral surface of the reactor vessel	length=19 m diameter=7 m $\delta_{Fe}=100$ mm	on the reactor surface	$\tau=0$ $\tau=6$ $\tau'=6$	0.18 0.035 0.035	0.35 0.31 0.003	0.20 0.035 -	0.73 0.38 0.04
Open hole in a cover of reactor	diameter =300 mm	above the hole	$\tau=0$ $\tau=6$ $\tau'=6$	0.022 0.004 0.004	0.06 0.05 0.0005	0.026 0.004 -	0.11 0.06 0.005
Dump tank of primary circuit	length=14 m diameter=4 m	on tank surface	$\tau=0$ $\tau=6$ $\tau'=6$	1.18 0.24 0.24	2.77 2.37 0.024	1.52 0.23 -	5.47 2.84 0.26
	$\delta_{Fe}=20$ mm $\delta_{isol}=40$ mm	At 0.5 m distance	$\tau=0$ $\tau=6$ $\tau'=6$	0.79 0.15 0.15	2.02 1.76 0.018	1.18 0.17 -	3.99 2.08 0.17

In connection with all stated above, it is possible to consider acceptable variant of filling of the primary circuit of new FBR with the spent primary sodium of the BN-350 reactor with its preliminary cleaning resulting in 100 times reduction of caesium isotopes content and ~ 6 years storage. Activity of the coolant thus will be as follows:

- On radionuclide $^{22}Na \leq 4.5\times10^{6}$ Bq/kg;
- On cesium nuclides $\leq 2.5\times10^{6}$ Bq/kg.

Although the permissible specific activity in case of equipment maintenance is higher than that for transportation of the coolant, this limiting value is acceptable for the following reasons:

- The transportation of sodium outside the NPP site is a single operation;
- It is possible to equip the transport means with biological shielding (~ 8 cm steel layer is required).

Thus, in order to achieve permissible activity level of the spent coolant of the primary circuit of the BN-350 (on $^{22}Na \leq 4.5\times10^{6}$ Bq/kg and $^{137}Cs \leq 2.5\times10^{6}$ Bq/kg) ~ 100 times reduction of cesium content in sodium and 6 years storage for the decay of Na^{22} is required. With regard to the secondary sodium, there are no restrictions for its reuse in new LMFRs from the standpoint of radioactivity.

5.3.1.2. Non-radioactive impurities content

The necessity of restriction on the content of some impurities in the sodium coolant is determined by their negative influence on the coolant quality. Among these impurities are carbon, oxygen, hydrogen, chlorine, nitrogen, potassium and iron. Sources of these impurities are:

- For iron: structural materials;
- For carbon: sodium pump lubrication system, structural materials and atmospheric air;
- For hydrogen: atmospheric air, corrosion processes in the third circuit, sodium water
- interactions;
- For oxygen and nitrogen: atmospheric air.

5.3.1.2.1. Carbon concentration

The presence of carbon in sodium results in the embitterment of the chromium-nickel steels of austenitic grade, loss of their ductility and as a consequence, decrease of their useful life. If austenitic and pearlitic steels are used in the same sodium system, carbon transfer from pearlitic steel to austenitic steel takes place. On the basis of the analysis of data on the changes of mechanical properties of steels caused by the processes of their decarburization and carburization, it is recommended to establish the following maximum permissible values of the carbon content for reactor sodium:

- For the primary circuit, consisting of only austenitic steel structures, operating at temperatures up to 700ºC - 30 ppm;
- For the secondary circuit, including structures of both austenitic and pearlitic steels, operating at temperatures up to 550ºC - 50 ppm.

5.3.1.2.2. Oxygen concentration

Oxygen, contained in sodium as sodium oxide, makes significant effect on the corrosion of structural materials. For instance, the increase of concentration of oxygen from 5-10 ppm for the conditions of the BN-350 primary circuit results in ~ 80 kg/year additional corrosion products yield rate. The concentration of oxygen in sodium is normally maintained at the level of 2-3 ppm by the operation of a cold trap, however in some periods of work (repair works carried out on the open circuit, etc.) an increase up to 10 ppm is authorized. On the stage of sodium delivery, taking into account low temperatures and possibility of the subsequent additional cleaning by cold traps, the level of the content of oxygen in sodium is recommended ≤ 10 ppm.

5.3.1.2.3. Nitrogen content

The presence of nitrogen containing impurities in sodium makes under certain conditions negative effect on the properties of structural materials (nitriding, loss of ductility and reduction of material life). On the basis of the analysis of data on the influence of the impurity of nitrogen on the characteristics of structural materials, it is recommended the level of the contents of nitrogen in sodium should be ≤10 ppm.

5.3.1.2.4. Potassium content

The restriction on the content of the potassium impurity is caused mainly by two factors:

- Potassium activation due to (n, p) - reaction with formation ^{41}Ar, which has hard gamma radiation;
- Ability to favour carbon transfer from the source to the austenitic steels.

On the basis of calculations of potassium activation (as compared to the activity of gaseous fission product ^{88}Kr) it is recommended to establish the level of the potassium contents in the primary sodium should be ≤ 200 ppm and ≤ 1 000 ppm in the secondary sodium. On the BN-350 reactor potassium content in the secondary sodium coolant is higher than the above presented value (~ 2 000 ppm). However, the influence of potassium on carbon transfer process is limited because of low temperature of sodium (≤ 420ºC). Potassium content in the primary sodium also exceeds recommended value by a factor of 2.

5.3.1.2.5. Hydrogen content

The restriction on the content of hydrogen is caused by the influence of sodium hydride on the corrosion process, sensitivity of the monitoring system for inter-circuit leakage, reactivity and neutrons spectrum. Recommended level of the content of hydrogen in the primary and secondary sodium is ≤ 0.5 ppm.

5.3.2. Requirement for quality of the coolant to be used in industry

Except for the radionuclides with the hard gamma-radiation (^{22}Na, ^{137}Cs, ^{134}Cs), spent sodium coolant contains impurities, which are dangerous for the human organism by their physical effect (tritium, plutonium, polonium and strontium). According to the Russian standards of radiation safety [4] there are levels of activity, below which substance content may not be regulated. For gama-active long-live isotopes (^{22}Na, ^{137}Cs, ^{134}Cs) this level is 1×10^4 Bq/kg. Proceeding from these standards, 3000 times reduction of the BN-350 reactor primary sodium activity caused by ^{22}Na is required (storage for 32 years) in order to use this sodium in industry. By that time ^{137}Cs decays down to the level much lower than permissible activity, and for ^{137}Cs isotope cleaning is required in order to decrease its content by ~ 2×10^4 times. Permissible specific values are as follows:

— For β-active tritium: 1×10^9 Bq/kg;
— For α-active isotopes:
  - ^{210}Po 1×10^4 Bq/kg;
  - ^{239}Pu 1×10^3 Bq/kg.

Besides, according to [4] it is required that the total activity does not exceed 1×10^4 Bq for the listed isotopes, except tritium, for which this value is 1×10^9 Bq. Thus, if the standard capacities of 100 l volume are used for the sodium transport, minimum levels of isotopes activity are:

— For tritium: 1×10^7 Bq/kg;
— For plutonium and polonium: 1×10^2 Bq/kg.

That is, neither primary nor secondary sodium of the BN-350 NPP would require purification to remove tritium; and 10 times decrease of α-nuclides content is required for the primary sodium. The secondary sodium of the BN-350 reactor can be used in industry without its purification.

5.3.3. Cleaning of the BN-350 spent sodium coolant from impurities

It follows from the previous section, that the use of the spent sodium coolant of the BN-350 reactor in the nuclear engineering and industries requires the reduction of impurities of potassium and radionuclides of cesium, ^{22}Na, ^{39}Pu^2 and ^{210}Po. The possibility of reducing these impurities within the necessary limits is now considered.

5.3.3.1. Potassium

Sodium, containing increased concentration of potassium, can be rather effectively cleaned by a method of unitary distillation [5]. Table 9 gives theoretical value of high purity sodium output calculated using Relay equation for various maximum contents of the potassium impurity depending on distillation temperature.

TABLE 9. OUTPUT OF PURE SODIUM IN THE PROCESS OF DISTILLATION OF COMMERCIAL SODIUM, CONTAINING UP TO 2 000 ppm OF POTASSIUM

Distillation temperature, °C	Output in fractions with potassium content not exceeding			
	100 ppm	50 ppm	10 ppm	5 ppm
250	94.2%	92.9%	89.9%	88.7%
300	87.6%	84.9%	79.1%	76.7%

In the experiments the distilled metal was kept at temperatures 250-290ºC, potassium being mainly sublimated. The output of sodium was at the average above 90%, content of potassium being at the level of 50-70 ppm. When cleaning spent sodium coolant of the BN-350 secondary circuit to reduce potassium impurity content from 2 000 down to 200 ppm, in case of its use in the primary circuit, it is not necessary to increase the sodium temperature above 250°C, while insignificant loss of sodium (no more than 5%) can be expected. It is expedient to dilute sodium of the BN-350 primary circuit with pure sodium for the reduction of potassium concentration from 400 ppm down to 200 ppm. It is also reasonable to dilute sodium of the BN-350 secondary circuit with pure sodium, in case of its use in the secondary circuits of future LMFRs, in order to reduce potassium concentration from 2 000 down to 1 000 ppm.

5.3.3.2. Radionuclides

Various devices were applied for cleaning sodium coolant from radionuclides of cesium, such as: cold traps of impurities, traps with the mesh made of steel, nickel, etc. However the traps with carbon-graphite materials (Russian names – MPG-6, PG, VPG, etc.) turned out to be the most efficient. Alongside with laboratory studies, cleaning of the primary circuit coolant from cesium nuclides with the help of graphite traps was carried out on the Rapsodie, EBR-II, BR-10, BOR-60, BN-350 and BN-600 reactors. The efficiency of cleaning varied from 2 times on the BN-600 reactor up to 8 times on the EBR-II reactor. During decommissioning of Rapsodie reactor, 37 tons of the primary sodium drained from the reactor with total activity 220 GBq (~ 6 MBq/L) were cleaned down to the level of 1.5 MBq/L during 12 days [6]. Recently studies on deep cleaning of sodium from caesium radionuclides [7] have been carried out at the SSC RF IPPE. The idea of the technique consists in step-by-step cleaning of the coolant, i.e. consecutive pumping (storage) of sodium through each trap till equilibrium distribution of caesium between sodium and graphite is achieved. The method allows reaching required level of cleaning of sodium from cesium radionuclides with significant reduction of amount of the sorbed material as compared to the case of the single loading. The method has not yet been developed for the use on industrial scale. Improvement of this technique on experimental rigs and in the reactor will allow carrying out cleaning of sodium coolant up to the required extent for its reuse in nuclear power engineering and chemical industry.

5.3.3.3. Tritium

Tritium, present in sodium circuits as NaT and NaOT compositions, is easily removed from sodium by cold traps, the flow rate of tritium in the cold trap depending on concentration of hydrogen in sodium. The cleaning rate is usually calculated using the following equation (without taking into account isotope exchange with a crystal phase in a cold trap) [8]:

$$J = Qs \cdot CT\ (1\text{-}CtH/CH),$$

where:
J – intensity of cleaning of tritium in cold trap, kg/s;
Qs – flow rate of sodium through cold trap, kg/s;
CH – hydrogen concentration in the circuit, ppm;
CtH – concentration of hydrogen in a trap, ppm;
CT – concentration of tritium in sodium, ppm.

The practice of studies of tritium distribution in the reactor systems shows, that more than 90% of tritium formed in the core is transferred to the coolant and caught by cold traps. Injection of the adjustable flow rate of hydrogen into the circuit makes it possible to achieve greater tritium supply to the cold trap. Thus, the cleaning of the spent sodium from tritium impurity after the BN-350 reactor decommissioning is feasible with the help of cold traps, i.e. the operation of cold traps can ensure required minimum levels of tritium in sodium, intended for the reuse.

5.3.3.4. Other radioactive isotopes

Metal radioactive impurities, including ^{54}Mn, ^{60}Co and ^{239}Pu, have rather low solubility in sodium (about 10^{-3}-10^{-4} ppm), because in principle it is possible to catch them using methods of settling-out at reduced sodium temperature or by setting up sodium flow through the cold traps. Feasibility of cold trapping of some radioactive products (^{54}Mn, ^{60}Co, ^{113}Sn, ^{125}Sb, etc.) has been confirmed experimentally [9].

The radioactive metal impurities were also effectively caught by nickel traps, where nickel foil was used as a getter. These traps have demonstrated rather high efficiency in catching corrosion products and fuel. There are also other methods of cleaning, for example, vacuum gauge separation, which can give high efficiency of cleaning. Applying these methods for cleaning of spent sodium coolant one can hope reaching required minimum level of concentration of the radioactive isotopes in sodium, intended for reuse, although today there is not yet experimental approval of this method.

The only impurity, posing extremely great problems when physical methods of cleaning are applied, is radioactive isotope of ^{22}Na with half-life period of 2.6 years. In order to decrease activity of this isotope, methods of storage during the time required for its decay, or dilution of the active coolant with pure sodium are applied. Taking into account values of required degree of sodium cleaning indicated above, it is necessary to store the BN-350 primary sodium during the following period of time: for the safe shipment – 14 years, for realization of commissioning works on newly constructed NPP ~ 6 years, for use in chemical industry ~ 32 years. Thus, applying various methods of sodium cleaning from impurities (distillation, pumping through cold traps and absorption traps, settling-out and storage, dilution with pure sodium), it is possible to lower the content of impurities down to the required limits. In order to make final decision on the method of the BN-350 spent sodium coolant management during NPP decommissioning, two options should be compared from the point of view of cost effectiveness and environmental impact, namely:

- Cleaning of sodium coolant for its reuse in the newly constructed fast neutron reactors or in the industry;
- Reprocessing of large amounts of sodium coolant for safe disposal;
- Reuse of the secondary circuit sodium on the new LMFRs and in the industry does not make problems, and no cleaning of this sodium is required.

5.4. REPROCESSING OF RADIOACTIVE SODIUM WASTES INTO SAFE CONDITION FOR DISPOSAL

Beside the spent sodium coolant, by the time of the BN-350 reactor decommissioning there will be some radioactive wastes, containing so-called sub-standard sodium, unsuitable for reuse. This sodium is strongly polluted by the products of its interaction with water or air, substances used for fire fighting, and sodium, removed from cold traps, caesium traps, etc. According to the estimations a few tens of tons of sub-standard radioactive sodium can be accumulated in the BN-350 reactor. Such sodium should be disposed under condition safe for the environment. That is, should be previously transferred into the solid state, chemically passive and steady with respect to water leaching.

Methods of radioactive sodium reprocessing by water (Rapsodie reactor [10]) and sodium hydroxide (EBR-II and DFR reactors [11]), developed by today, cannot be applied for sub-standard sodium, since it would quickly plug sodium injecting devices used for these methods. In this connection other ways of reprocessing are considered which can be used for sub-standard sodium.

5.4.1. Burning in air

This method, as well as many others, is based on preliminary transformation of sodium into a less active chemical condition by its oxidation. It is widely applied for non radioactive sodium and characterized by sufficient productivity. The rate of reaction is easily adjusted by controlling supply of sodium or air. The disadvantages of this method are high temperature and necessity of cleaning of large amounts of damped air from sodium aerosols. In case of radioactive sodium, cleaning of discharged air from radioactivity is required. When cleaning air in the water scrubber, alkali and hydrogen will be formed, and the damped air needs in addition to be dried, since the moisture contains alkali and radioactive substances.

In order to ensure complete sodium burning, occurring in the tray, the proper mixing of sodium is required. Periodic clearing of the tray from the solid residues is also necessary. Wastes of the burning process should not be considered as a final product, suitable for the long storage or disposal. Sodium oxide, filters polluted by sodium oxides and alkali, trays and other equipment located in the room where sodium will be burnt, the room itself and water solutions – all these will be polluted by the radioactive products, and therefore development of methods of the their further handling is required. In case of the use of scrubbers attention should be paid to the hydrogen produced.

5.4.2. Reprocessing using steam and water

Method of steam/gas/water washing was offered earlier for pre-repair washing of the equipment from sodium, ensuring insignificant corrosion impact on the equipment. This method is rather simple and it was widely used in national practice for washing the equipment and circuits from the residues of non-drained sodium. It is sufficient to mention that the primary circuit of the BR-10 (BR-5) reactor was washed three times using this method.

The steam/gas/water method was used for washing of the standard components of the BR-10, BN-350 and BN-600 reactors, such as fuel subassemblies, control rods and removable parts of primary sodium pump (PSP). Modified evaporator modules of the BN-350 steam generators were also washed by this method. Basically this method can be used for destruction of the small amounts of sub-standard sodium by the steam or steam/gas mixture. However, this method is rather insidious, especially if the necessity of destruction of thick layer of sodium rather than the thin film is considered. The fact is that the interaction of sodium with steam would result in alkali and hydrogen appearance. The safe method of hydrogen permanent removal and its cleaning from radioactive impurities is required. In cases when it is not possible to provide its permanent reliable drainage, the alkali formed can block, as the crust, access to the inner layers of sodium. Further, in the end of the process, when the equipment is filled with water, the crust is dissolved, and vigorous, practically uncontrolled process of interaction of sodium and water becomes probable.

In the recent years modified version of this method has been developed and tested by the OKBM and IPPE, namely washout of the equipment from sodium by water under vacuum. This method can be used as well for destruction of sodium. Advantage of this method is its relative simplicity and insignificant corrosion effect on the structural materials. However, in case of loss of the vacuum there exists the possibility of the creation of the explosive mixture [12-15]. The rate of the process is controlled by a combination of temperature and vacuum value and water flow rate. During destruction of sodium, as well as in the previous case, hydrogen and water solution of alkali are formed.

In 1977 one more modification of the water method, so-called method of water-oil emulsion was developed and tested on the BN-350 reactor. It was assumed that this technology would improve safety. However, it was found that under certain conditions water-oil emulsion can be destroyed. It is necessary to emphasize once again, that in all these cases – destruction of sodium by steam, water under vacuum or water-oil emulsion, development of a safe way of permanent removal of hydrogen and alkali is required.

Water radioactive solutions of alkali will require additional reprocessing for their transformation into the form, suitable for long-lived storage. With this aim, the process of immobilization of alkali into solid mineral-like cements [16] has been developed at the SSC RF IPPE.

5.4.3. Gas-phase oxidation of sodium

Oxygen, carbon dioxide or their mixtures can be used as oxidizers of sodium [5]. If the oxygen is used, dry sodium oxide is formed, which is then processed by water, to be turned into alkali. In case of the use of carbon dioxide, dry powder, consisting of sodium carbonate and carbon, will be formed. The process of gas-phase oxidation will be carried out at 300-350°C , this being easily controlled by varying gas flow rate.

The absence of hydrogen release – in case of completeness of oxidizing reaction, increases safety of the process. The completeness of sodium oxidation reaction is provided by sodium blending in the mixture with the mineral additives. Using technology developed at the SSC RF IPPE, sodium carbonate and carbon powder is further transformed into water-proof slag-alkaline cement stone, suitable for safe disposal.

5.4.4. Solid state sodium oxidation

Sodium oxidation can be carried out by metal oxides (CuO, Fe_2O_3, Al_2O_3, etc.). As a result of the reaction going at ~ 800°C stone-like product will be formed, suitable for disposal. This high rate reaction cannot be controlled. For the complete sodium oxidizing good sodium mixing with the oxides of metal and uniformity of reaction process over the entire volume should be provided [6].

5.4.5. Reprocessing of radioactive alkaline wastes using liquid state oxidation method

The technology of this method assumes the realization of reprocessing of sodium and sodium-potassium alloy in two main stages [7]:

Stage 1: Dispersion of radioactive waste of the alkaline metal on the solid inert carrier with the aim of obtaining dry, loose cement-like mixture;

Stage 2: Tempering of the obtained mixture together with solid clay additives by water solution by metering out the mixture in turn with the water solution, formed solution-cement mass being mixed.

Composition and ratio of components used in these two operations (alkali metal radwastes, inert filler, clay additives and tempering water solution) have been determined taking into account that spontaneous solidification of obtained solution-cement mass would result in the final product of reprocessing, namely high stability slag-alkaline cement stone suitable for environmentally safe disposal. On the first stage maximum increase of the alkali metal surface is achieved, allowing to provide required kinetic parameters and completeness of the reaction on the second stage.

On the tempering stage chemical transformation of alkali metal into alkali with its simultaneous immobilization into aluminosilicate binder product is reached. Liquid radioactive waste can be used as tempering water solution, thus enabling the reprocess two kinds of wastes simultaneously.

The process is controlled by varying input flows of the appropriate components (slag-alkaline mixture and tempering solution), its monitoring being implemented by the content of hydrogen in gas cavity. The process temperature is < 150°C. The process requires safe removal and decontamination of hydrogen released.

Final product of reprocessing, namely alkaline aluminosilicate stone contains 10-12 wt% of alkali metals radioactive waste and has high strength parameters (10^{-5}-10^{-6} kg/m^2 per day water stability and 15-20 MPa compression strength).

If sodium solid waste, mixed with fire fighting means, are reprocessed by any method, they should be previously well crushed (pounded). In order to ensure completeness of sodium reaction permanent mixing of waste should be provided during their reprocessing and subsequent grouting.

None of the methods of reprocessing of sub-standard sodium, mentioned-above, have been developed for industrial scale application. Therefore in order to improve technology and select the optimum variant creation of the experimental facility is required to perform the necessary scope of experimental studies with subsequent development on their basis of industrial technology and means for its realization. The fulfilment of this work is supposed to be carried out during decommissioning of the BR-10 test reactor.

5.5. WASHING THE RESIDUAL SODIUM DEPOSITS AND DECONTAMINATION OF REACTOR CIRCUITS, EQUIPMENT AND STRUCTURAL MATERIALS

After sodium coolant is drained from the BN-350 reactor systems in the process of its decommissioning, some sodium will still remain as a film on the inner surfaces of circuits, as well as in some places where complete drainage is impeded. In connection with its high chemical and radiation activity sodium should be removed before the secondary use or disposal of sodium equipment.

The equipment of sodium circuits and systems of LMFRs is made of expensive stainless steel. Therefore it is expedient from the economical standpoint not to dispose this equipment, thereby increasing the amount of radioactive waste, but to reuse it wherever possible in industry or to re-melt it in the metallurgy. For this purpose after washing components from sodium it is necessary to carry out their decontamination from radioactive deposit, sorbed on the inner surfaces. Below possible methods of the LMFR pipelines and components removal-cleaning and washing from sodium, sodium-potassium and decontamination are discussed.

5.5.1. Input data for the choice of methods

5.5.1.1. Primary circuit

The primary circuit of the BN-350 reactor consists of the reactor itself and six identical loops. According to the estimates made, after draining of ~ 3.5 m^3 the sodium remains on the inner surfaces of the reactor, including those of fuel subassembly mock-ups. About 1×10^{-2} m^3 of sodium remain on the inner surfaces of pipelines of the primary circuit systems of impurity indication and removal after draining, while each one of five cold traps will contain ~ 3.2 m^3 of sodium residues. After removal of sodium from 10 primary circuit drain tanks in each of them would contain on the inner surface and on the bottom ~ 2×10^{-2} m^3 of sodium. In total, ~ 22 m^3 of sodium remain in the primary circuits of the BN-350 reactor after draining of the coolant.

5.5.1.2. Secondary circuit

Each one of six independent loops of the secondary circuit of the BN-350 reactor would contain ~ 0.65 m^3 sodium with specific activity ~ 5×10^5 Bq/kg remaining on the inner surfaces after the coolant draining. Besides ~ 6.7 m^3 of sodium would remain in the non-drained tube bundles on the secondary circuit side of the intermediate heat-exchanger of each loop. About 1×10^{-2} m^3 of sodium would remain on the inner surfaces of the pipelines of the system of impurity indication and removal of each one of six loops of the secondary circuit after the coolant is drained, while ~ 3.2 m^3 of sodium remain in the cold trap. Each of four drain tanks of the secondary circuit after removal of the coolant would contain ~ 2×10^{-2} m^3 of sodium residues. In total, ~ 63 m^3 of sodium would still remain in the secondary circuit of the BN-350 reactor after the coolant draining.

5.5.1.3. System of cold traps cooling by sodium-potassium alloy

After the coolant is drained from the system, ~ 0.4 m^3 of non-radioactive sodium-potassium eutectic would still remain in the pipelines of the system. Each one of two cold traps of the system would contain ~ 0.16 m^3 of sodium-potassium alloy, while ~ 0.35 m^3 of sodium-potassium alloy are kept in the casing and coil of cooling system of each one of 14 cold traps. Five drain tanks of the system would contain ~ 0.12 m^3 of sodium-potassium alloy after the coolant removal. In total, ~ 5.7 m^3 of sodium-potassium alloy are kept in the system of cooling of the cold traps of the BN-350 reactor after coolant draining.

5.5.1.4. Cold traps of spent fuel storage cooling system and coolant processing system

In the cold trap of dismantled cooling system of spent fuel storage, ~ 3.2 m^3 of the sodium-potassium alloy is contained. On the stage of the BN-350 reactor decommissioning, each one of two cold traps of the coolant processing system would contain ~ 3.2 m^3 of non-radioactive sodium.

5.5.2. Methods for removal of coolant from undrainable parts of reactor and equipment

There are no effective and safe methods for washing and decontamination of the systems, having elements with the located significant masses of coolant. Therefore coolant removal from the undrainable equipment is necessary in order to reduce the amount of residual sodium so that it would be possible from the standpoint of safety to apply some washing technique, this being necessary condition for the development of particular technologies. One shall consider the possibilities of removal of local mass of sodium from the pipelines and equipment.

5.5.2.1. Core diagrid and pressure plenum

It is required to remove sodium remaining on the bottom of the core diagrid of the reactor vessel (~ 3 m^3) using draining pipe, reaching to the bottom. The requirement of the draining device is that it should provide < 200 L sodium residual volume after its removal. This requirement is based on the assumption of instantaneous interaction of sodium mass with the excess of water, when the pressure in the reactor vessel (~ 150 m^3), filled with the inert gas at 0.1 MPa, should not exceed permissible value of 0.17 MPa.

5.5.2.2. Cold traps

5.5.2.2.1. Cold traps of secondary circuit and coolant processing system

The contents of cold traps can be released from contents using technology, developed and applied for the regeneration of cold traps of the secondary circuit of the BN-350 reactor [17]. Firstly, ~ 2/3 of the total amount of sodium is pressed out of the trap from the inlet pipe by the inert gas, supplied under pressure through the outlet pipe. The transformation of impurities, accumulated in the trap as Na_2O and NaH, into the sodium hydroxide NaOH is based on the following reaction:

$$Na_2O + NaH \leftrightarrow NaOH + 2Na \qquad (1)$$

The process is realized by heating of a cold trap up to 470-500°C during two days. The products of regeneration have melting temperature of 390-400°C and can be removed from the trap at the temperature higher than that indicated above. In order to ensure complete removal of sodium and products of the reaction (1) the trap should be equipped with a special line providing the possibility of removal of sodium, its residual amount being within permissible limits to guarantee the safety, i.e. that the internal pressure of the cold trap should not exceed permissible value (1.0 MPa) in case of instantaneous sodium interaction residues (~ 20 kg of sodium) with the excess water.

5.5.2.2.2. Cold traps of the primary circuit

The work with cold traps of the primary circuit is complicated by a high radiation background, created by the accumulated radionuclides. The feature of the primary cold traps is in that they mainly accumulate sodium oxide Na_2O. Therefore, in order to make hydration, supply of hydrogen from outside is necessary. The experiments made under special conditions [18] have shown the possibility of realization of operation of complete draining of a cold trap of the primary circuit from accumulated impurities. The technology of the draining of cold traps of the primary circuit should be developed on the basis of existing method developed for the secondary circuit traps, taking into account high radioactivity.

5.5.2.2.3. Cold traps of Na-K cooling systems of cold traps and spent fuel storage

The technology of draining of the cold traps of these systems should be proved out experimentally under special conditions using principal approach, described in the previous sections. For realization of such approach it is necessary to ensure electrical heating of traps, since electrical heating has not been provided by the design of sodium-potassium alloy cooling systems.

5.5.2.3. Intermediate heat exchangers

Draining, washing and decontamination of the intermediate heat exchanger sections is carried out using design technology in the rotary decontamination pit of the BN-350 reactor washing and decontamination system. First, sodium draining (pouring out) is made from the U-shaped tube bundle by its 93° tilting.

5.5.2.4. Na-K alloy in casings and coils of cold traps cooling systems

It is assumed that sodium-potassium alloy will be removed from casings and coils of the cold traps cooling systems using method of vacuum distillation with the use of standard systems of electrical heating of cold traps. The technology of distillation should be tested on the models under test rig conditions.

5.5.2.5. Drainage tanks

It is expedient to remove sodium residues from drainage tanks using method of vacuum distillation. This technology should be tested under special conditions on models, taking into account tanks design and volume.

5.5.3. Methods for washing and decontamination of circuits and equipment

5.5.3.1. Washing the residual sodium deposits

Steam and gas washing method and subsequent decontamination by demineralised water solutions is considered as the principal method of washing and decontamination of pipelines and equipment of BN-350. Such conclusion has been proved by the following reasons:

- Presence of designed systems for washing and decontamination of the BN-350 equipment;
- Experience gained from application of the method under reactor conditions (BR-10, BOR-60, BN-350, BN-600) [20] and presence of the skilled personnel for its implementation;
- Efficiency, profitability and relative safety of the method;
- Presence of designed systems of primary processing of wastes (including radioactive) and experience gained from its operation;
- Successful experience (three times) of application of the method for washing and decontamination of primary circuit of experimental reactor BR-10 [19].

It is reasonable to apply vacuum distillation method as an auxiliary process (used before steam/gas washing) of cleaning of circuits and equipment from sodium. This method is remarkable for the high safety, since it is not accompanied by the release of hydrogen. All necessary conditions for its implementation are available on the BN-350 reactor, namely the presence of designed vacuum and heating systems on the most part of the equipment and pipelines. Results of application of this method will be the reduction of total mass of sodium to be removed from the pipelines and equipment and improvement of safety of procedures using water methods.

Washing of the equipment by light spirits cannot be recommended for use on large scale because of high fire risk, although this method is still used sometimes for washing the experimental fuel subassemblies from sodium. The method of washing of the equipment and pipelines from sodium using heavy spirits (butylcellosolve, carbitol) is considered [20].

However, these methods have their disadvantages. In this case, as well as in steam/gas/water washing, there is a danger, connected with the release of hydrogen. Moreover, it is necessary to take into account hypothetical possibility of decomposition of organic compositions at high temperatures. It is well known, that washing procedure carried out with the help of carbitol on the drain tank on the Rapsodie plant, resulted in the burst of the tank wall.

The IWG-FR at its Meeting in May 1999 summarizing results of the discussions took note of a presentation of the French delegation on the above mentioned accident and made the following conclusion: under certain circumstances (e.g. closed geometries and sodium puddles), the use of alcohol to clean components or to destroy sodium can be dangerous. This is proven for ethylcarbitol, but at present can not be excluded completely for other alcohols. It was therefore strongly recommended to perform appropriate studies prior to the use of alcohol together with sodium.

Rather promising technology being developed, is the use of water under vacuum for washing equipment from sodium [21]. Advantages of this method are simplicity, low level of temperatures and controllability. Danger caused by the use of this method is related to possible loss of vacuum and the subsequent filling of the circuit with the atmospheric air that can lead to formation of explosive mixture.

5.5.3.2. Decontamination

As regards decontamination procedures, application of solutions, tested in fast reactors under operation [19] is expedient. The general tendency of perfection of decontamination processes is aimed at “softening” of their corrosive influence on structural materials, if reuse of the equipment or the whole circuits is assumed. The main purpose of decontamination of the equipment and pipelines of the primary circuit on the stage of the BN-350 reactor decommissioning will probably consist in assurance of removal, to as great as possible extent, of a superficial layer, polluted by radionuclides of sorption/diffusion nature, with subsequent utilization of decontaminated structural material. Therefore, at least on the initial stage of decontamination of the primary circuit equipment it is expedient to use solutions, used on BR-10, BN-350, and BN-600 reactors [19].

More aggressive solutions could be used for the final decontamination with the purpose of utilization of metal in the national industry. The possibility of additional cleaning of superficially polluted metals by their melting has been demonstrated by tests, which could be taken into account when making choice of final technology of decontamination.

5.5.4. Technology for washing and decontamination of reactor circuits and equipment

5.5.4.1. Primary circuit

Technology of washing and decontamination of the primary circuit of the BN-350 reactor should be developed on the basis of experience gained on washing and decontamination of the primary circuit of the BR-10 experimental reactor [19] taking into account the scale factor. This technology assumes the use of vacuum distillation of sodium with subsequent steam/gas washing, after sodium is drained from the circuit and undrainable sections (reactor, impurity indication and removal system and drain tanks) are isolated. In order to implement this method on the BN-350 reactor, it is necessary to dismantle sections of the intermediate heat exchangers. The procedures are as follows: the circuit is heated up to 130-150°C; the steam/gas mixture is then supplied to the top of the primary circuit loops at ~ 130°C as well; the steam together with alkali is dumped from the drainage pipelines to the condenser.

Permanent hydrogen release is provided from the top of the circuit. The process is monitored by the hydrogen release into gas phase and by alkali content in the steam condensate.

The next operation i.e. controllable batch filling of the circuit, which is filled with the inert gas at the bottom, with demineralized water and its subsequent circulation in turn through all loops by the standard pumps. It is also reasonable to consider the safety requirements for using method of washing by water under vacuum for this purpose. After washing the loops of the primary circuit except for the reactor vessel, they are subject to several cycles of decontamination using the following solutions:

— 0.5% $KMnO_4$ at 70°C;
— 5% HNO_3 + 1% $H_2C_2O_4$ at 70°C;
— demineralized water at 60-80°C.

Time and the number of decontamination cycles should be determined using model tests, taking into account the scale factor and measurements of gamma-background carried out during decontamination process. Final washing of the equipment by demineralized water will be carried out until the neutral reaction of the washing water is obtained. The washing procedure is followed by drying of the primary circuit by its heating under vacuum and blow-down by dry nitrogen.

5.5.4.2. Secondary circuit

Before starting dismantling the secondary circuit, sections of the intermediate heat exchangers should be replaced by pipelines. Washing of the secondary circuit is made in the order, similar to that used for the primary circuit (except for decontamination), including the following operations:

— Drainage of sodium;
— Isolation of the undrainable sections (systems of impurities control, drainage tanks);
— Vacuum distillation of sodium;
— Steam/gas washing;
— Gradual fillings of the circuit, filled with the inert gas, with demineralized water with its subsequent circulation in closed circuit of each loop. It is expedient to consider the possibility of safe washing method using water under vacuum;
— Drying of loops.

Application of such approach would allow washing of sodium cavities of the steam generators before their dismantling. However, it is necessary to consider more carefully the expediency of preliminary washing of the steam generator, to be made independently of the rest part of the circuit.

5.5.4.3. Cold trap Na-K cooling system and spent fuel storage coolant cleaning system

Washing of the drainable equipment and pipelines of systems of cooling of the cold traps and cleaning of the cooling circuit of the spent fuel storage can be carried out using available technology with step-by-step dismantling with subsequent washing in decontamination pits. This can be done after the discharge of the drainable sections of the circuit. Washing method based on cleaning media circulation is inexpedient in view of absence of electrical heating of the pipelines, valves and equipment, absence of mechanical pumps in the circuit and in views of necessity of preliminary removal of undrainable amounts of sodium-potassium alloy from casings and coils of the cold traps.

5.5.4.4. Washing of the undrainable equipment (with decontamination of the primary circuit components)

Washing and decontamination of the undrainable equipment of the primary and secondary circuits, cooling alloy system of spent fuel storage and cold trap of the cooling circuit of spent fuel storage can be carried out after removal of the coolant and impurities in order that their residual amounts would allow safe washing and decontamination procedures to be fulfilled by the methods, mentioned above. For these purposes special technological system of washing and decontamination should be created. Those components which cannot be processed using design means of washing (decontamination pit), should be connected in turn to the special system. Washing and decontamination of these components have to be made by means of design technologies, namely: vacuum distillation, washing by steam/gas, demineralized water or possibly by water under vacuum, and decontamination.

5.5.4.5. Washing of the reactor

The clearing (washing) of the reactor becomes necessary sooner or later to eliminate the possibility of uncontrollable interaction of the sodium remaining in the reactor vessel and water with the subsequent probable formation of an explosive mixture. However, decontamination of the inner surfaces makes no sense because of high of induced activity. Initial condition of the reactor assumes unloaded core, reactor vessel being isolated from the circuits, filled with the inert gas and cooled down. The process of cleaning of the reactor will include the following procedures:

— Development and installation for special system of removal of the main amount of sodium residues;
— Heating up of the reactor and drainage of the main amount of sodium;
— Vacuum distillation of sodium (if the method of reactor heating works out);
— Installation of system for supply of water and steam/gas mixture and removal of water alkali solution;
— Final destruction of the sodium residues by water in the inert atmosphere, water being supplied in measured portions into the reactor vessel at low temperature.

Instead of three former operations it is necessary to consider:

— Possibility of exclusion of the third operation; or
— Its replacement by washing of the sodium residues with butylcellosolve, mixed with indifferent dilute(nt).

It is necessary to emphasize once more that this section gives results of preliminary development on the system of washing of the equipment and pipelines of the primary, secondary and auxiliary (sodium-potassium) circuits. This system is based on rather large industrial experience and results of special experiments, some of which are completed, while others are still under way. Taking into account the scale factor, hydrogen related danger, accompanying washing processes, poor development and comparative evaluation of the alternative methods, the work on substantiation of the optimum and the most safe washing methods will be continued. As a result of washing and decontamination of the equipment and pipelines of the primary circuit considerable amount of liquid radioactive wastes will be accumulated. Evaluation of their amount (without results of washing of the reactor and undrainable sections) is shown in Table 10, and their total activity is shown in Table 11. As it can be seen from the tables, the total liquid radioactive waste amount according to these assumptions is equal from 9 000 to 12 000 tons, with their activity consisting of ~ 4 TBq (~ 110 Ci).

TABLE 10. LIQUID RADWASTES AFTER WASHING AND DECONTAMINATION OF PRIMARY CIRCUIT OF (REACTOR VESSEL AND UNDRAINABLE ELEMENTS ARE NOT TAKEN INTO ACCOUNT)

		BR-10, measurement	BN-350, estimation		
			1 loop	6 loops	Total
Surface, m^2		100	1 800	10 800	-
Volume, m^3		2	70	420	-
Steam-gas	Steam, t	30	540*/1 050**	3 240*/6 300**	3 660*/6 720**
	Distillate, t	2	70	420	-
Decontamination	D-solutions, t	30	420	2 520	5 040
	Distillate, t	30	420	2 520	-

Note: Estimations are given in the assumption of application of washing and decontamination technology, tested on the BR-10 reactor. The initial data on the BR-10 reactor are taken from the results of maximum washing;

* calculation was made on the basis of the ratio of the processed areas of surface;
** calculation was made on the basis of ratio of circuit volumes.

TABLE 11. TOTAL ACTIVITY OF LIQUID WASTE AFTER STEAM/WATER WASHING AND CHEMICAL DECONTAMINATION OF ONE LOOP OF THE BN-350 REACTOR (COLD TRAP AND EQUIPMENT OF REACTOR VESSEL CAVITY ARE EXCLUDED) ["Hot" surface ~ 1220 m^2 (50 m^2 pipes, 1170 m^2 IHX), "Cold" surface ~ 110 m^2 (90 m^2 pipes, 20 m^2 PSP, valves)]

	Surface contami-nation	Total activity of 1 loop	Total activity of water after steam/water washing, GBq		Total activity of water after chemical decontamination, GBq		Total activity of water, * GBq
Nuclides	MBq/m^2	GBq	t = 0	5 years	t = 0	5 years	5 years
^{22}Na	3	4	4	1	0	0	0
^{137}Cs	500	660	590	520	70	60	580
^{134}Cs	50	70	60	12	10	2	14
^{54}Mn	-	-	-	-	-	-	-
hot	1.500	1.820	-	-	-	-	-
cold	750	2026	1416	18.5	470	5	23.5
^{60}Co	100	-	100	50	30	15	65
Total	-	2 760 (75 Ci)	2 170 (58 Ci)	~ 600 (16.2 Ci)	580 (15.5 Ci)	~ 82 (2.2 Ci)	~ 682 (18.4 Ci)

Notes: Total activity of liquid radioactive waste in case of washing of all six loops will be ~ 4 GBq (~ 110 Ci);
* after steam/water washing and chemical decontamination

5.6. RADIATION CONDITIONS IN THE PRIMARY CIRCUIT CELLS, ACTIVITY OF REACTOR BLOCK ELEMENTS

Activity values of various coolants and equipment of the reactor installation presented in Section 8.2 demonstrate the possibility of the selective approach from the point of view of radiological safety assurance, if some works on the reactor decommissioning are under way. The following points can have influence on the choice of technical approaches (for example, remote or usual methods of the equipment handling, etc.):

- Wide range of low and high activities of the equipment;
- Possibility of realization of technological operations after short or long storage;
- Possibility of realization of decontamination of the polluted equipment.

5.6.1. Radiological conditions of dismantling works in the primary circuit cells

Calculated data on the radiological conditions in case of dismantling work carried out in the primary circuit loops cells are presented in this section. Only the radioactive sources located in the sodium coolant or on the inner surfaces of the equipment were considered. Radioactive contamination of outer surfaces of the equipment is absent. Table 12 shows the dose rates of gamma-radiation in the cells and on the surface of the equipment for both sodium filled and drained conditions, of the primary circuit after reactor shutdown and decay of ^{24}Na are shown,. The data are presented for various times after final reactor shutdown.

TABLE 12. DOSE RATES OF GAMMA-RADIATION ON THE EQUIPMENT OF THE PRIMARY CIRCUIT, μSv/s

Time after reactor stop, y	average gamma-background in the cell		Surface of thermal insulation of ∅ 600 mm pipeline		Surface of thermal insulation of PSP		Surface of thermal insulation of IHX		Surface of thermal insulation of cold traps	
	F	D	F	D	F	D	F	D	F	D
0	0.7	0.07	3.5	0.35	3.8	0.38	1.2	0.9	4.0	2.0
5	0.4	0.015	2.0	0.075	2.2	0.08	0.7	0.2	3.0	1.9
10	0.3	0.01	1.6	0.055	1.7	0.06	0.5	0.15	2.0	1.5
50	0.1	0.002	0.5	0.015	0.55	0.015	0.18	0.03	0.6	0.5
100	0.03	0.0005	0.15	0.005	0.15	0.005	0.05	0.009	0.15	0.13
150	0.01	0.0001	0.05	0.001	0.05	0.001	0.013	0.003	0.05	0.04

Note: F - the circuit filled with sodium; D - the circuit drained.

The radiological conditions indicated in Table 12 allow estimating the permissible time for group A[8] personnel to stay in the cells of the primary circuit under radiological conditions indicated in Table 12. In this case, let us take 90 μSv value as permissible dose limit for one shift. The dose rates of gamma-radiation in the cells of the primary circuit after reactor shutdown and decay of ^{24}Na are within the range of 0.5-4 μSv/s. Assuming a value of 90 μSv as permissible dose limit for one shift, it follows that the permissible time for personnel to stay in the vicinity of the cell entrance is ~ 3 minutes, and ~ 0.3 minutes near the components. Only visual examination can be made during these short times.

As it can be seen from Table 12, the storage during 5–10 years results in only 1.5-2 times increase of permissible operating times for the primary circuit cells without drainage of the coolant. The work of the personnel of the A category in the primary circuit cells during whole shift (8 hours) can be permitted only after ~ 150 years. The coolant drainage reduces by about 10 times dose rate of the gamma-radiation from the pipelines and pumps, while the dose rate from heat exchangers and cold traps is reduced by 1.2-2 times. After coolant drainage the permissible operating times of personnel in the cells during one shift are estimated as follows:

- At the entrance of the cell ~ 30 min;
- In the vicinity of pipelines and PSP ~ 7-10 min;
- In the vicinity of IHX and cold trap ~ 1-2 min.

As has been already stated, high dose rate of gamma-radiation (50-100 μSv/s) will remain on the short sections of the PSP sodium overflow pipeline. The storage during 5-10 years reduces dose rates of gamma-radiation on the drained equipment (e.g. pipelines, PSP and IHX) by about 5–7 times, while for the cold traps this reduction is only ~ 30%. Apparently, the work of the personnel of the A category in the cells of the primary circuit during total shift time (8 hours) can be permitted only about 100 years after shutdown of the reactor. In order to increase significantly permissible time of staying of working time of the personnel in the primary circuit cells, steam/water washing and chemical decontamination of the inner surfaces of pipelines and equipment are required. For this purpose it is necessary first of all to isolate reactor from the circuit. The cold traps should be isolated after the drainage of the coolant before steam/water washing of the sodium loop.

Chemical decontamination, as well as steam/gas/water washing of the sodium loops, should be carried out after their isolation from the reactor. This is necessary in order to avoid transfer of radioactive products of corrosion of the in-vessel elements of the reactor, and for the reasons of washing process safety. In order to reduce contamination of the air in the cells, caused by the “dusting” operations (cutting and welding of metal, etc.) during preparatory stage of decontamination and also during the general dismantling work, the working places should be equipped with local ventilation, and the personnel should have individual means of protection of the respiration organs. Table 13 shows the activity reduction ratio of superficial deposits caused by steam/water washing and chemical decontamination of smooth surfaces of the primary circuit equipment.

[8] Equivalent dose for the personnel of group A dealing directly with industrial sources, in accordance with Radioactive Safety Standards (NRB-96) is 20 mSv per year.

TABLE 13. ACTIVITY REDUCTION RATIO OF SUPERFICIAL DEPOSITS CAUSED BY STEAM/WATER WASHING AND CHEMICAL DECONTAMINATION OF SMOOTH SURFACES OF THE PRIMARY CIRCUIT EQUIPMENT

Environment of the primary circuit components	Radionuclides	After steam/water washing	After chemical decontamination	Total reduction
Gas	^{137}Cs, ^{134}Cs	100	50	5 000
Gas	^{54}Mn, ^{60}Co	2	30	60
Sodium	^{137}Cs, ^{134}Cs	10	300	3 000
Sodium	^{54}Mn, ^{60}Co	4	50	200

Table 14 presents levels of superficial contamination of the primary components, caused by the main radionuclides, after steam/water washing and chemical decontamination.

TABLE 14. SUPERFICIAL CONTAMINATION OF THE EQUIPMENT OF THE PRIMARY CIRCUIT AFTER STEAM/WATER WASHING AND CHEMICAL DECONTAMINATION, kBq/m^2

Time after reactor shutdown, years	On the surface in contact with sodium				On the surface in contact with gas	
	^{137}Cs	^{134}Cs	^{60}Co	^{54}Mn*	^{137}Cs	^{134}Cs
0	160	16	500	7.500	1.000	100
5	140	3	260	100	900	20
10	125	0.07	135	1.5	800	5
50	40	0.006	20	-	300	0.05
100	12	-	0.7	-	70	-
150	3	-	0.001	-	20	-

* Hot sodium section

Table 15 shows the calculated dose rates of gamma-radiation on the equipment of the primary circuit in the sodium loops cells after steam/water (SW) washing and chemical decontamination (CD). Dose rates at 0.5 m distance from the surface of thermal insulation of the pipelines and equipment will be about 1.5 times less, than those on the surface. After steam/water washing and chemical decontamination the permissible operating time of the operators for one shift inside the cells of the primary circuit is estimated as follows:

TABLE 15. DOSE RATES OF GAMMA-RADIATION ON THE SURFACE OF THERMAL INSULATION OF THE EQUIPMENT OF PRIMARY CIRCUIT AFTER STEAM/WATER WASHING AND CHEMICAL DECONTAMINATION, kBq/m^2

Time after reactor shutdown, y	Average gamma-background in the cell		∅ 600 mm pipelines		PSP		IHX		Local point (pipe of sodium overflow drainage from PSP)	
	SW	CD	SW	CD	SW	CD	SW	CD	SW	CD
0	0.015	0.001	0.08	0.005	0.09	0.006	0.2	0.013	1.0	0.02
5	0.0025	0.0002	0.012	0.001	0.015	0.001	0.03	0.0015	0.9	0.018
10	0.0015	0.00006	0.0075	0.0003	0.008	0.0003	0.02	0.0008	0.8	0.016
50	0.00025	2×10^{-6}	0.0015	1×10^{-5}	0.0015	1×10^{-5}	0.003	5×10^{-6}	0.3	0.005
100	0.0001	-	0.0005	-	0.0005	-	0.0009	-	0.07	0.0015
150	2×10^{-5}	-	0.0001	-	0.0001	-	0.0002	-	0.02	0.0004

After steam/water washing:

— At the cell entrance and in the middle area — 1.5-2 hours;
— In the area of pipelines and primary sodium pump (PSP) — 0.3 hours;
— In the vicinity of IHX — 0.25 hours.

After chemical decontamination:

— At the cell entrance and in the middle area — without restriction;
— In the area of pipelines and PSP — 4 hours;
— In the vicinity of — 2 hours.

Additional storage of the equipment during 5–10 years (after the steam/water washing and chemical decontamination) can practically eliminate the restriction on working time of the group A in the cells of the primary circuit. Of course, these procedures would result in formation of certain liquid wastes, requiring additional efforts for their solidification. The dose rates of gamma-radiation on a surface of the equipment will be about two times higher than those on the surface of thermal insulation.

5.6.2. Radiological characteristics of activated in-vessel structures, reactor vessel and reactor cavity

After final reactor shutdown, the project of the BN-350 decommissioning is required for fulfilment of preparatory work, unloading of all fuel assemblies and control rods from the reactor, drainage of the coolant and washing of the equipment and pipelines from the residues of sodium. Further it will be possible to commence the initial stages of dismantling of some reactor structures. It is necessary to take into account that the unloading of the fuel subassemblies is accompanied by their replacement with the non-radioactive steel dummy subassemblies, which should require in the future washing from the sodium residues and decontamination.

The main amount of solid radioactive wastes is caused by the activated structural materials of the reactor and shielding materials, located in the reactor vessel cavity. Reactor vessel is located in the concrete cavity and is separated from the concrete cavity walls by the lateral radiation shielding, consisting of steel plates and the reinforcing cage filled by the iron ore concentrate. On the foundation concrete beam under the reactor vessel the bottom shielding is located, which is made of materials, similar to those used for the lateral shielding. In the neck of the reactor vessel rotating plugs made of steel sheets, graphite and thermal insulation are located, playing a role of thermal and biological shielding. The upper section of the cavity beyond the reactor vessel neck is blocked by the upper stationary shielding, consisting of steel sheets, space between them being filled with the serpentinite concrete.

The activity of this structure was estimated for 2003 as a conditional date of expiration of the reactor life. Until 1st of December 1997, the real history of the reactor operation was taken into account, while after this date average capacity of the reactor for the previous period was used.

5.6.2.1. Calculation codes of spatial-power distributions of neutrons and activation of materials [22–24]

The spatial power distribution of neutrons in the reactor and in the ex-vessel shielding structures was calculated by the TWODANT code in RZ-geometry with the use of 26-group data base library ABBN-93. Analytical studies on accumulation of the radionuclides in the main structures of the reactor, shielding and building materials were carried out by the FISPACT code, taking into account irradiation history of materials during 30 years of the reactor operation.

5.6.2.2. Initial contents of elements in structural materials

Impurities were taken into account in the calculations of the radionuclides accumulation in the materials. The experimental data on the impurities contents in the iron ore filling and Cr18Ni9 steel of the BN-350 reactor are presented in Tables 16 and 17. These tables also give expertise estimated data on the contents of some impurities, which were not determined by the experiments, although they can influence the total value of the induced activity.

TABLE 16. THE CONTENTS OF BASIC ELEMENTS AND IMPURITIES IN IRON ORE FILLING (wt%)

Elements	Concentrations	Elements	Concentrations	Elements	Concentrations	Elements	Concentrations
O	26.9	Ca	0.079	Cu	0.015	Eu*	0.1E-4
Na	0.096	Sc	0.580E-4	As	0.270E-2	Nb*	0.1E-3
Mg	0.170E-2	Ti	0.10	Zn	0.081	Ni*	0.1E-2
Al	0.036	V	0.650E-2	Mo	0.170E-2	Cs*	0.1E-3
Si	3.3	Cr	0.072	Ag	0.110E-3	Sm*	0.1E-3
P	0.022	Mn	0.210	In	0.013	N*	0.01
S	0.001	Fe	68.6	Sb	0.110E-2	-	-
K	0.11	Co	0.380E-2	W	0.800E-2	-	-

* Elements which were taken into account in calculation in addition to those measured.

TABLE 17. THE CONTENTS OF BASIC ELEMENTS AND IMPURITIES IN 1CR18NI9 STAINLESS STEEL (wt%)

Elements	Concen-trations	Elements	Concen-trations	Elements	Concen-trations	Elements	Concen-trations
C	0.08	Cr	18.0	Mo	0.11E-2	Eu*	0.1E-4
N	0.027	Mn	1.36	Ag	0.71E-4	Cs*	0.1E-3
Mg	0.4E-2	Fe	70.5	Cd	0.11E-2	Sm*	0.1E-3
Al	0.011	Co	0.021	In	0.021	-	-
Si	0.43	Ni	9.4	Sn	0.45E-2	-	-
P	0.01	Cu	0.064	Sb	0.34E-2	-	-
S	0.01	Zn	0.017	W	0.42E-2	-	-
Ti	0.013	As	0.21E-2	Pb	0.20E-3	-	-
V	0.75E-2	Nb	0.25E-2	-	-	-	-

* Elements which were taken into account in calculation in addition to those measured.

Experimental studies were not made on the contents of the impurities in standard and serpentinite concrete, thermal insulation of plugs and graphite and 3-grade steel of the BN-350 reactor. Table 18 shows the most complete data on the contents of the impurities in concrete were published in the article [26].

TABLE 18. THE CONTENTS OF BASIC ELEMENTS AND IMPURITIES IN SHIELDING CONCRETE (1.0E-4 wt%)

Element	Content spread	Ave. cont.	Ele-ment	Content spread	Ave. cont.	Ele-ment	Content spread	Ave. cont.
H	-	0.610	Mn	56.0-990.0	377.0	Ba	20.0-7.060	950
Li	-	20.0	Fe*	0.5-24.0	3.9	La	2.9-28.0	13.0
B	-	20.0	Co	1.1-31.0	9.8	Ce	6.2-52.0	24.3
C	-	1.1	Ni	11.9-87.0	38.0	Nd	-	12.6
N	-	120.0	Cu	10.0-60.0	25.0	Sm	0.42-4.2	2.0
O	-	47.7	Zn	8.4-340.0	75.0	Eu	0.11-1.2	0.55
Na	0.0176-1.89	0.739	Ga	1.05-20.0	8.8	Tb	0.11-.78	0.41
Mg	-	0.0200	As	0.89-29.0	7.9	Dy	0.55-4.3	2.3
Al	0.53-6.1	3.1	Se	0.26-2.0	0.92	Ho	-	0.9
Si	3.9-32.4	16.8	Br	1.0-5.6	2.5	Yb	0.38-3.0	1.4
P	-	< 0.5	Rb	2.5-70.0	35.0	Lu	0.15-0.5	0.27
S	0.2-0.46	0.31	Sr	220.0-940.0	438.0	Hf	0.65-5.7	2.2
Cl	11.0-59.0	45.0	Y	3.0-96.0	18.2	Ta	0.09-0.9	0.44
K	0.047-2.5	0.75	Zr	27.0-160.0	71.0	W	0.39-3.9	1.4
Ca	8.8-34.7	18.3	Nb	1.3-9.3	4.3	Ir	-	< 0.01
Sc	0.73-17.4	6.5	Mo	1.8-36.0	10.3	Pb	5.4-560.0	61.0
Ti	0.023-0.79	0.212	Ag	-	< 0.2	Th	0.75-12.0	3.5
V	13.3-490.0	103.0	Sb	0.16-13.0	1.8	U	1.4-4.4	2.7
Cr	29.0-540.0	103.0	Cs	0.32-6.2	1.3	-	-	-

These data on the impurities were taken as the basis for the serpentinite concrete and thermal insulation of the plugs, the composition of the main elements for these materials being presented in Table 19.

TABLE 19. THE CONTENTS OF BASIC ELEMENTS IN SERPENTINITE CONCRETE AND IN MINERAL COTTON WOOL (THERMAL INSULATION IN THE PLUG), wt%

	H	O	Na	Mg	Al	Si	S	K	Ca	Fe	Dens., g/cm^3
Mineral cotton wool (heat ins.)	-	39.9	2.04	5.04	5.47	19.76	-	2.28	15.8	3.96	0.15
Serpentinite concrete [6]	1.27	48.5	-	15.5	1.45	18.6	0.036	-	-	5.68	2.2

Tables 20 and 21 show the expertise estimated contents of the impurity elements in graphite and 3-grade steel are presented.

TABLE 20. THE CONTENTS OF BASIC AND IMPURITIES ELEMENTS IN GRAPHITE, wt%

Elements	Concentrations	Elements	Concentrations	Elements	Concentrations
Li	5.8E-6	Co	5.9E-6	Sm	2.5E-5
C	~ 100	Ni	1.8E-3	Eu	7.6E-7
N	4.7E-4	Nb	7.7E-4	Hg	6.7E-5
Cl	1.3E-3	Ag	9.0E-6	U	2.0E-4
Ca	1.4E-2	Sn	4.9E-5	-	-
Fe	2.0E-3	Ba	1.1E-3	-	-

TABLE 21. THE CONTENTS OF BASIC ELEMENTS AND IMPURITIES IN 3-GRADE STEEL, wt%

Elements	Concentrations	Elements	Concentrations	Elements	Concentrations
Cr	0.25	Co*	0.021	Mo*	0.11E-2
Mn	0.25	Ni	0.25	Eu*	0.1E-4
Fe	~ 99	Nb*	0.25E-2	-	-

* Concentrations of elements assumed to be equal to those of Cr18Ni9 steel

5.6.2.3. Results of calculation of materials activity

Values of activity induced in the reactor materials and compositions of radionuclides produced under irradiation, vary in a wide range. For the purpose of convenience, analysis of calculation results is given for individual structural elements of the reactor and shielding, apart from analysis of activity values summed up for all radioactive reactor materials.

5.6.2.3.1. Activity of steel of in-vessel lateral shielding (band), shielding subassemblies and core diagrid

Tables 22 and 23 represent average specific activity values of nuclides contained in the steel of the band with the shielding subassemblies and in the steel of the core diagrid. The total activity values of steels depending on the duration of storage after irradiation are also presented in these tables.

TABLE 22. AVERAGE SPECIFIC ACTIVITIES OF NUCLIDES IN STEEL OF BAND AND SHIELDING SUBASSEMBLIES (Bq/kg) AND TOTAL ACTIVITY OF THE STEEL (Bq)

Nuclide composition	Storage duration, years					
	0	10	50	100	150	10 000
^{60}Co	2.52E+11	6.77E+10	3.50E+8	4.86E+5	2.39E+3	-
^{63}Ni	7.21E+10	6.73E+10	5.09E+10	3.60E+10	2.55E+10	-
^{59}Ni	1.19E+9	1.19E+9	1.19E+9	1.19E+9	1.19E+9	1.09E+9
^{94}Nb	1.26E+7	1.26E+7	1.26E+7	1.26E+7	1.26E+7	8.93E+6
^{55}Fe	3.56E+12	2.74E+11	1.30E+7	-	-	-
^{66}Mn	6.18E+12	-	-	-	-	-
^{51}Cr	1.23E+12	-	-	-	-	-
^{60m}Co	1.50E+11	-	-	-	-	-
^{59}Fe	7.73E+11	-	-	-	-	-
Ni^{65}	5.20E+10	-	-	-	-	-
^{58}Co	2.32E+12	-	-	-	-	-
^{58m}Co	1.15E+12	-	-	-	-	-
^{55}Cr	8.10E+10	-	-	-	-	-
^{52}V	7.37E+10	-	-	-	-	-
Sum, Bq/kg	1.79E+13	4.11E+11	5.26E+10	3.72E+10	2.67E+10	1.10E+9
Sum, Bq	6.13E+17	1.41E+16	1.80E+15	1.28E+15	9.15E+14	3.76E+13

TABLE 23. AVERAGE SPECIFIC ACTIVITY OF NUCLIDES IN THE STEEL OF THE CORE DIAGRID (Bq/kg) AND TOTAL ACTIVITY OF STEEL (Bq)

Nuclide composition	Storage duration, years					
	0	10	50	100	150	10 000
^{60}Co	2.33E+11	6.27E+10	3.23E+8	4.50E+5	7.48E+2	-
^{63}Ni	3.60E+10	3.33E+10	2.53E+10	1.79E+10	1.27E+10	-
^{59}Ni	2.60E+8	2.60E+8	2.60E+8	2.60E+8	2.60E+8	2.38E+8
^{94}Nb	2.13E+6	2.13E+6	2.13E+6	2.13E+6	2.13E+6	1.50E+6
^{55}Fe	6.29E+11	4.84E+10	-	-	-	-
^{66}Mn	3.55E+12	-	-	-	-	-
^{51}Cr	6.78E+11	-	-	-	-	-
^{60m}Co	1.20E+11	-	-	-	-	-
^{59}Fe	7.63E+10	-	-	-	-	-
^{65}Ni	9.09E+9	-	-	-	-	-
^{95}Nb	1.65E+9	-	-	-	-	-
^{58}Co	7.48E+8	-	-	-	-	-
^{64}Cu	6.05E+8	-	-	-	-	-
^{54}Mn	5.22E+8	-	-	-	-	-
Sum, Bq/kg	5.37E+12	1.45E+11	2.58E+10	1.82E+10	1.29E+10	2.40E+8
Sum, Bq	8.97E+16	2.42E+15	4.30E+14	3.04E+14	2.16E+14	4.01E+12

The analysis of results of calculation of induced activity of the bond steel, shielding subassemblies and core diagrid has shown that the value of activity of steel of these structures in the whole range of storage duration values exceeds the allowable level, at which no regulation of the radioactive material by the RSS (radiation safety standards) is required [25].

After several years of storage the activity is determined by ^{60}Co, ^{63}Ni, ^{59}Ni and ^{94}Nb. All these steel structures should be isolated from the human environment.

5.6.2.3.2. Activity of steel of thermal shielding, main and guard reactor vessels and supporting ring

Tables 24 and 25 present the average specific activities of nuclides in the steel of structures mentioned above. The total activities of steels depending on time of storage after irradiation are also given in these tables. The analysis of results of calculation has shown, that the induced activity of the significant portion of steel mass of the main reactor vessel and guard vessel (~ 80%) and all steel of the supporting ring exceeds the allowable level, at which the regulation of the radioactive material by the RSS is not required.

TABLE 24. AVERAGE SPECIFIC ACTIVITIES OF NUCLIDES IN STEEL OF THERMAL SHIELDING OF THE VESSEL (Bq/kg) AND TOTAL ACTIVITY OF STEEL (Bq)

Nuclide composition	Storage duration, years					
	0	10	50	100	150	10 000
^{60}Co	1.49E+11	3.95E+10	2.06E+8	2.87E+5	4.42E+2	-
^{63}Ni	2.81E+10	2.62E+10	1.99E+10	1.41E+10	9.98E+9	-
^{59}Ni	2.41E+8	2.41E+8	2.41E+8	2.41E+8	2.41E+8	2.20E+8
^{94}Nb	1.28E+6	1.28E+6	1.28E+6	1.28E+6	1.28E+6	9.11E+5
^{55}Fe	3.89E+11	3.01E+10	1.04E+6	-	-	-
^{56}Mn	2.10E+12	-	-	-	-	-
^{51}Cr	4.97E+11	-	-	-	-	-
^{60m}Co	7.75E+10	-	-	-	-	-
^{59}Fe	3.46E+10	-	-	-	-	-
^{65}Ni	5.56E+9	-	-	-	-	-
^{55}Cr	7.79E+9	-	-	-	-	-
^{52}V	2.95E+9	-	-	-	-	-
^{58}Co	4.45E+8	-	-	-	-	-
Total, Bq/kg	3.31E+12	9.61E+10	2.03E+10	1.43E+10	1.02E+10	2.21E+8
Total, Bq	1.78E+17	5.17E+15	1.09E+15	7.69E+14	5.47E+14	1.19E+13

TABLE 25. AVERAGE SPECIFIC ACTIVITIES OF NUCLIDES IN STEEL OF THE MAIN REACTOR VESSEL, GUARD VESSEL AND SUPPORT RING (Bq/kg) AND TOTAL ACTIVITY OF STEEL (Bq)

Nuclide composition	Storage duration, years					
	0	10	50	100	150	10 000
^{60}Co	2.32E+10	6.39E+9	3.31E+7	4.59E+4	6.30E+1	-
^{63}Ni	3.59E+9	3.13E+9	2.37E+9	1.67E+9	1.19E+9	-
^{59}Ni	3.52E+7	3.52E+7	3.52E+7	3.52E+7	3.52E+7	3.20E+7
^{94}Nb	2.13E+5	2.13E+5	2.13E+5	2.13E+5	2.13E+5	1.52E+5
^{55}Fe	5.12E+10	3.93E+9	1.37E+5	-	-	-
^{66}Mn	2.75E+11	-	-	-	-	-
^{51}Cr	6.16E+10	-	-	-	-	-
^{60m}Co	1.32E+10	-	-	-	-	-
^{59}Fe	5.12E+9	-	-	-	-	-
^{55}Cr	9.53E+8	-	-	-	-	-
^{65}Ni	7.34E+8	-	-	-	-	-
^{52}V	1.03E+8	-	-	-	-	-
^{58}Co	5.62E+7	-	-	-	-	-
^{58m}Co	2.82E+7	-	-	-	-	-
Total, Bq/kg	4.35E+11	1.35E+10	2.44E+9	1.71E+9	1.22E+9	3.21E+7
Total, Bq	4.42E+16	1.37E+15	2.47E+14	1.74E+14	1.23E+14	3.26E+12

The analysis of results of calculation has shown, that the induced activity of the significant portion of steel mass of the main reactor vessel and guard vessel (~ 80%) and all steel of the supporting ring exceeds the allowable level, at which the regulation of the radioactive material by the RSS is not required. After several years of storage the activity is determined by ^{60}C, ^{63}Ni, ^{59}Ni and ^{94}Nb isotopes.

5.6.2.3.3. Activity of steel of rotating plugs, plugs shielding structure and central column

The central column penetrates through the rotating plugs and shielding structure the activity at the bottom part making the main contribution to the total activity of steel of the reactor structures. Some steel sheets located in the bottom part of rotating plugs have the activity higher than the permissible limits of RSS.

The induced activity of graphite and thermal insulation is lower than the permitted value, i.e. no RSS regulation of radioactive material is required. Fifty years later the activity of rotating plugs will be lower than the permitted value, i.e. no RSS regulation of radioactive material is required. A hundred years later ~ 80% of steel shielding of rotating plugs will have an activity level, indicated above. Average specific activity values of nuclides in the steel of structures indicated above are presented in Table 26. In this Table the total activity values of steel depending on time of storage after irradiation are also presented. Value of the activity after several years of storage is determined by ^{60}C, ^{63}Ni, ^{59}Ni and ^{94}Nb isotopes.

TABLE 26. AVERAGE SPECIFIC ACTIVITIES OF NUCLIDES IN THE STEEL OF THE ROTATING PLUGS, SHIELDING STRUCTURE AND CENTRAL COLUMN (Bq/kg), AND TOTAL ACTIVITY OF THE STEEL (Bq)

Nuclides composition	Storage duration, years					
	0	10	50	100	150	10 000
^{60}Co	1.04E+9	2.78E+8	1.43E+6	1.99E+3	-	-
^{63}Ni	6.58E+8	6.14E+8	4.96E+8	3.30E+8	2.33E+8	-
^{59}Ni	6.20E+6	6.20E+6	6.20E+6	6.20E+6	6.20E+6	5.64E+6
^{94}Nb	4.0E+3	4.00E+3	4.00E+3	4.00E+3	4.00E+3	2.86E+3
^{55}Fe	7.04E+9	5.37E+8	-	-	-	-
^{56}Mn	1.75E+10	-	-	-	-	-
^{51}Cr	1.07E+10	-	-	-	-	-
^{60m}Co	5.78E+8	-	-	-	-	-
^{59}Fe	3.52E+8	-	-	-	-	-
^{55}Cr	1.33E+8	-	-	-	-	-
^{65}Ni	9.67E+7	-	-	-	-	-
Total, Bq/kg	3.78E+10	1.44E+9	4.74E+8	3.35E+8	2.39E+8	5.64E+6
Total, Bq	4.73E+15	1.80E+14	5.92E+13	4.19E+13	2.99E+13	7.06E+11

5.6.2.3.4. Activity of steel shielding, iron ore filling and steel lining of concrete

The activity of all shielding steel, located inside the iron ore filling, will exceed the level, which is critical from the standpoint of the radioactive material regulation by the RSS. A hundred years from then only steel of the concrete lining and that of the back wall of the reinforcing cage of the iron ore filling will have activity below the RSS limit value. Average specific activities of the nuclides in the steel of the structures mentioned above are given in Table 27. In this table the total activities of steel depending on time of storage after irradiation are also presented. Value of activity of 3 grade steel after several years of storage is determined by ^{60}Co, ^{63}Ni, ^{59}Ni, ^{94}Nb, ^{152}Eu, and ^{154}Eu isotopes.

TABLE 27. AVERAGE SPECIFIC ACTIVITIES OF THE NUCLIDES IN THE STEEL OF SHIELDING, IRON ORE FILLING AND CONCRETE LINING (Bq/kg), AND TOTAL ACTIVITY OF STEEL (Bq)

Nuclide composition	Storage duration, years					
	0	10	50	100	150	10 000
^{60}Co	7.65E+9	2.05E+9	1.06E+7	1.46E+4	2.04E+1	-
^{63}Ni	3.55E+7	3.3E+7	2.50E+7	1.77E+7	1.25E+7	-
^{59}Ni	3.88E+5	3.88E+5	3.88E+5	3.88E+5	3.88E+5	3.53E+5
^{94}Nb	8.99E+4	8.99E+4	8.99E+4	8.99E+4	8.99E+4	6.39E+4
^{152}Eu	7.69E+6	4.58E+6	5.71E+5	4.24E+4	3.15E+3	-
^{154}Eu	9.38E+6	4.17E+6	1.67E+5	2.97E+3	5.27E+1	-
^{55}Fe	2.84E+10	2.28E+9	7.52E+4	-	-	-
^{56}Mn	1.99E+10	-	-	-	-	-
^{51}Cr	3.46E+8	-	-	-	-	-
^{60m}Co	4.26E+9	-	-	-	-	-
^{59}Fe	2.79E+9	-	-	-	-	-
^{65}Ni	7.69E+7	-	-	-	-	-
Total, Bq/kg	6.33E+10	4.26E+9	3.66E+7	1.81E+7	1.29E+7	4.17E+5
Total, Bq	2.18E+16	1.47E+15	1.26E+13	6.23E+12	4.44E+12	1.44E+11

5.6.2.3.5. Activity of the iron ore filling

Iron ore filling serves as concrete shielding against radiation. After 50 years of storage, about 65% of the iron ore filling will have activity above limiting RSS values. A hundred years after the reactor shutdown the portion of the filling having such activity would not exceed 15% of its initial volume. Average specific activities of the nuclides in iron ore filling are shown in Table 28 In this table the total activity values depending on the time of storage after irradiation are also presented. Value of activity of the iron ore filling after several years of storage is determined by ^{60}Co, ^{152}Eu, ^{154}Eu, ^{94}Nb, ^{151}Sm, ^{14}C and ^{41}Ca isotopes [27].

TABLE 28. AVERAGE SPECIFIC ACTIVITIES OF THE NUCLIDES IN IRON ORE FILLING (BQ/KG) AND TOTAL ACTIVITY OF IRON ORE FILLING (BQ)

Nuclide composition	Storage duration, years					
	N	10	50	100	150	10 000
^{60}Co	2.22E+7	5.92E+6	3.07E+4	4.27E+1	-	-
^{63}Ni	1.94E+3	1.81E+3	1.37E+3	9.73E+2	6.85E+2	-
^{59}Ni	2.11E+1	2.11E+1	2.11E+1	2.11E+1	2.11E+1	1.92E+1
^{94}Nb	4.87E+2	4.87E+2	4.87E+2	4.87E+2	4.87E+2	3.47E+2
^{152}Eu	1.81E+5	1.08E+4	1.35E+3	9.93E+1	7.33E+1	-
^{154}Eu	1.77E+5	7.06E+3	6.50E+2	5.01E+1	-	-
^{151}Sm	1.59E+4	1.47E+4	1.08E+4	7.33E+3	4.97E+3	-
^{14}C	7.29E+3	7.25E+3	7.23E+3	7.19E+3	7.06E+3	2.12E+3
^{41}Ca	2.53E+2	2.52E+2	2.53E+2	2.53E+2	2.53E+2	2.36E+2
^{39}Ar	3.75E+2	3.66E+2	3.29E+2	2.89E+2	2.53E+2	-
^{55}Fe	2.66E+8	2.04E+7	7.06E+2	-	-	-
^{134}Cs	1.19E+6	4.14E+4	-	-	-	-
^{59}Fe	1.28E+7	-	-	-	-	-
^{60m}Co	1.24E+7	-	-	-	-	-
^{153}Sm	1.92E+6	-	-	-	-	-
^{42}K	1.13E+6	-	-	-	-	-
Total, Bq/kg	3.29E+8	2.64+E7	7.88+E4	1.62+E4	1.34+E4	2.73E+3
Total, Bq	4.28E+14	3.44+E13	1.03+E11	2.12+E10	1.74+E10	3.55E+9

5.6.2.3.6. Activity of the concrete

Just after the reactor shutdown the activity of about 381 m^3 of concrete exceeds the RSS limit values. After 100 years of storage the amount of radioactive concrete decreases down to ~ 8% of the initial value. After 120 years of storage the whole amount of concrete will have activity not exceeding the limiting RSS values. The average specific activities of nuclides in concrete are given in Table 29. In this Table the total activities depending on time of storage after irradiation are also presented. The concrete activity value after several years of storage is determined by Eu^{152}, Eu^{154}, Co^{60}, H^3, Ni^{63}, C^{14} and Ca^{41} isotopes.

TABLE 29. AVERAGE SPECIFIC ACTIVITIES OF THE NUCLIDES IN CONCRETE (BQ/KG) AND TOTAL ACTIVITY OF THE CONCRETE (BQ)

Nuclide composition	Storage duration, years					
	0	10	50	100	150	10 000
^{60}Co	3.55E+4	9.53E+3	4.92E+1	-	-	-
^{55}Fe	4.04E+5	3.09E+4	-	-	-	-
^{3}H	2.11E+5	9.91E+4	1.27E+4	7.64E+2	4.61E+1	-
^{152}Eu	3.52E+4	2.29E+4	2.86E+3	2.12E+2	1.57E+1	-
^{154}Eu	3.55E+3	1.75E+3	6.87E+1	-	-	-
^{41}Ca	1.69E+3	1.69E+3	1.69E+3	1.69E+3	1.69E+3	1.58E+3
^{63}Ni	2.87E+2	2.68E+2	2.03E+2	1.43E+2	1.02E+2	-
^{14}C	3.11E+2	3.10E+2	3.09E+2	3.06E+2	3.05E+2	9.30E+1
^{134}Cs	1.79E+4	1.30E+2	-	-	-	-
^{59}Fe	1.25E+4	-	-	-	-	-
^{136m}Ba	1.00E+4	-	-	-	-	-
^{139}Ba	7.93E+3	-	-	-	-	-
^{152m}Eu	2.49E+4	-	-	-	-	-
^{60m}Co	2.00E+4	-	-	-	-	-
Total, Bq/kg	7.90E+5	1.65E+5	1.78E+4	3.12E+3	2.20E+3	1.69E+3
Total, Bq	6.93E+1	1.45E+11	1.56E+10	2.74E+9	1.93E+9	1.48E+9

5.6.2.3.7. Activity of the graphite

Graphite placed in the plugs, has activity below the limiting RSS value. About 50% of graphite located in the ionization chambers unit (ICU), also has activity below RSS limit. After ~ 150 years of storage all graphite will have activity below RSS limit values. Average specific activities of the nuclides in graphite of ICU are presented in Table 30.

TABLE 30. AVERAGE SPECIFIC ACTIVITIES OF THE NUCLIDES IN GRAPHITE OF THE ICU (BQ/KG) AND TOTAL ACTIVITY OF GRAPHITE (BQ)

Nuclide composition	Storage duration, years					
	0	10	50	100	150	10 000
^{60}Co	1.75E+5	3.01E+4	1.56E+3	2.2E-1	-	-
^{152}Eu	2.88E+5	1.71E+5	2.13E+4	1.58E+3	1.18E+2	-
^{154}Eu	4.18E+4	1.85E+4	7.39E+2	-	-	-
^{94}Nb	4.38E+2	4.38E+2	4.38E+2	4.38E+2	4.38E+2	3.12E+2
^{63}Ni	6.85E+4	6.37E+4	4.84E+4	3.42E+4	2.42E+4	9.30E+1
^{14}C	1.05E+4	1.05E+4	1.04E+4	1.03E+4	1.03E+4	3.13E+3
^{59}Ni	6.27E+2	6.27E+2	6.27E+2	6.27E+2	6.27E+2	5.71E+2
^{41}Ca	6.32E+2	6.32E+2	6.32E+2	6.32E+2	6.32E+2	5.91E+2
^{134}Cs	3.67E+6	1.26E+5	-	-	-	-
^{55}Fe	1.04E+5	7.78E+3	-	-	-	-
Total, Bq/kg	4.36E+6	4.29E+5	8.41E+4	4.78E+4	3.63E+4	4.61E+3
Total, Bq	8.99E+10	8.85E+9	1.73E+9	9.85E+8	7.49E+8	9.49E+7

The total activities depending on time of storage after irradiation are also presented in this table. Graphite activity value after several years of storage is determined by ^{152}Eu, ^{154}Eu, ^{60}Co, ^{94}Nb, ^{63}Ni, ^{59}Ni, ^{14}C and ^{41}Ca isotopes. Total activities of the nuclides for all reactor materials are presented in the Table 31.

TABLE 31. TOTAL ACTIVITY OF THE NUCLIDES OF THE REACTOR MATERIALS (Bq)

Nuclide composition	Storage duration, years					
	0	10	50	100	150	10 000
^{60}Co	2.57E+16	6.89E+15	3.57E+13	4.96E+10	1.32E+8	-
^{63}Ni	5.04E+15	4.68E+15	3.55E+15	2.50E+15	1.78E+15	-
^{59}Ni	6.26E+13	6.26E+13	6.26E+13	6.26E+13	6.26E+13	5.27E+13
^{94}Nb	5.90E+11	5.90E+11	5.90E+11	5.90E+11	5.90E+11	4.18E+11
^{152}Eu	2.62E+12	1.58E+12	1.97E+11	1.46E+9	1.08E+8	-
^{154}Eu	3.23E+12	1.44E+12	5.75E+10	1.02E+9	1.81E+7	-
^{55}Fe	1.69E+17	1.30E+16	5.28E+11	-	-	-
^{56}Mn	4.21E+17	-	-	-	-	-
^{58}Co	7.96E+16	-	-	-	-	-
^{58m}Co	3.94E+16	-	-	-	-	-
^{51}Cr	8.80E+16	-	-	-	-	-
^{59}Fe	3.12E+16	-	-	-	-	-
^{60m}Co	1.49E+16	-	-	-	-	-
^{55}Cr	3.31E+15	-	-	-	-	-
^{52}V	2.72E+15	-	-	-	-	-
^{65}Ni	2.32E+15	-	-	-	-	-
^{95}Nb	2.75E+13	-	-	-	-	-
^{64}Cu	1.01E+13	-	-	-	-	-
^{54}Mn	8.70E+12	-	-	-	-	-
Total, Bq	8.82E+17	2.47E+16	3.65E+15	2.56E+15	1.84E+15	5.77E+13

5.6.2.3.8. Total activity of reactor materials and total volumes of radioactive waste

Table 32 gives total volumes and weights of the radioactive waste, formed as a result of neutron irradiation of reactor materials, having induced activity above RSS limit values. The main contribution to the amount of radioactive waste is made by the iron-ore filling and reactor structures made of stainless steel and 3 grade carbon steel. During 100 years of storage the amount of the radioactive waste having induced activity higher than the RSS limit value, will be decreased from ~ 860 m^3 (~ 2.900 t) down to ~ 150 m^3 (~ 680 t). After storage during 120-150 years the whole mass of concrete and graphite will have activity, at which these radioactive materials are free from the RSS regulation. The radiological conditions in the reactor vessel cavity will be determined by the products of activation of Co, Eu and Nb, contained as the impurities in structural and shielding materials of the reactor.

TABLE 32. VOLUMES AND WEIGHTS OF THE REACTOR MATERIALS, HAVING INDUCED ACTIVITY EXCEEDING RSS LIMITS

		Storage duration, years				
		0	50	100	150	10 000
Stainless	m^3	42.5	35.6	25.6	25.6	18.8
Steel Cr18Ni9	tons	331.5	277.7	199.7	199.7	146.6
3-grade steel	m^3	44.2	44.2	30.1	30.1	30.1
	tons	344.8	344.8	234.8	234.8	234.8
Iron-ore	m^3	376.3	240.3	56.0	56.0	42.0
Filling	tons	1303.0	720.9	168.0	168.0	126.0
Graphite	m^3	12.9	6.0	3.0	0.0	0.0
	tons	20.6	9.6	4.8	0.0	0.0
Concrete	m^3	381.0	114.3	30.0	0.0	0.0
	tons	876.3	263.0	69.0	0.0	0.0
Total for the	m^3	856.9	440.4	144.7	111.7	90.9
whole reactor	tons	2876.2	1616.0	676.3	602.5	507.4

Maximum dose rates of gamma-radiation on the surface of activated structures of the reactor are presented in Table 33.

TABLE 33. DOSE RATES OF GAMMA-RADIATION ON THE SURFACE OF ACTIVATED STRUCTURES, μSV/S

Elements of reactor installation		Time after reactor shutdown, years				
		5	10	50	100	150
Elements in the core midplane	Core area	1.5×10^5	7.5×10^4	4×10^3	1.0	0.5
	Reactor vessel	7.5×10^3	3.7×10^3	1.7×10^1	0.065	0.04
	Iron ore shielding	4×10^3	2×10^3	1.0	0.01	5×10^{-3}
	Concrete	0.08	0.033	5×10^{-4}	2×10^{-5}	-
Elements of lower reactor structures	Reactor vessel	4×10^2	2×10^2	1.0	3.5×10^{-3}	2×10^{-3}
	Iron ore shielding	4×10^2	2×10^2	0.16	7×10^{-4}	1×10^{-4}
	Concrete	0.8	0.33	5×10^{-3}	2×10^{-4}	10^{-5}
Elements of upper reactor structures	Bottom part of the "floating" shielding	1.5×10^3	7.5×10^2	3.7	7.5×10^{-3}	2.5×10^{-3}
	Top part of the "floating" shielding	1.5	0.75	4×10^{-3}	10^{-5}	6×10^{-6}
	Rotating plugs shielding	0.25	0.12	6×10^{-4}	10^{-6}	7×10^{-7}
Graphite of ICU of the control rods		6.5	2.3	0.08	8×10^{-3}	2.5×10^{-3}

It was assumed that dose rate of gamma-radiation was formed only due to activation of the material of the above indicated structure. The possible contribution to the dose rate of gamma-radiation made by the adjacent structures is not shown. When estimating the radiological conditions at the reactor top structures, it is necessary to take into account the increased amount of radioactive products deposits on the reactor structural elements having been in contact with the primary cover gas. The total values of activity of the materials after several years of storage are determine by ^{60}Co, ^{63}Ni, ^{59}Ni, ^{152}Eu, ^{154}Eu, ^{94}Nb and ^{55}Feisotopes.

Dose rate of gamma-radiation on the surface of structural elements only due to this contamination can reach ~ 2 μSv/s value, which is comparable to the effect of induced activity of structural materials located above the "floating" shielding. The analysis of data presented in Tables 29-33 shows that in case of short storage duration it is impossible to carry out dismantling of the reactor installation without application of robotics. The complete dismantling of the reactor installation without application of robotics can be carried out only after 100 years of storage.

5.6.2.4. Protection techniques

It follows from the analysis of the radiological characteristics of the equipment and technological media of the shutdown reactor, that in order to assure radiological safety of the personnel during the installation decommissioning it is necessary to meet the following two requirements:

- Maximum use of the operating systems of the reactor installation;
- Development of additional systems, allowing safe implementation of dismantling work.

The ratio between these directions depends on the chosen scenario of the reactor installation decommissioning. Below are given some specific recommendations, directed to assurance of the radiological safety of the BN-350 reactor decommissioning.

5.6.2.4.1. Use of operating systems of the reactor

On the stage of the BN-350 reactor decommissioning, the following systems required for the radiological safety assurance, should be used to the maximum extent:

- System of the radiological control;
- System of fuel element cladding failure detection and location;
- System of special ventilation;
- Systems of sodium fire fighting;
- System of special drains;
- Systems of fuel transport;
- Systems of reprocessing and disposal of the liquid and solid radioactive waste;
- "Hot" chambers for works with high activity products;
- Protection barriers and sanitary barriers;
- System of monitoring of the radiological situation on the NPP site.

The majority of these systems should keep their serviceability, practically at all stages of decommissioning. After removal of spent fuel and liquid metal coolant of the primary and secondary circuits and spent fuel storage cooling circuit from the BN-350 NPP, some systems, assuring radiological safety or their elements, such as the system of fuel element cladding failure detection, the monitoring system of inert radioactive gases and the system of sodium fire fighting can be dismantled.

5.6.2.4.2. Additional systems for assurance of radiological safety for putting in prolonged storage and dismantling of the reactor components

In order to assure radiological safety on the stage of putting in prolonged storage and dismantling of the reactor installation, development of new systems or modification of some existing systems is required:

— Creation of the additional sanitary barriers to prevent spreading of the radioactive contamination (for example, sanitary lock in the area of cells of the primary circuit loops, reactor dome, etc.;
— Replacement of closed ventilation systems of the primary circuit cells and reactor vessel cavity by a special ventilation systems having aerosol filters, gas being released to the atmosphere through the stack (it is necessary to exclude the ingress to the attended rooms of ozone and nitrogen oxides, caused by gamma-radiation of the high activity structural elements of the reactor). On the stage of putting reactor cavity structures in prolonged storage it is expedient to have ventilation system based on the natural draft provided by the stack;
— Creation of local systems of collecting, sorting, reprocessing, storage (putting in prolonged storage) and disposal of the solid radioactive waste, including high activity items (control rods and their sleeves, photo neutron sources, cesium absorbers, etc.);
— Development of the remote tools, robotics and devices for realization of dismantling works, including those carried out under water;
— Creation of an additional network of stationary instruments, intended for the control of superficial contamination of personal protection means, equipment, etc. by the radioactive products;
— Organization of system for analysis of radionuclide composition of material samples from the dismantled equipment (including α and β control). All these activities should be taken into account in the project of installation decommissioning and, as a part of it, in the report on the nuclear and radiological safety on the stage of the BN-350 reactor decommissioning.

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6. LEAD-BISMUTH COOLED (LBC) SHIP REACTOR

6.1. INTRODUCTION

The advantages of lead-bismuth and lead reactor cooling are high boiling temperatures and the relative inertness to water as compared with sodium. The melting and boiling points of sodium are 98 and 883°C, respectively. For lead-bismuth eutectic, these values are 123.5 and 1670°C, respectively, and for lead 327 and 1740°C at atmospheric pressure. In a lead-bismuth (Pb-Bi) or lead cooled reactor, the coolant boiling point may increase up to about 2 300°C because of high coolant pressure inside the core. However, the boiling points are well above cladding failure temperatures. The specific heat per unit volume of lead-bismuth and lead are similar to that of sodium but the conductivities are lower about a factor of four [1]. Studies of lead-bismuth cooled (LBC) fast reactors have been carried out in the Russian Federation organizations SSC RF IPPE (Institute of Physics and Power Engineering and EDO GIDROPRESS in which a great deal of experience has been accumulated in the course of the development and operation of submarine reactors cooled with lead-bismuth eutectic [1, 2]. The key results of operating experience of the propulsion nuclear steam supply system (NSSS) using lead-bismuth coolant, R&D on LBC reactor technology, as well as SVBR-75/100 reactor design features and parameters are discussed in this section.

6.2. OPERATING EXPERIENCE WITH LBC SHIP REACTOR PLANTS

In the course of operation of NSSS using LBC, accidents have occurred on three units, and that was the cause of impossibility of their further operation:

— The accident in 1968: the core was plugged by the oxides and products of steam/water interaction causing core partial meltdown [3];
— The accident in 1971: the damage of the primary circuit pipelines due to corrosion at their outer side and radioactive coolant leakage [4];
— The accident in 1982: the global corrosion damage of steam generator (SG) tube bundle caused by poor quality of feedwater; ~ 150 L of radioactive coolant leaked into the compartment due to personnel errors [5].

6.2.1. The reactor core meltdown

In the early stages of development, the formation of deposits of heavy metal oxide and other impurities posed problems. An uncontrollable accumulation of significant masses of oxides in the primary circuit could have formed when, during maintenance and repair operation, the pipelines of the primary circuit gas system were depressurized, and thus air penetrated into the primary circuit. Besides that, the primary circuit was contaminated by products of oil pyrolysis, which was used as lubricant for the pump shaft seal designed to prevent gas leaks from the primary circuit. The lubricant entered the primary circuit because of faulty seals. When the rate of SG leakage increased suddenly (it had started some time before the accident), the oxides accumulated and other impurities filled the core, which was the cause of the violent deterioration of heat removal. A negative temperature reactivity effect was the cause of transfer of the automatic power control rod to the upper switch terminal and spontaneous power reduction to 7% of rated value. This was the first symptom of the accident happened in May 1968.

However, operational documentation did not include any necessary instructions for the operator how to act when that kind of situation arose. Instead of resetting the emergency protection (EP) at the left side reactor, the operator followed the commander's directions (it occurred in the course of navy training) and tried to maintain the given power level by continuous removal of shim control rods (SHRs) out of the core. All reactivity reserve for 12 SHRs was released in about 30 minutes, though it was intended to provide for the power reserve generation for about 4000 efficient hours. When SHRs were pulled out, the fuel in the core area, where heat removal was deteriorated, melted and left the core together with the coolant flow. Signals of radiation hazard in the compartment that called for shutting down the reactor and removing the crew out being distantly removed from the reactor were not taken into account. After this accident, the work on the coolant technology problem has been launched. For many years, this work has been carried out at the number of organizations under the scientific supervision of SSC RF IPPE. As a result, the problem has been solved successfully and it was corroborated by the many-years experience of further NSSS operation. The following main technical measures have been developed for eliminating the causes of such accidents:

- In order to eliminate the accumulation of oxides, excess inert gas pressure was maintained in the primary gas system in case of equipment repair and reactor refueling.
- In order to eliminate the possibility of air penetration into the primary circuit and the radioactivity release to the environment, the most possible tightness of the primary circuit has been provided. For this purpose, special repair and refueling equipment have been developed;
- The sensors of thermodynamic oxygen activity enabling control of the content of oxygen dissolved in LBC and detection of alloy oxidation at the very early stages have been designed and introduced;
- Rejecting the use of the oil seals of the pump shafts and adoption of water seals or gas-tight electric drives of the primary pumps. This eliminates oil penetration into the primary circuit and contamination of LBC by the products of oil pyrolysis;
- Using the ejection system for high-temperature hydrogen regeneration built-in into the NSSS in order to ensure chemical recovery of lead oxides by hydrogen (the explosion proof compound of helium and hydrogen is used) and enable, if necessary, purification of even strongly contaminated circuit from lead oxides;
- Sing continuously operated system of coolant purification from irreducible impurities on the glass fabric filters;
- Using automatic system of coolant quality control equipped with sensors of continuous control of coolant and cover gas quality ensures preservation of oxide films on the surface of the primary circuit structural materials contacting with the coolant, eliminates their corrosion deterioration and ensures the early diagnostics of abnormal conditions.

6.2.2. Radioactive coolant leaks caused by damage to the primary pipelines

Since the beginning of the NSSS tests in 1970 and its further operations in 1971 and 1972, its operation has been accompanied by a higher content of moisture in the air of the tight compartment (TC) where the NSSS was installed. The tests have shown that the causes of moisture content increase were poor air-tightness of the seal of one SG cover because of the flaw in the nickel gasket (which was later replaced) and steam leak through the steam heating system welds, which have been made unsoundly, and there was no possibility to eliminate this leakage because of tight assembly.

As a result of cold surface sweating inside TC, water drops were the cause of wetting the thermal insulation and “dry” protection materials containing chlorides. The drops of water saturated with chlorides touched the primary circuit hot auxiliary pipelines made of austenitic steel and gave rise to their corrosion cracking on the outer surface that has been fully verified by the results of the NSSS inspection performed. Corrosion damages of the primary circuit auxiliary pipelines at two of three loops and impossibility of their repair because of compact assembly caused the decision of removing this unit out of the Navy and carrying out the NSSS inspection. Below are described engineering measures eliminating the causes of such accidents.

In the designed advanced NSSS, pool type primary circuit arrangement has been used. This fully eliminates any primary circuit pipelines extending from unit vessel including comparatively thin-wall auxiliary pipelines of small diameter, and no valves. Therefore, ramified steam heating system was eliminated. Manufacturing the NSSS monoblock unit under the plant conditions ensures high quality and delivery of the reactor unit available for operation. The integral (pool) arrangement almost completely eliminated the possibility of coolant leakage.

6.2.3. Radioactive coolant leaks owing to corrosion damage of the steam generator

Global corrosion damage of the SG evaporator sections tubes made of perlitic steel occurred as a result of disregard of requirements to water-chemical regime (WCR) for SG feed water. It was the result of that under real operating conditions, the way of reducing oxygen content in the feed water by electron-ion-exchanging filter with copper-containing charge provided in the design, caused copper release to the secondary circuit that was the cause of severe electric-chemical corrosion of the piping system of the SG evaporator sections.

As a result of through piping damage, steam from the secondary circuit began to penetrate into the primary circuit, where after separation from coolant it condensed in the emergency condenser (EC) specially provided in gas system for the case of leakage in the SG. As EC internal cavity had been filled with the condensate step by step, a signal was produced to the operator who repeatedly drained the EC by removing the condensate accumulated to the appropriate reservoir, and thus he eliminated the essential pressure increase in the primary circuit gas system. However, the EC drainage was stopped because of unclear reasons. Heat transfer surface of the EC was completely flooded by water and the condensation of penetrating steam stopped.

Pressure increase began in the primary gas system. The strength of the gas system and primary circuit could bear full working pressure of the secondary circuit. That is why in that case there could not be any loss of integrity of the primary circuit.

Nevertheless, loss of integrity occurred, and it was caused by the events described below. The gas pocket in the leakage re-injection pump (LRP) located in the pump tank below the LBC level had adjusting manometer with ultimate pressure of 4 kg/cm^2. According to the instruction, if the RI was in operation, this manometer had to be shut off by the valve. The instruction requirement was violated and the valve occurred to be open. Due to this fact, when steam pressure in the gas pocket of the LRP tank reached ~ 6 kg/cm^2 value and the LBC level in the internal pocket of the LRP increased with the increase of gas pressure, the sensitive manometer element was destroyed, gas escaped from the pump pocket, that was the cause of filling up gas pocket of the LRP with lead-bismuth alloy and its further leakage through the damaged manometer into the inhabited section of the reactor compartment (Fig. 1).

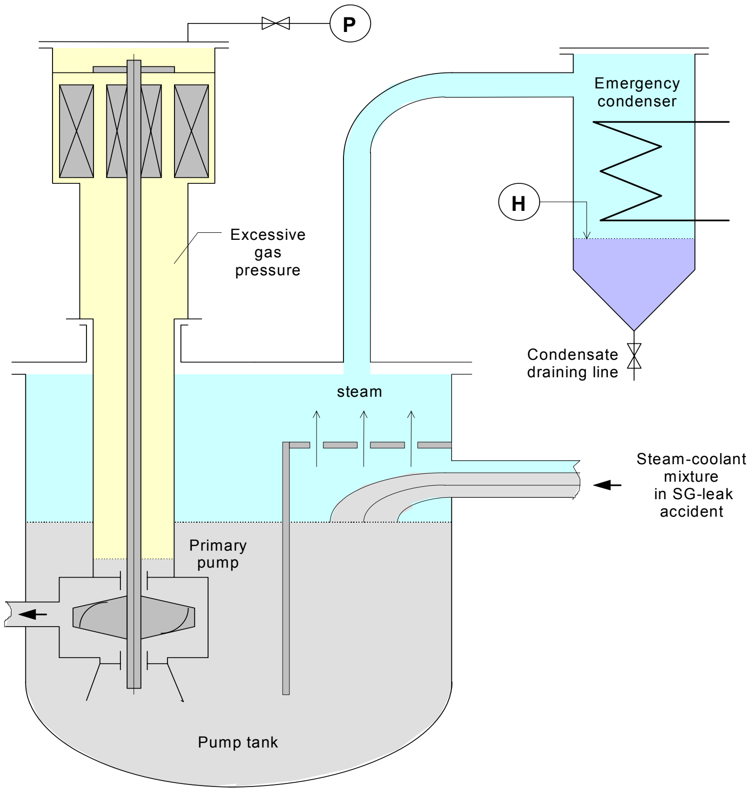

FIG. 1. Primary pump with auxiliary system: flow diagram.

Air contamination by ^{210}Po aerosols reached 10 maximum permissible concentration (MPC). Due to the following proper actions, the crew irradiation and radioactive contamination were within the permissible limits. The analyses of crew bio-samples, which had been performed by medical service, demonstrated that none of the crew had the content of ^{210}Po more than 10% of maximum permissible value.

Reactor installation (RI) examination showed that it could be reconditioned. However, another decision was accepted. It was decided to replace the whole reactor compartment of this unit with a new one manufactured earlier. The motive for this decision was as follows: in the course of this RI fabrication at the machine-building plant, there was faulty replacement of the relevant piping made of high-nickel corrosion-resistant steel by the same size piping made of common stainless steel. This error was found out after the NSSS was fabricated and it was impossible to change the piping. Since service lifetime of stainless steel piping was restricted by corrosion, a decision was made to limit service lifetime of the reactor unit by 25 000 hours and fabricate a reserve NSSS unit in order to use it for the replacement of off-spec one in the course of the submarine overhaul period. In 1982, service life of stainless steel piping had to be expired, and that was the motive for replacing reactor the compartment. The analysis performed showed that the cause of this accident discussed in the previous section did not relate to the use of LBC. The following technical measures ensure the elimination of such accidents at the new NSSS generation:

— Elimination of use of copper-containing materials in the water-steam circuit;
— Preferable use of corrosion-resistant steel for SG water-steam tubes instead of low-alloyed perlitic steel;
— Providing passive drainage of the emergency condenser (EC) when filling it up by the condensate up to the given level;

— Uniting gas volume inside the pump electric motor with the total one above the free coolant surface;
— Providing great extent of control-fitness and repair-fitness of the NSSS.

6.3. R&D ON LBC REACTOR TECHNOLOGY

Among the key problems, the problem which have been solved in the course of design and operation of this type installations was that of LBC technology, i.e., development of systems and devices ensuring measurement and maintenance of the LBC quality required during its long-time operation both under normal conditions of leak-proof circuit and in the case of partial loss of integrity of the circuit during repair and reactor refueling. Functioning of those systems and devices are necessary for eliminating structural material corrosion and circuit slagging by the lead oxides [6]. It should be pointed out that in the early days of mastering LBC, when the necessity of developing and implementing measures on the coolant technology had not been realized, there were cases of reducing the coolant flow cross sections down to the full blockage of coolant flow because of depositing lead oxides and other impurities and all the resulted consequences.

The basic kind of corrosion damage, which is the most dangerous for structural materials both in Pb-Bi and in Pb coolants, is local corrosion of materials appearing as the separate corrosion-erosion centres (“pittings”). Local through corrosion damages of structural elements may appear at temperatures over 550°C after holding for some hundred hours under the following conditions: unbalance of alloying elements and impurities in steel, poor quality of metal, absence of coolant quality control and non-optimal coolant flow regimes. The typical corrosion rate in such cases is estimated as 2.55 mm/year.

The principle solutions ensuring high corrosion resistance of structural materials in heavy liquid metal coolant were found using oxygen dissolved in the coolant. It has been shown as a result of long-term studies that this corrosion resistance essentially depends on concentration of dissolved oxygen.

Upon reaching certain level of concentration of dissolved oxygen corrosion processes is stopped due to protective oxide film formed on the steel surface. At high temperatures an indispensable condition of corrosion inhibition is presence of silicon in steel as additional alloying element. The silicon content in steels is varied within 1-3.5% range depending on steel type. Oxide films formed on the steel surface prevent it from interaction with liquid lead. Since breakdown of oxide films is possible during operation, precautions must be taken for resuming and maintaining their thickness and density.

Thus, steel corrosion in molten lead can be significantly slowed down by the oxide film formed on the steel surface. The main technological problem is maintaining such oxygen content in the coolant which, on the one hand, would provide stability of oxide film (Fe_3O_4) on the steel surfaces, but, on the other hand, would preclude generation of lead oxide (PbO) in the coolant, that could result in the circuit slagging. There are some ranges of content of oxygen dissolved in lead meeting these two conditions, for instance ($\sim 5\times10^{-6}$-10^{-3} wt %) range. Oxygen content in lead can be controlled either by injecting gaseous oxygen or by dissolving solid PbO.

Required oxygen content in lead can be maintained in two ways: (a) bubbling of argon, hydrogen and water vapour mixture or gaseous oxygen through molten lead; (b) lead oxide filling through which molten lead is pumped. In order to change oxygen content and remove surplus PbO, reactions with water vapour or hydrogen can be used. To determine oxygen content in molten lead (similarly to Pb-Bi technology development) galvanic cell can be used.

The problem of hyperthermal corrosion resistance of structural materials was got over by development of preliminary protective coatings for the working steel surfaces. In particular, the most important structural units of circuit, e.g. fuel rod claddings and steam generator tubes, are covered by these coatings at the final stage of their manufacture. Additional barriers are also formed directly on the inner surfaces of liquid metal circuit under effect of the coolant in the early stage of the reactor operation.

The best results of using technology of preliminary oxidation of circuit components were achieved by application of media with low partial pressure of oxygen, namely, Pb-Bi-O, $H_2O + H_2$ and CO_2. These methods, first of all, make it possible to avoid critical kinetic stage of preliminary passivation of uncoated surfaces of steel structures of the circuit. Moreover, they prove to extend the range of permissible decrease of oxygen concentration in the coolant. Therefore, the basic factors ensuring high corrosion-erosion resistance of structural materials in heavy liquid metal coolant (Pb, Pb-Bi) are as follows:

- Application of silicon alloyed steels;
- Passivation by oxygen using special regime of coolant;
- Using additional corrosion barriers such as oxide films formed on working surfaces of circuit components under reactor start-up conditions.

In reaction of PbO reduction, water vapours are efficiently removed from the circuit. Small amount of moisture acts as diluted oxidizer preventing from achieving reduction conditions for oxide films on the steel surface. Parameters of all these processes have to be developed with necessary control of hydrogen content in cover gas and oxygen activity in liquid lead.

In conclusion, it appears that corrosion resistance of structural materials have been ensured by using special steel alloys, applying protective films to them in advance and maintaining necessary concentration of corrosion inhibitor - dissolved oxygen - in LBC. The importance of these measures has been corroborated by the fact that when the necessary coolant quality was maintained, for several thousands of hours at 650°C temperature, there was no corrosion of the fuel element steel cladding. However, when dissolved oxygen concentration was inadequate (under specially provided conditions), it took about 20 hours for through corrosion damage of the 5 mm thick pipe under the same temperature [7, 8].

Melting point of LBC is about 124°C. Maintaining liquid state of LBC in all NSSS operation regimes is ensured by using SG with multiple circulation over the secondary circuit, besides that, water temperature at the SG inlet is higher than LBC melting point. For initial heating-up and maintaining the primary circuit under hot condition at a low power level in the core, a system of steam heating or electrical heating may be used.

The substantiation of the possibility of multiple coolant "freezing-unfreezing" in the NSSS was an important practical problem. Low shrinkage of LBC during solidification and rather high plasticity with low strength in solid state facilitate the elimination of reactor plant damage when alloy is transiting from liquid to solid form and its further cooling down to the ambient temperature. A special order of the temperature-time heating regime has been developed for safe reactor plant "unfreezing". This problem is dealt with in paper [9].

The following characteristics have been obtained in the course of NSSS test and operation:

— The specific feature of LBC is the formation of α-active ^{210}Po radionuclide with a half-life of ~ 138 days when bismuth is irradiated with neutrons.
— The major reason for its radiation danger is the formation of radioactive polonium aerosols when hot LBC contacts with air. It could happen under conditions of emergency integrity loss of the primary circuit and coolant spillage. In this case, as the NSSS operating experience in the nuclear plant has shown, the yield of polonium aerosols and air radioactivity (according to thermodynamics laws) reduces rapidly with temperature decrease and spilled alloy solidifying. Rapid solidification of spilled LBC restricts the area of radioactive contamination and simplifies its removal in the form of solid radioactive waste.
— Low polonium concentration in the coolant (about 10^{-8} at.%) and formation of thermodynamically proof chemical polonium-lead compound additionally reduces the polonium pressure by 1000 times.
— Personnel individual and collective protection facilities, methods for equipment decontamination and recording activity on the surfaces, as well as, performing repair procedures (also in case of LBC penetrating into the secondary circuit as a result of unit crew errors) have been developed.

There have been no cases of personnel overdose by this radionuclide above the permissible limits. This positive practical result is in a good agreement with conclusions of foreign experts who have investigated polonium hazard problem if LBC is used for nuclear reactor cooling [10, 11]. Reference [12] has focused on the analysis of this important issue.

In summary, the following characteristics have been obtained in the course of NSSS tests and operation: power and parameters of installation, core lifetime, reactivity margin, reactivity coefficients, poisoning effects, temperature distribution, dynamic parameters, coolant radioactivity, and dose rates of caused by neutron and γ-radiation behind the shield. They were in sufficiently good agreement with calculation results.

Among the positive properties of the reactor plant using LBC, which have been discovered in the course of operation, one can point out the following: the simplicity of control, high maneuverability and short time of reaching the power regime out of subcritical reactor state, the possibility of NSSS operation if there is small leakage in the SG pipe system, high repair-fitness of the SG by plugging the depressurized pipes, the possibility of reactor plant stable operation at any low power levels, the possibility of quick changing the circulation regime of coolant with essential change of its flow rate, and almost complete generation of designed power by cores under normal and acceptable conditions of tightness of fuel rod claddings.

6.4. CONCLUSIONS

In the early stages of development, the formation of deposits of heavy metal oxide and other impurities posed problems. A careful control of the purity of the coolant is required to avoid the formation of such deposits. It was necessary to improve corrosion resistant steels and pre-treat the surface of components, and also to use special inhibitors in the lead-bismuth coolant.

Lead-bismuth alloy is inflammable in air or water. In principle, lead-bismuth cooled reactor would not have to have an intermediate circuit separating the primary coolant and water/steam. However, as it has been mentioned above and emphasised in Ref. [3], there have been incidents with lead-bismuth cooled reactor. Some areas of the reactor core were plugged by lead oxides and other impurities as well as by the products of water and lead-bismuth interaction due to SG leaks, causing meltdown of the core. Therefore, the elimination of the intermediate circuit needs additional R&D efforts.

It was found [13] that specific α -activity of the typical lead - bismuth coolant is defined by ^{210m}Bi (half-life = 3.6×10^6 years), generated in reaction ^{209}Bi (n,γ) ^{210m}Bi. The long-lived β-activity of ^{208}Bi (half-life = 3.65×10^5 y) is produced in reaction ^{209}Bi (n, 2n) ^{208}Bi. Thus, the residual activity of lead-bismuth coolant is expected to be as high as millions of years. As it is pointed out in [13], that purification of lead-bismuth from the long-lived radionuclides would be too expensive[9].

Eight submarines with lead-bismuth reactors have been constructed in the former USSR [14]. In the course of operation of NSSS with lead-bismuth reactor, there were accidents on three units that have resulted in premature closing down [4].

The production of α-radioactive ^{210}Po having 138 days half-life undergoes α-decay, some problems are caused by bismuth because of its migration from the coolant to the cover gas and formation of aerosols. ^{210}Po is volatile, so that the leakage from the cover gas poses some hazard to the plant operators [1].

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7. LMFR CHARACTERISTICS AND THEIR DEMONSTRATION DURING THE FINAL STAGE OF OPERATION

7.1. PASSIVE CHARACTERISTICS WITH REGARD TO REACTOR SHUTDOWN

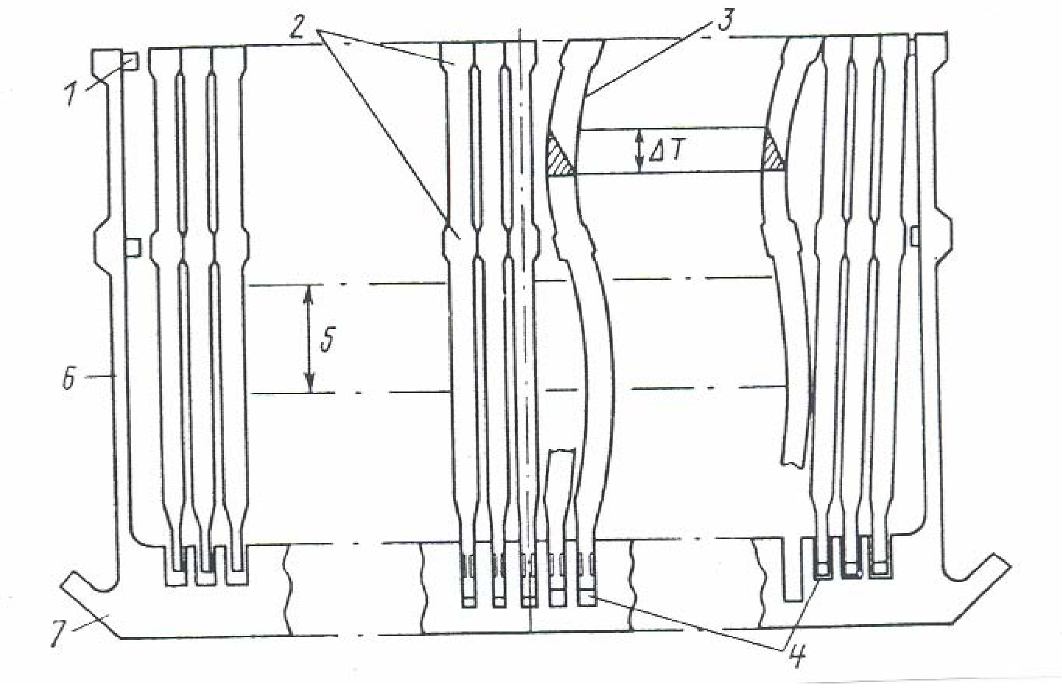
The history of industrial development suggests that complex engineering problems are often solved in simple ways if the subject's intrinsically inherent properties of self-regulation are grasped and brought into play in a timely manner. If any permissible limits of the operating parameters - having appeared at first glance to be critical from the standpoint of normal conditions - are expanded, the difficulties are overcome by virtue of natural feedbacks. As a rule, this has led to refined and technologically simple solutions.

In this context, the range of temperatures permitted under emergency conditions for the fast reactor as an important issue. The experience which has been acquired through the operation of experimental reactors and rigs demonstrates that the temperature of the coolant and structural elements of a reactor, which are made of high-temperature stainless steels, can be increased appreciably for a short time (a period of a few hours) without negatively affecting either strength characteristics or serviceability. For a fast reactor, this is possible due to the thermal properties of the sodium coolant: a high boiling margin (300°C), the constancy of the efficient cooling of fuel element cladding in the case of a dramatic coolant velocity decrease in the core, and the low level of the corrosive and chemical activity that exists between sodium, the structural material (stainless steel), and almost all fuel materials at temperatures that appreciably exceed the operating temperature.

As calculations and experiments demonstrate, allowance for an appreciable short-term temperature increase within a reactor is of decisive importance to the realization of maximum values for radial and axial core expansion, the ”centrifugal” bending of the subassemblies and the elongation of the absorber rod transfer bars, and the corresponding movement of the absorber rods into the core. The formation of negative reactivity in the core gives rise to the development of self-regulation factors. When the design process is target-oriented, this in turn results in reducing reactor power to the decay heat level, even in the case of failure of the safety rods to operate and reactor heating-up as consequence of pump shutdown and ceasing of feed water supply to the steam generators. In other words, allowance for possible peak temperature conditions within a reactor and the concomitant appearance of self-regulation effects can, in principle, make the reactor sub-critical without scram.

Increased temperatures of the coolant and the “external” structural elements, as exemplified by the main and guard vessels, go a long way toward laying the foundation for heat removal based on passive principles. In such situations, heat can be discharged into the atmosphere as a result of natural air circulation in the space between the vessel's hot wall and the well in which the reactor or steam generator is located, as well as into special “sodium-air” heat exchangers that also operate on the principle of natural circulation.

The core of a fast reactor is characterized by negative power reactivity coefficients that result from both axial core expansion and a negative Doppler coefficient. The latter is especially noticeable in the presence of considerable burnup, when the fuel adheres to the cladding and expands with it. An increase of core power and coolant temperature rise in the core brings a reactivity effect into play that result from the S-shaped “centrifugal” bending of the subassemblies in the core region. It is suggested that when the assembly is spaced in a height-wise fashion at three points (an articulated joint where the subassembly joins the housing of the core diagrid, a spacer plate at the upper edge of the core, and a spacer plate at the upper end of the subassembly), its bending leads to a negative reactivity effect (Fig. 1).



1-core former; 2-duct load pads; 3-S-shaped configuration due to contact of the load pads; 4-articulated joint of subassembly foot with diagrid; 5-fuel; 6-core barrel; and 7-core support structure

FIG. 1. Sketch of the restraint and spacing of the fuel assemblies for enhancing the negative feedback reactivity during elevation of the temperature.

As with other measures, reactor design optimization for the purpose of intensifying negative reactivity feedback also affects the selection of method of the vessel support in a pool type reactor. The conclusion is reached, that a vessel bottom support and the co-location of the rotating plugs and the control rod drive column on the cover plate could be the optimum. In this case, the effect of the elongation of the safety system absorber rod transfer bars is utilized to the fullest extent possible so as to facilitate the easy insertion of these rods into the core.

The self-regulated reactor concept makes allowance for the possible introduction of positive reactivity due to the erroneous withdrawal of the control and/or safety rods.

Self-actuating absorber rods are being developed for the purpose of preventing serious residual effects in advanced LMFR reactor designs.

Such absorber rod designs make use of electromagnetic clamps and transfer bars that are located in the coolant. The electromagnet's windings are powered by a heat-sensitive transformer that is mounted on the rod guide tube above the core. A stream of hot sodium from the core flows around the transformer. When the temperature of the transformer's core reaches the threshold value (Curie point), it loses its magnetic properties, the connection between the primary and secondary windings is broken, and the supply of electric power to the electromagnet's winding is terminated, which causes the rod to detach from the transfer bar and drop into the core. Alternatives involving hydraulically suspended safety rods are being examined for the purpose of resolving similar problems with respect to pump failure.

Other engineering measures that are aimed at reducing the positive component of reactivity feedback include the use of a neutron absorber that is dissolved in the primary circuit coolant. This alternative is similar to the concept employed in thermal reactors where boric acid is dissolved in water. In such designs, the rapid absorber elimination and reactor runaway become physically impossible. Indium is being considered possible absorbing solute. The melting point of indium is 156°C, i.e. about 60°C higher than that of sodium. The requisite indium concentration in the sodium can be maintained over a wide range of temperatures via the crystallization of sodium-indium alloy in the cold traps. At the suggested indium concentration, the melting point of the sodium-indium alloy does not exceed 204°C, as a result of which the solidification of the coolant in the loop is prevented. The indium dissolved in the sodium is gradually retained in the cold traps over the course of reactor operation, which offsets the reactivity decrease associated with fuel burnup. The neutron capture cross-section of indium is equal to 0.45 barns for the neutron spectrum of an LMFR reactor, or 20% of the capture cross-section of ^{10}B. For example, the indium content needed in the sodium in order to offset 10 dollars excess reactivity in the 400 MW (e) reactor is 6.5 atomic percent. The interaction occurring between a liquid sodium-indium alloy and austenitic steel was investigated on a test bed represented by a closed loop made of the 304 steel. The welding joints present in this loop made it possible to study the influence exerted by indium compounds on the welding material as a function of microstructure. Tests performed for a period of 2 000 hours at a steel temperature of 540°C demonstrated the absence of any noticeable structural material penetration by the indium. The welds and adjoining sections were not excluded from the overall pattern. Indium-nickel-manganese compound crystals were detected in the cold sections of the loop. These crystals precipitated onto the walls of the piping. Based on this finding, the conclusion can be reached that the only type of interaction engaged in by indium that has been precipitated from the coolant is the reaction occurring between indium and nickel that might be present.

7.2. PASSIVE CHARACTERISTICS WITH REGARD TO EMERGENCY HEAT REMOVAL

The process of advancing nuclear-power engineering is accompanied by the inevitable complication of reactor systems and equipment. As a result, the structure and complement of components of the nuclear power plant (NPP) are more complex than those of a fossil fuel power plant. For example, the NPP equipped with a light-water reactor having an electrical capacity of 1000 MW contains about 30 thousand valves (gate and other types) as compared to 4 thousand valves in a fossil fuel power plant having identical capacity.

NPP development has resulted in two challenges:

(i) Heat removal systems that are unrelated to nuclear processes must be designed and manufactured in compliance with strenuous engineering standards, and
(ii) The complexity and multifunctional nature of the systems involved have increased the likelihood of failures, thereby a high degree of quality control is needed. This has resulted in a concomitant cost increase. In principle, the above mentioned complexicity is also likely to increase the possibility of human error and of automatic system malfunctions.

Among the specialists of various countries, the race is on to create emergency heat removal systems that are based entirely on passive features and do not require intervention on the part of an operator or an automatic system. Being based on precise physical laws, such systems would function continuously. However, their capability would be intensified if a menacing deviation in limiting parameters were to happen; thus, the parameters of the reactor and any associated equipment would be maintained within predetermined limits by virtue of self-regulation effects.

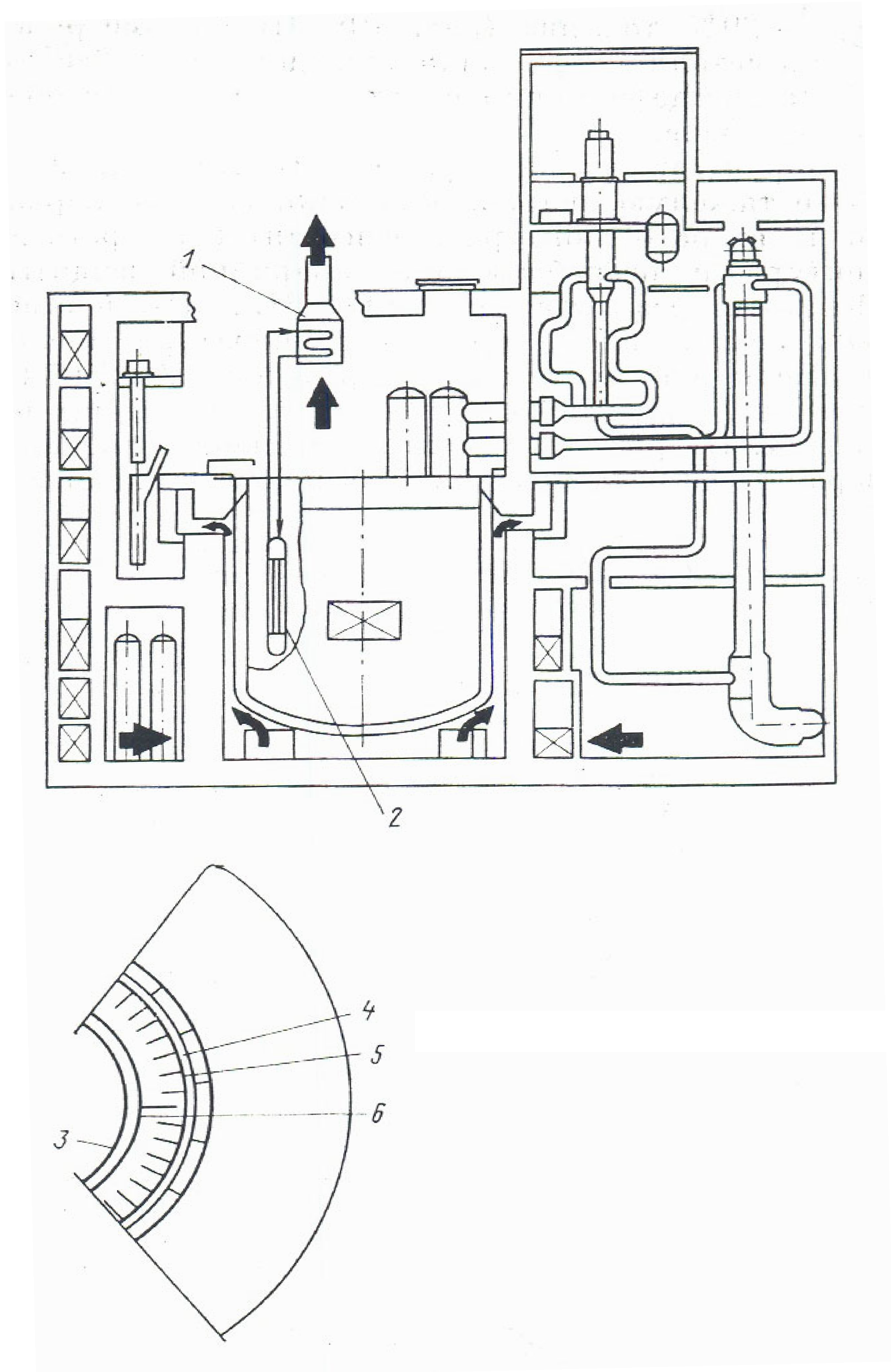
The nuclear power plant operator must concentrate attention on observing the course of normal processes and equipment performance, as well as on preventing minor malfunctions from turning into major accidents. The provision of emergency shutdown cooling in fast reactors based on the use of passive features alone without bringing the main heat removal systems into play — i.e. the elimination of secondary circuit and tertiary circuit systems from this process — does away with a number of the strenuous and costly requirements that would otherwise be imposed on the design, manufacture, and operation of such systems. As previously mentioned, this makes the reactor in particular and the power block as a whole less expensive. The essence of passive features with respect to fast reactors consists of making use of the natural circulation of the coolant and the dissipation of the heat into the environment via thermal radiation, thus removing the heat from the core and the reactor vessel.

It should be noted that the possibility of employing passive features in systems that are essential to safety has already been given consideration in early reactor designs. However, full use was not made of these features. As previously mentioned, special active safety systems that are switched on by an operator or by an automatic system when the threat of an emergency situation is impending have performed and continue to perform protective functions. Special systems are also employed. In all the designs that have been developed up to date, passive factors of are represented by:

— An extended pump coast down time (due to the use of special flywheels), with the transition being made to coolant natural circulation, and the incorporation of special emergency shutdown-cooling heat exchangers into the vessel of pool type reactors. These heat exchangers are connected to the air heat exchangers by means of external sodium lines. In all three loops, heat removal is accomplished by means of natural circulation owing to the height-wise separation of the emergency shutdown-cooling heat exchangers and the air heat exchangers (Fig. 2).

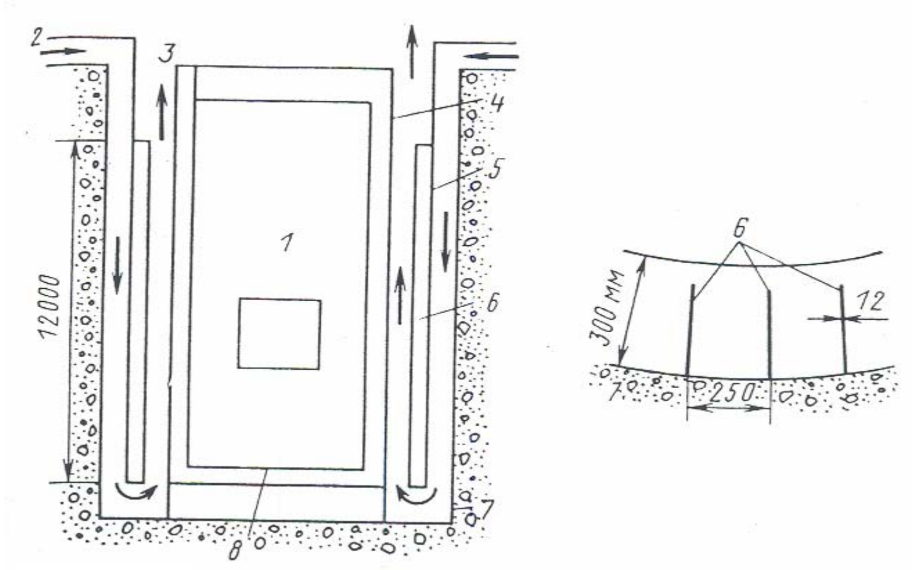
— The removal of heat from the guard vessel via thermal radiation and convection on the part of the air circulating through the annular space between the vessel and the reactor well lining (Fig. 3). In the presence of a sodium volume increase that results from heating up in the upper chamber, the sodium overflows into the annular cavity and streams downward along it, transmitting heat through the vessel wall, the gas space, and the guard vessel's wall to the outside air (Fig. 4). Design measures provide the requisite natural sodium circulation flow rate within the reactor. In order to ensure natural air circulation, the annular well space is connected to a short ventilation pipe. Power dissipation in this case does not exceed 0.4% of the reactor's thermal capacity.

A heat removal system based on the external air cooling of the reactor's backup vessel is examined. The fining of either the guard vessel or the well lining is being considered for the purpose of intensifying heat removal by the air. It has been proposed that a copper with a linear expansion coefficient identical to that of steel, a high degree of thermal conductivity, and a high level of blackness (as a result of surface oxidation) should be used as the fin material. These fins would be fashioned on the well lining. One potential problem might be the fouling and clogging of the space between the fins by impurities present in the air. A positive natural circulation propelling head is created by means of heating the air present in the space between the guard vessel and the finned well lining.



1-air heat exchanger; 2-heat exchanger for emergency cooling; 3-core; 4-insulation; 5-finned shell; 6-reactor vessel

FIG. 2. The system for emergency cooling of sodium advanced fast reactor (SAFR) based on passive features.



1-core; 2, 3-inlet, and outlet respectively; 4-guard vessel; 5-finned shell; 6-fins; 7-concrete; 8-vessel

FIG. 3. Passive heat rejection system from the vesse1 of the power reactor innovative small module (PRISM) reactor.

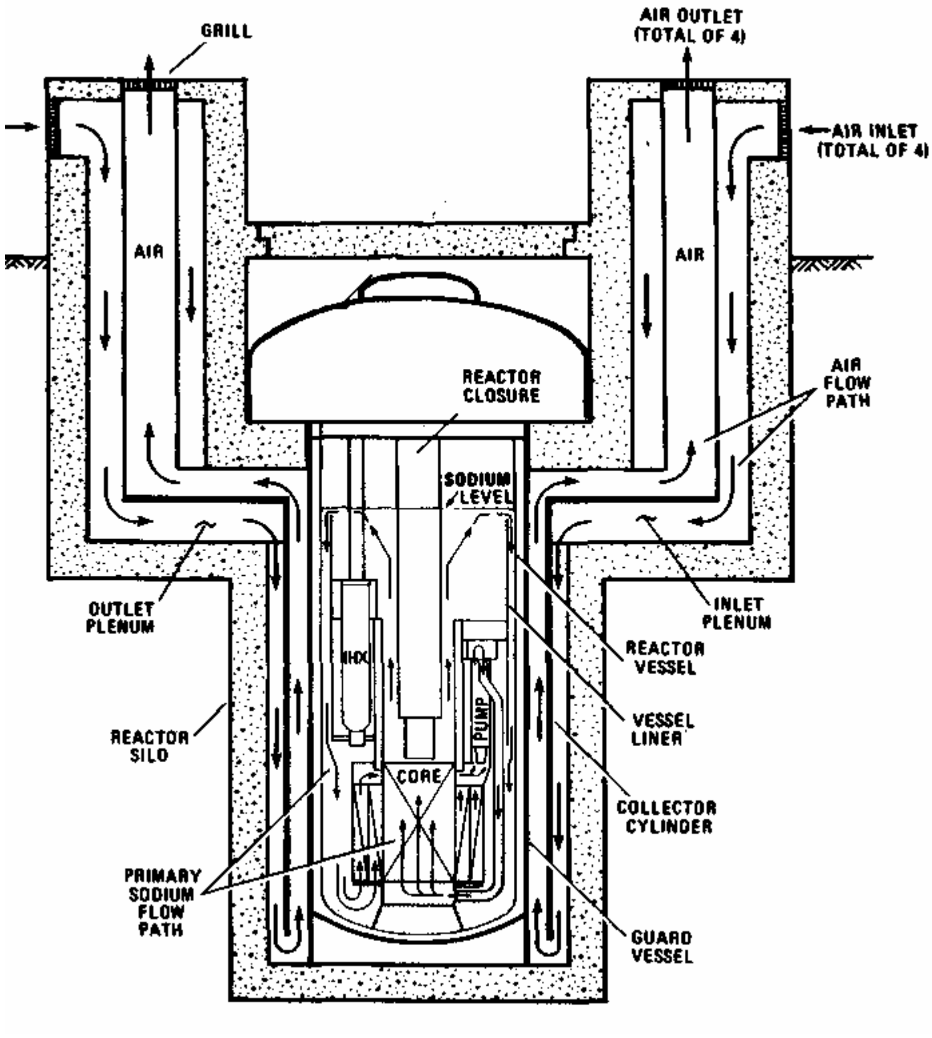
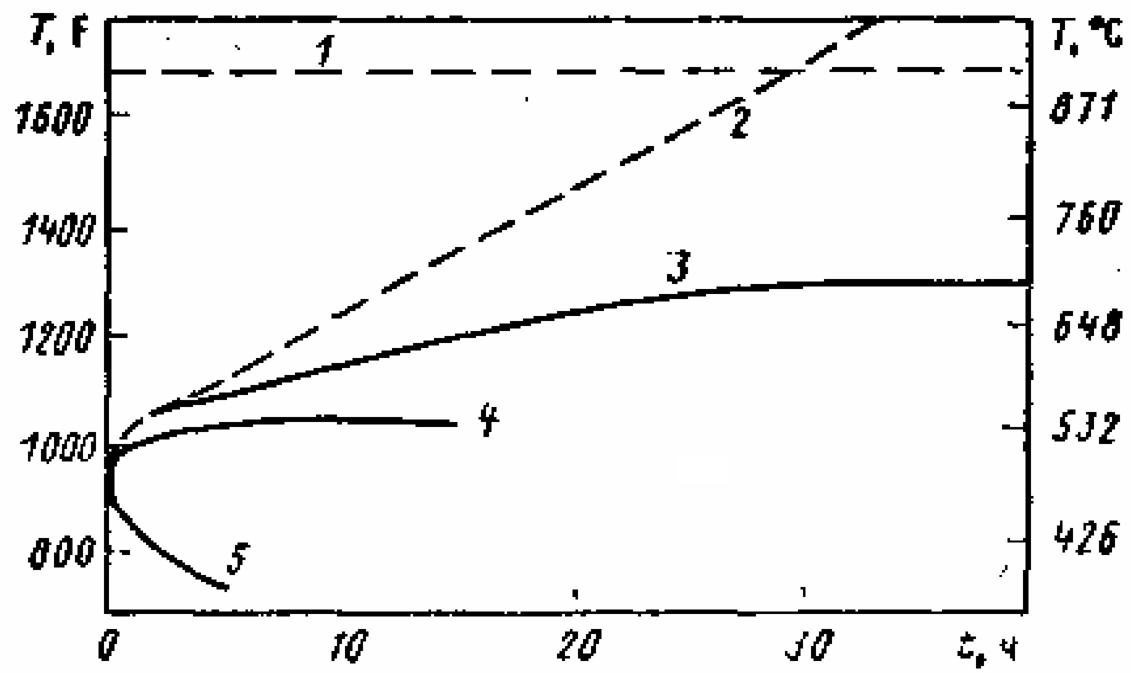


FIG. 4. Primary sodium flow path and passive emergency heat rejection from the vesse1 of PRISM reactor.

Under emergency conditions, when the temperature of the sodium and structural elements are elevated, a rapid increase in the heat transferred into the air take place – proportional to a cubic of the wall temperature value. Calculations have shown that such a system is capable of removing a thermal output of 1% of the rated value. Here, the coolant temperature in the reactor does not exceed 650°C. Thus, the system at hand is entirely passive and is self-regulated. The efficiency of the subject system increases with an increase in the temperature value in place in the reactor. This system is activated without the participation of the operator and can cool the reactor for an extended period of time. Calculation results are presented further on in this report which demonstrate that the safety of the power reactor innovative small module (PRISM) reactor over the course of an accident involving the total failure of the heat removal system without scram is ensured exclusively by an air-cooling system.

Of course, the possibility of employing such a simple emergency shutdown-cooling system - one based on passive principles alone - is a result of the low capacity of the PRISM reactor; i.e. of the existence of beneficial relationship between the surface of the vessel and the thermal capacity of the reactor. This constitutes one of the principal advantages of modular reactors. The aforementioned relationship is diminishes as reactor output increases. In addition to an external heat removal system, the sodium advanced fast reactor (SAFR), having an electrical capacity of 350 MW, also employs a heat removal system of the internal type, the latter taking the form of emergency shutdown-cooling heat exchangers that are built into the vessel and are connected to the external sodium-air exchangers. The thermal processes taking place in the SAFR reactor in the presence of the emergency conditions under discussion are depicted in Fig. 5.



1-boiling temp. of sodium; 2-reactor cooling due to the heat capacity of sodium and steel; 3-cooling by air only through the guard vessel; 4-heat rejection by the emergency heat exchanger and by air cooling of the guard vessel; 5-normal shutdown of the reactor.

FIG. 5. Temperature of the coolant in the upper chamber of SAFR reactor during an accident situation with loss of heat sink without scram.

In this reactor, the external natural air circulation vessel cooling system removes thermal outputs of 1.5 MW under normal conditions and 5 MW under emergency conditions. It should be noted that the internal emergency shutdown-cooling system does not function when the reactor is operating normally. The flow of air through the air heat exchanger is closed off by a damper and the emergency shutdown cooling heat exchanger is filled with helium. The fact that the aforementioned passive systems directly abut on the reactor is important, since the equipment supporting the heat removal loops and the steam-water circuit is thus eliminated from the makeup of the systems that are essential to safety. This reduces the volume of vital and complex equipment and systems, curtails the complement of operating personnel, shortens equipment manufacture time frames, and curbs expenditures for equipment repair. In addition, it becomes possible to enhance safety while simultaneously simplifying and reducing the cost of a nuclear steam supply system. The chief advantages of the utilization of passive features in the reactor protection system design are as follows:

— The assessment of the risk of serious accidents is simplified due to a decrease in both the number of systems that ensure safety and the number of probable operator errors; and
— It becomes possible to experimentally verify reactor safety, which is important both to specialists and to the public, since the latter doesn't often understand the complex probabilistic criteria that characterize NPP safety at all.

7.3. DEMONSTRATION OF SAFETY CHARACTERISTICS WITH EXPERIMENTAL LMFRs

Following the accidents at Three Mile Island and particularly at the Chernobyl NPP, those countries which were engaged in the development of fast reactors began to focus a great deal of attention on reactor protection systems that function in accordance with passive principles without the participation of automatic systems or the operator. Experiments conducted in direct relation to the safety of three fast reactors - the Rapsodie, the EBR-II, and the FFTF - gave great impetus to the evolution of this research. During these experiments, the most serious emergency situations were simulated, as exemplified by the failure of the regular heat removal systems with simultaneous failure of the shutdown system. On the whole, the subject experiments demonstrated the existence of a direct dependence between core power rate and power removal. For example, when the absorber rods of the EBR-II reactor “jammed”, reactor output was correspondingly reduced in proportion to the feed water flow rate decrease that resulted from reactivity feedback effects. During the shutdown of the primary pumps accompanied by the safety rods failure reactor output was reduced to the level of decay heat that could be removed from the core by means of natural circulation. Thus, the marked self-regulation properties of fast reactors were clearly confirmed by these experiments. Reactor experiments aimed at simulating the most serious emergency situations were begun in Europe. Such experiments were first authorized and set up in France.

7.3.1. Rapsodie reactor

7.3.1.1. Main features

The Rapsodie experimental sodium cooled reactor, the first French fast neutron reactor located at the Cadarache Research Centre and was operated by the CEA. The construction was started in 1962 within an association of CEA and EURATOM. The reactor went critical on 28 January 1967, reaching 20 MW(th) power on 17 March 1967. The core and equipment were modified in 1970 to increase the thermal power level to 40 MW (FORTISSIMO) with a peak neutron flux of 3.2×10^{15} ncm^{-2}s^{-1}. In operation, the primary sodium entered the core at 400°C and flowed out at a mean temperature of 550°C. The operating parameters were similar to those in large commercial size reactors.

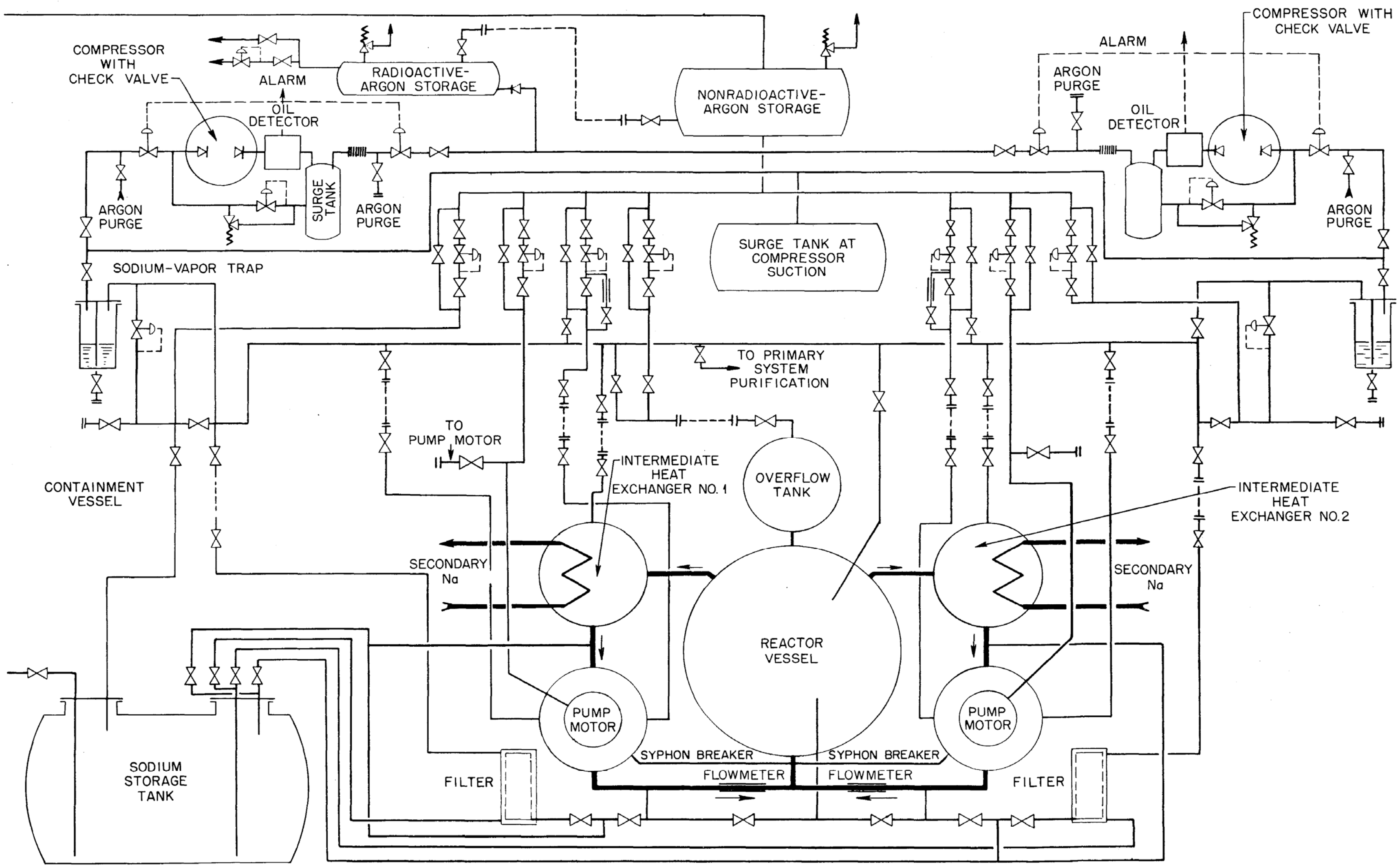


FIG. 6. Rapsodie primary and inert gas system.

The reactor core was cooled by two identical loops each comprising a primary sodium circuit from which thermal power is transferred to a secondary sodium circuit through an intermediate (sodium/sodium) heat exchanger (IHX) by means of a primary pump (Fig. 6). The system lines are enclosed in concrete cells inside a double containment barrier (Fig. 7).

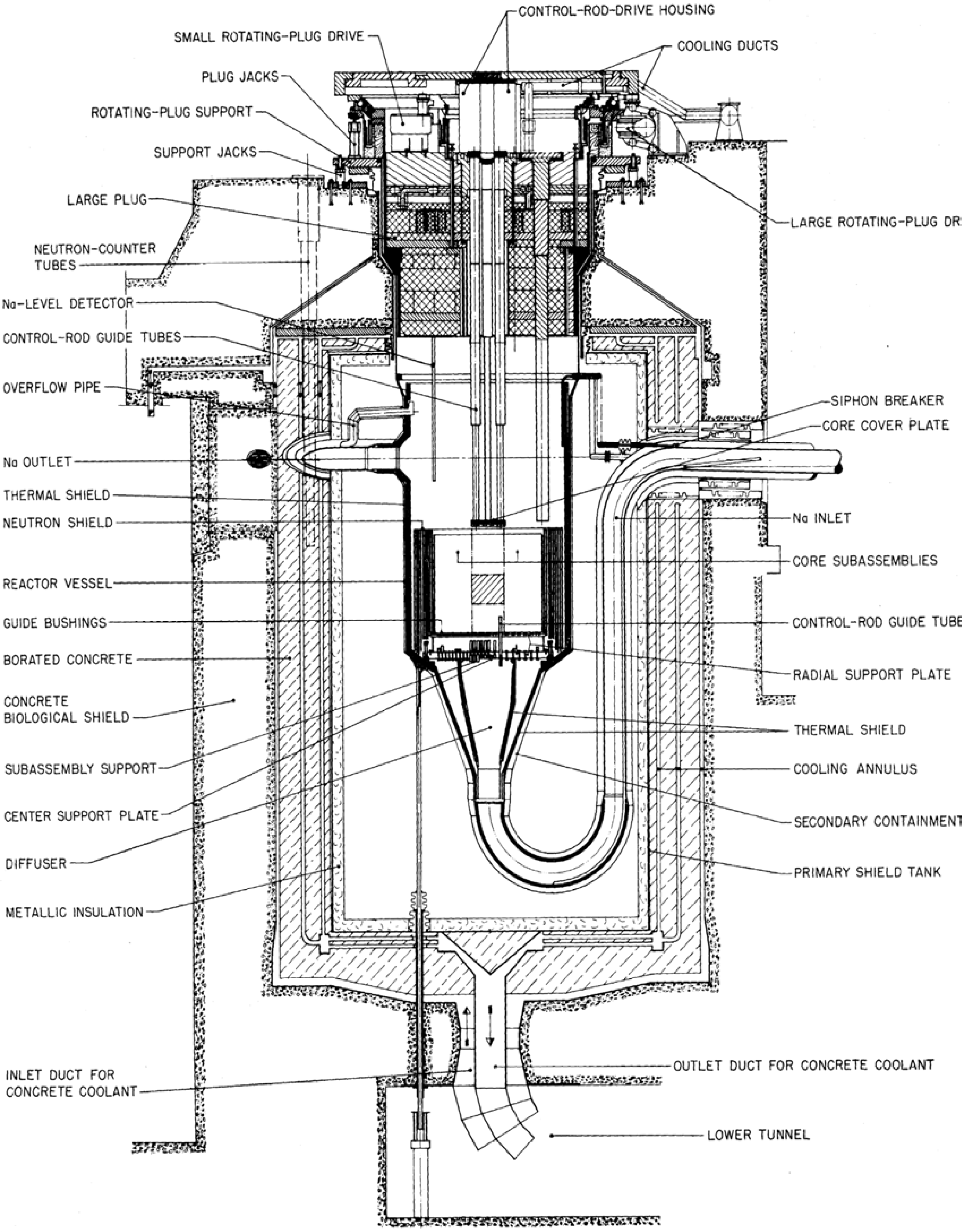


FIG. 7. Rapsodie reactor cross section.

The principal geometric specifications of the primary pipes system are the following:

- — Core to intermediate heat exchanger (IHX): D = 300/314 mm, 16 m long;
- — IHX to pump: D = 300/314 mm, 8.5 m long;
- — Pump to Y junction: D = 200/208 mm, 18 m long;
- — IHX vessel dimensions: D = 884 mm, 5.2 m high;
- — Pump vessel dimensions: D = 850 mm, 4.5 m high;
- — Expansion tank: 36 m^2 surface area.

7.3.1.2. The installation

The installation consists of six main buildings, access to three of them is restricted. These are:

(1) The reactor building or secondary containment including the reactor vessel and its upper closures as well as the two primary loops, each of which was equipped with a mechanical pump and an intermediate heat exchanger. All these components were enclosed in concrete cells to provide radiation shielding. The secondary, non-radioactive sodium, is piped to a conventional building containing the components of the two secondary loops including a sodium/air heat exchanger in each;

(2) The active building comprising interim storage facilities for both fresh and used fuel, and various other facilities such as the washing cell for decontaminating components polluted with primary sodium, and a dismantling hot cell used for conditioning used irradiated equipment for long term storage as waste;

(3) The fuel assembly dismantling building comprising hot cells for non-destructive examination of fuel pins, and the assembly of experimental sub-assemblies.

All the circuits and components are made of austenitic stainless steel, the main pipes and vessels have a double wall. The reactor vessel is immediately surrounded by special high density concrete containing rare earth oxides, called Sercoter. This was protected externally by a steel liner which was considered to constitute the second barrier.

Rapsodie was designed, built and operated to obtain data on the physical behaviour of a fast neutron reactor under static and dynamic conditions, to offer information of direct use for the design of future LMFRs, and to supply a fast neutron flux for irradiation tests of fuels and materials. Mixed oxides were used as reactor fuel. During its 15 years of operation, more than 30 000 fuel pins of the driver core were irradiated, of which about 10 000 reached a burnup beyond 10%, and 300 irradiation experiments and more than 1000 tests were performed. In 1971, the irradiations performed in the core revealed a phenomenon of irradiation swelling in the stainless steel of the wrapper and the fuel cladding in the high neutron flax. The Rapsodie results have been extrapolated in the Phénix reactor. The operating history of Rapsodie has been reviewed in journals and conference papers, and have regularly been published by the TWG-FR. The decision to stop running the reactor was taken after two successive defects were detected in the primary system containment.

The first defect, which appeared in 1978, consisted of a sodium micro leak: radioactive sodium aerosols were found in the double wall reactor vessel. Investigations did not find any liquid sodium in the gap nor locate the defect. The reactor was subsequently operated at a reduced power level (~ 0.6 P_N), which was high enough for irradiation needs but did not cause the leak to reappear. The second defect appeared in 1982 and consisted of a small leak from the nitrogen blanket surrounding the primary system. Before the final shutdown of the reactor, a series of end-of-life tests were conducted in April 1983. The LMFR Rapsodie was shut down in April 1983. Pre-decommissioning operations were then conducted until 1986. They consisted essentially in unloading the fuel and fertile assemblies, and in draining the sodium from the primary and secondary circuits. Decommissioning operations started in 1987.

7.3.1.3. Experiments with the Rapsodie reactor

Two series of tests performed on the Rapsodie reactor, the purpose of which was to investigate the serviceability of this reactor's core and of the reactor as a whole under extreme conditions that were characterized by an exceedingly high temperature. The first series of tests to be performed called for an experimental inquiry into the behavior of fuel elements (FEs) during fuel melting. Over the course of these tests, the fuel pin linear power observed on two test subassemblies reached 1000-1060 W/cm; i.e. two times greater than that normally used in commercial reactors. Subsequent materialogical investigations of FEs with different fuel-cladding gaps (air or helium) revealed that 54% of the fuel present was melted in the first instance, while 29% was melted in the second instance (Figs 8 and 9).

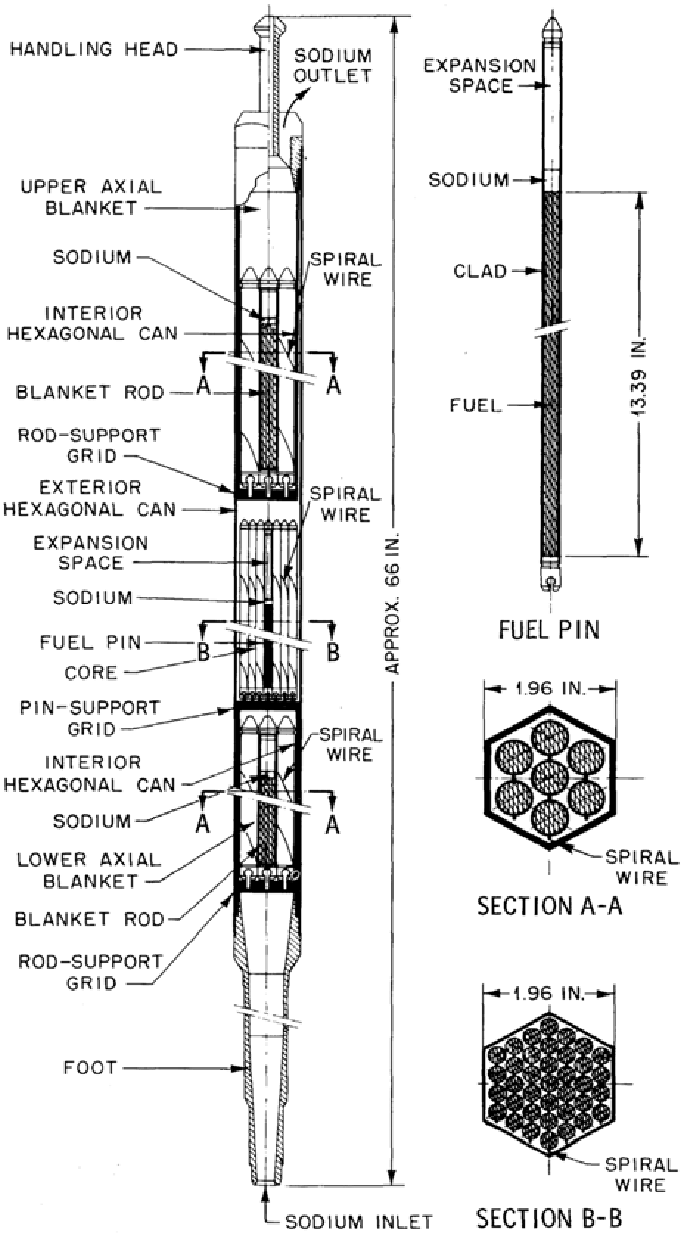
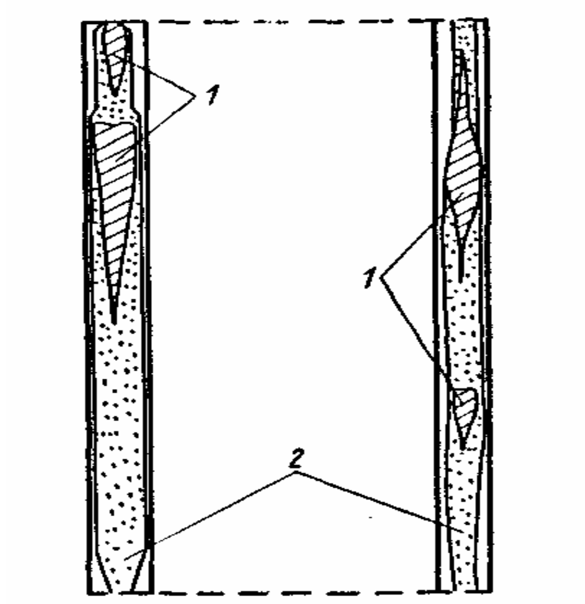


FIG. 8. Rapsodie fuel subassembly and fuel pin.



1-empty space; 2-molten fuel

FIG. 9. Fuel pin containing molten fuel of Rapsodie reactor.

Profilometric and metallographic research demonstrated the absence of deformation and of integrity loss of the FE claddings. These experiments provided the careful monitoring of the temperature status by the thermocouples that were incorporated into the cladding and the fuel. The FEs were held at the aforementioned peak linear power for 10 minutes. The second series of experiments simulated the most serious accident, which consisted of the shutdown of the primary-circuit and secondary-circuit pumps, as well as the tertiary-circuit fans, and the non-operation of the safety rods. Here, reactor output reached 21.2 MW (more than 50% of the rated value), while the mean coolant temperatures at the reactor inlet and outlet came to 402 and 507°C, respectively. The principal characteristics of these tests are depicted in Fig. 10.

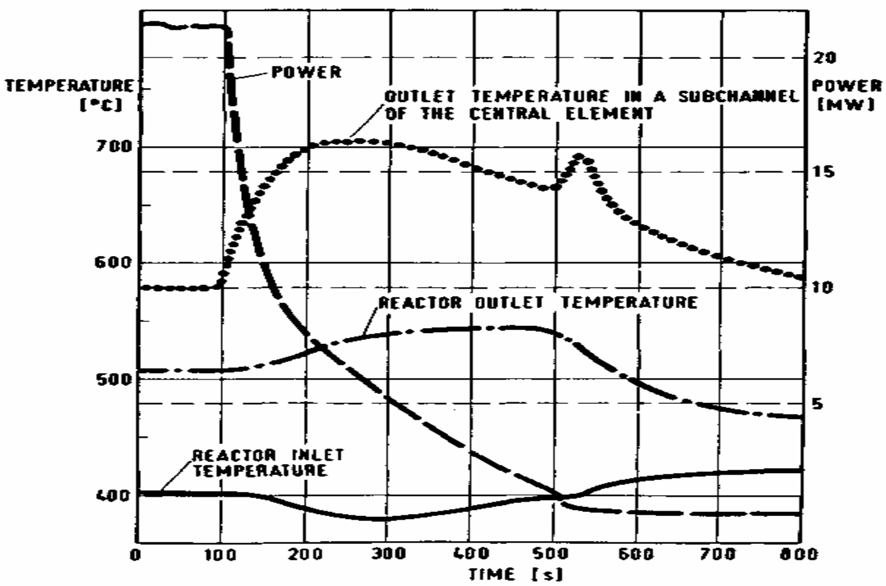


FIG. 10. Rapsodie end-of-life test: characteristics during a LOFWS transient[10].

[10] In the related Rapsodie tests the pump stop time was about 400 s; the halving time τ appears to be about 45 s.

Yet another important fact should be mentioned with respect to the results emanating from the experiments conducted on the Rapsodie reactor. Since the Doppler effect of the Rapsodie core is equal to zero, it was possible to define, with a high degree of precision, the influence exerted by axial fuel expansion. Here, it was essential to understand the manner in which the fuel expanded, independent of the cladding or in conjunction with the cladding. A comparison of calculation results and experimental data demonstrated that the fuel residing in the core shared a state of coalescence with the FE cladding and expanded with the cladding upon heating-up. It is in such instances precisely that good agreement is reached between the calculation results and the experimental data concerning the coolant temperature at the subassembly outlet.

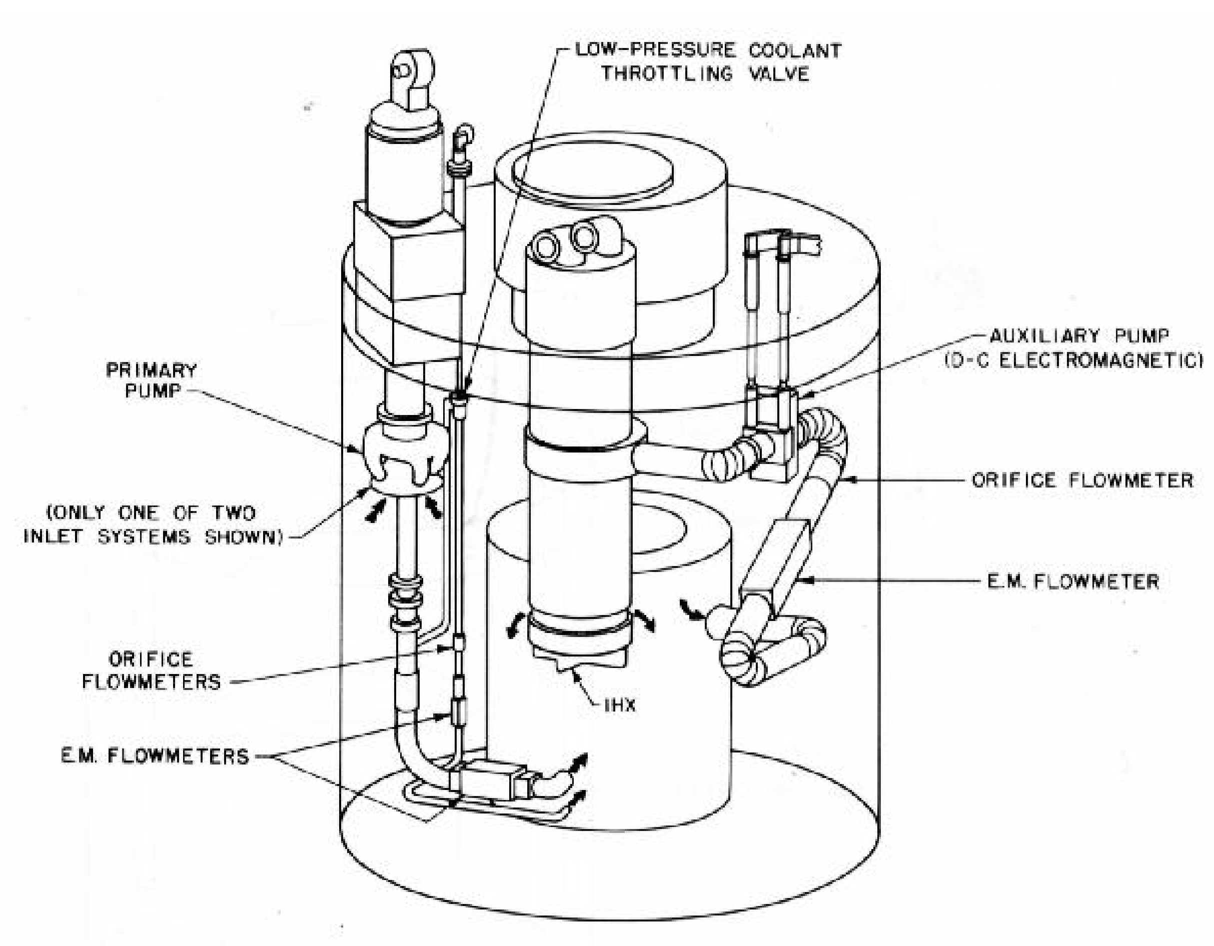


FIG. 11. EBR-II equipment and pipes.

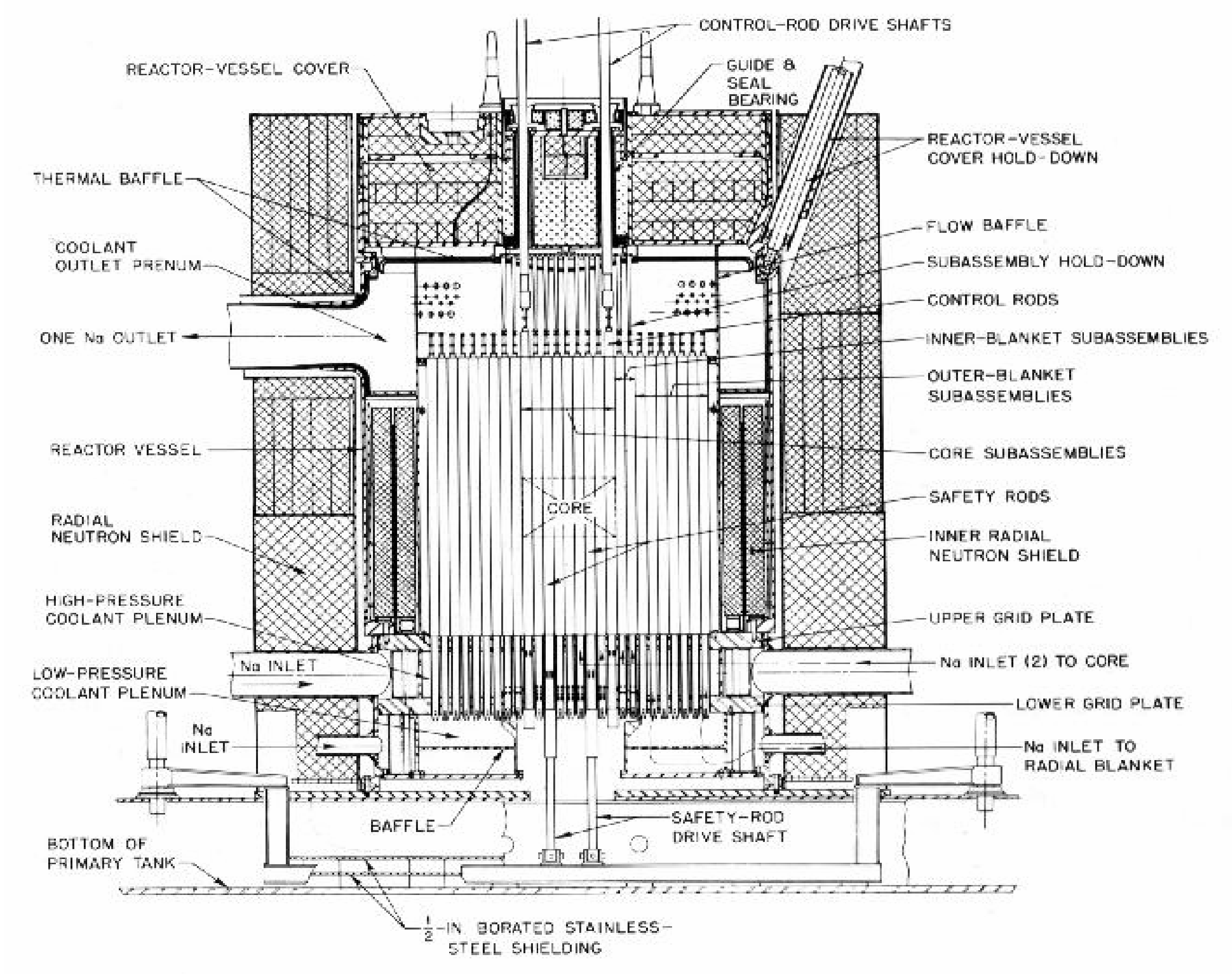


FIG. 12. EBR-II reactor cut off.

The primary sodium contained within this tank represents the primary cooling system for removal of the heat from the reactor core. Liquid sodium, with a boiling point of approximately 900°C, has excellent thermal properties and is thus an optimum coolant. The primary system contains about 330 m^3 of sodium, and transfers heat to the secondary sodium system (about 50 m^3) through a sodium-to-sodium intermediate heat exchanger.

The secondary sodium was circulated in a closed loop through superheaters and steam generators outside of the reactor containment (Fig. 13). The high pressure steam produced in the steam generator drove a turbine generator to produce electricity. EBR-II fuel subassembly and fuel element are shown in Fig. 14.

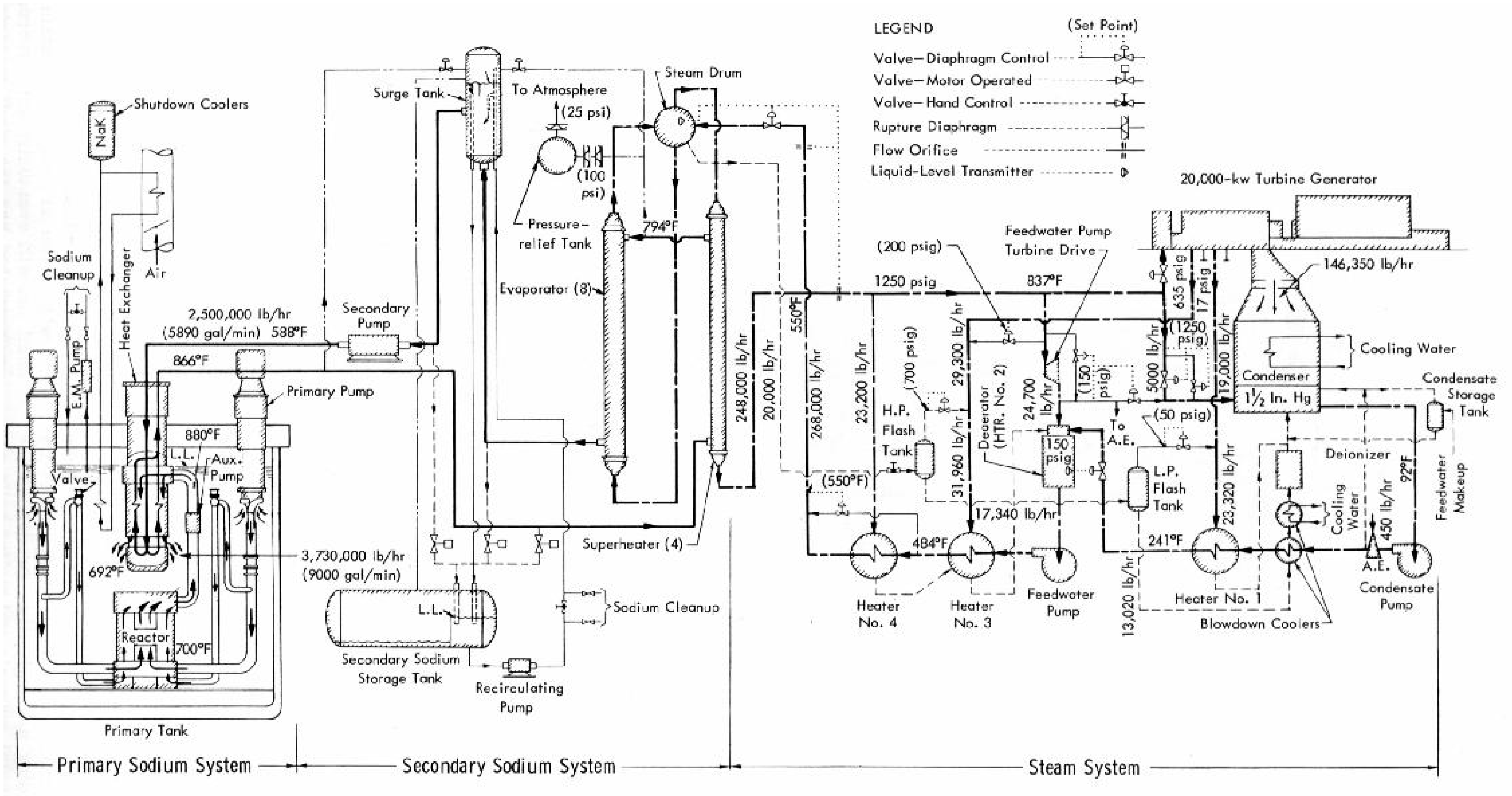


FIG. 13. EBR-II schematic heat transfer and flow diagram.

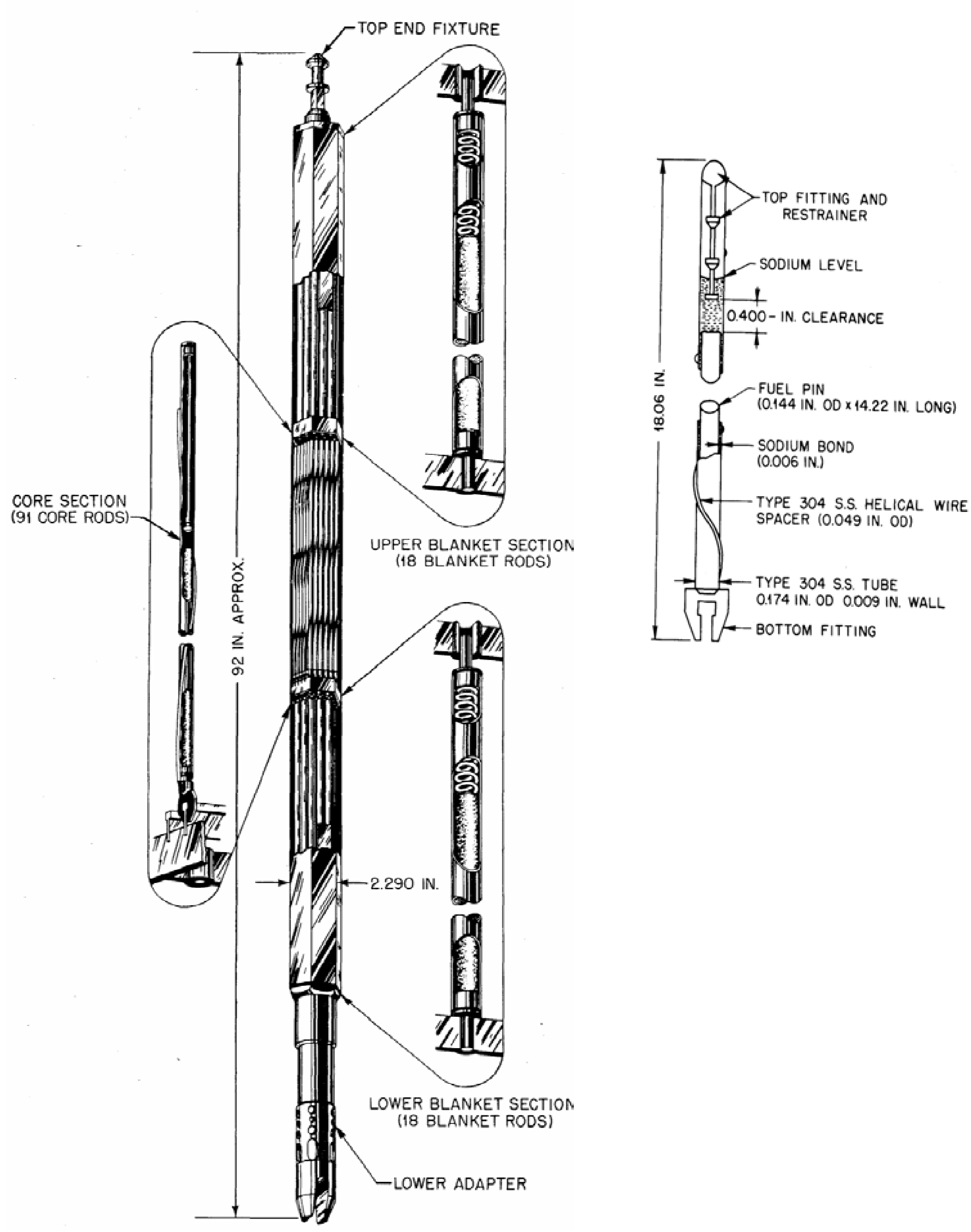


FIG. 14. EBR-II fuel subassembly and fuel element.

7.3.2. EBR-II reactor

7.3.2.1. Design features

The Experimental Breeder Reactor-II (EBR-II) was a sodium cooled metal-fuelled liquid metal research reactor located in the South-Eastern portion of the Idaho National Engineering and Environmental Laboratory (INEEL). The EBR-II was a 62.5 MW(th) reactor that began operation in July 1964, and when fully operational, was capable of producing ~ 19.5 MW(e). Plant capacity factors typically exceeded 70% and approached 80%, even though the reactor supported an extensive testing program while simultaneously producing electricity as a complete LMR power plant. The following key characteristics common to LMRs made reliable, safe operation feasible: low pressure sodium coolant, limited thermal stress, limited corrosion of components, and simplicity of layout in both the primary and secondary sodium systems.

The EBR-II reactor building was connected to the fuel conditioning facility, a large inert atmosphere hot cell facility. The EBR-II reactor building, a cylindrical structure with a hemispherical domed top, had a steel containment shell with an inner diameter of 24.4 m and a height of 42.4 m. The bottom and sides were 2.54 cm thick steel plates and the dome was 1.27 cm thick, lined with a 10.2 cm concrete missile shield. The 1.70 m diameter reactor vessel and its shield were immersed in a sodium pool within the 7.9 m diameter by 7.9 m height primary tank (Figs 11 and 12). The reactor served as a test facility for fuels development, materials irradiation, system and control theory tests, and hardware development. The EBR-II core and blanket subassemblies were contained within the reactor vessel prior to defuelling (Fig. 12).

7.3.2.2. History of operation

The EBR-II had been operated for 30 years. Given the scope of what has been developed and demonstrated over those years, it is arguably the most successful test reactor operation ever. Tests have been carried out on virtually every fast reactor fuel type, and the reactor itself has been extensively characterized. The most dramatic safety tests were conducted on 3 April 1986, when it was demonstrated that an LMR with metallic fuel could safely accommodate anticipated transients: loss of flow or loss of heat-sink without scram.

The EBR-II before closed was operated as the integral fast reactor (IFR) prototype, demonstrating important innovations in safety, plant design, fuel design, and actinide recycle. The ability to passively accommodate anticipated transients without scram has resulted in significant benefits related to simplification of the reactor plant, primarily through less reliance on emergency power and by virtue of not requiring the secondary sodium or steam systems to be safety-grade. These advantages have been quantified in a probabilistic risk assessment (PRA) conducted for the EBR-II that demonstrated considerable safety advantages over other reactor concepts. The uranium-plutonium-zirconium alloy fuel is fundamental to the superior safety and operating characteristics of the reactor. The results of assessments, analyses, and tests, indicated that the reasonable expected technical lifetime estimate for EBR-II was well beyond 30 years and possibly 50 years or more before approaching any aging limits. The plant-life extension program was refocused to build on the original results of the plant engineering and operational assessment.

In January of 1994, the Department of Energy mandated the termination of the Integral Fast Reactor (IFR) Program, effective as of 1 October 1994. To comply with this decision, Argonne National Laboratory-West (ANL-W) prepared a plan providing detailed requirements to place the EBR-II in a radiologically and industrially safe condition, including removal of all irradiated fuel assemblies from the reactor plant, and removal and stabilization of the primary and secondary sodium used to transfer heat within the reactor plant.

The EBR-II's long, successful operating history provided an important source of informationon for the long-term reliability of LMRs. Major programs conducted in the EBR-II included metal fuel irradiation testing and demonstration of the inherently safe response of a metal-fuelled, pool-type LMR. The EBR-II also served as an important test bed for key features of innovative LMR designs, such as flexible pipe joints and materials and improvements in instrument and control systems. Other major tests conducted included those to determine the efforts of running beyond cladding breach, and the response of oxide fuel to operational transients in the US-Japanese joint program.

7.3.2.3. Experiments with the EBR-II reactor

A number of devices were fabricated in order to conduct a series of tests. Among them, instrumented fuel subassemblies were designed, and one is shown in Fig. 15.

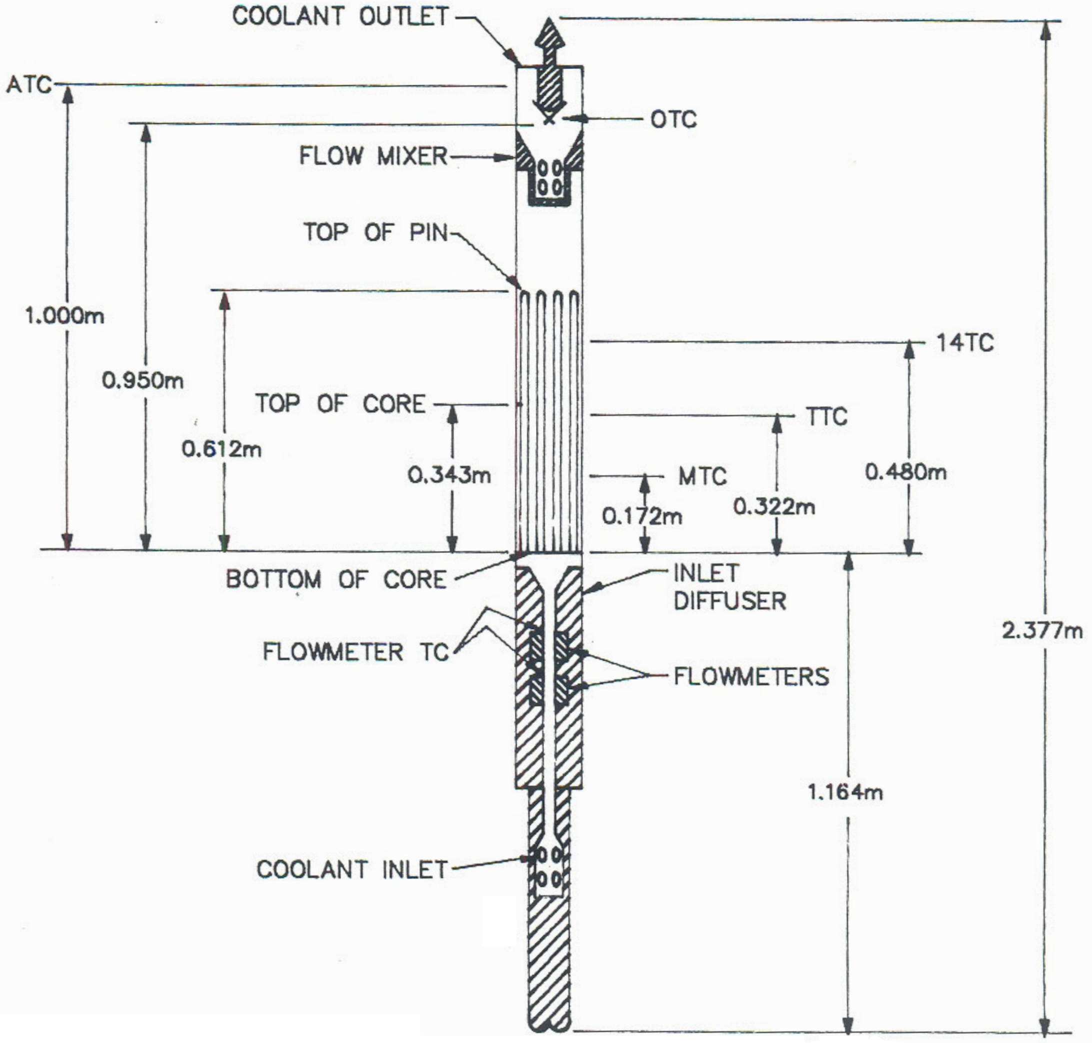
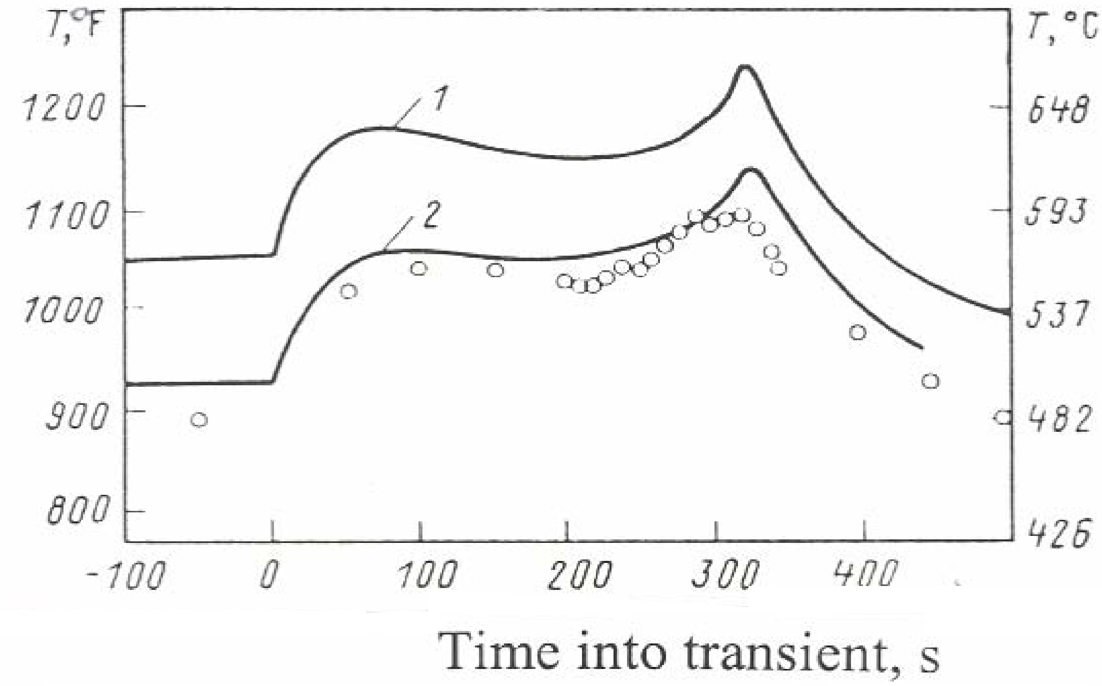


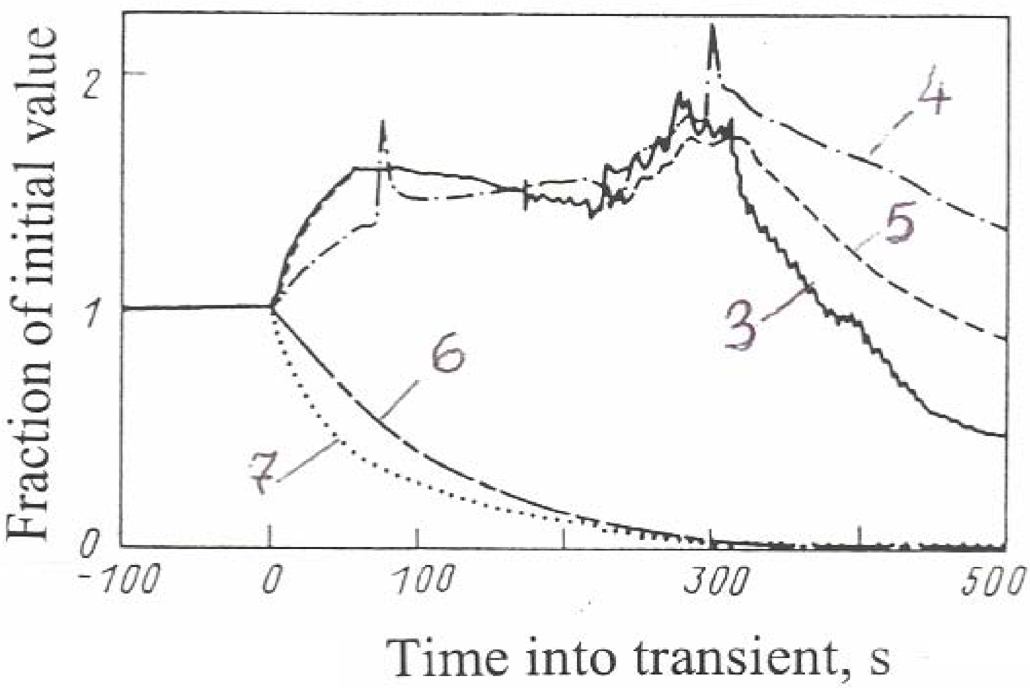
FIG. 15. Sketch of EBR-II instrumented fuel subassembly.

There are two flowmeters located in the lower shield below the active core region. The spacer-wire thermocouples were contained in a metal sheath of 316 stainless steel and were made of type 'K' chromel/alumel with magnesium oxide insulation. There are a total of 28 thermocouples: two located in the flowmeters near the inlet (TC), five at the active core midplane (MTC), thirteen near the top-of core (TTC), several above the core (TC), two at the subassembly outlet (OTC) and two in the thimble region of the subassembly (ATC).

Several series of tests were performed on the EBR-II reactor for the purpose of simulating the most serious of accidents: for example, the loss of the supply of electric power to the main pumps and the safety rod failure at full reactor power, as well as the failure of the secondary and tertiary circuits, together with the simultaneous failure of the safety rods.

The characteristics of this accident are depicted in Fig. 16.

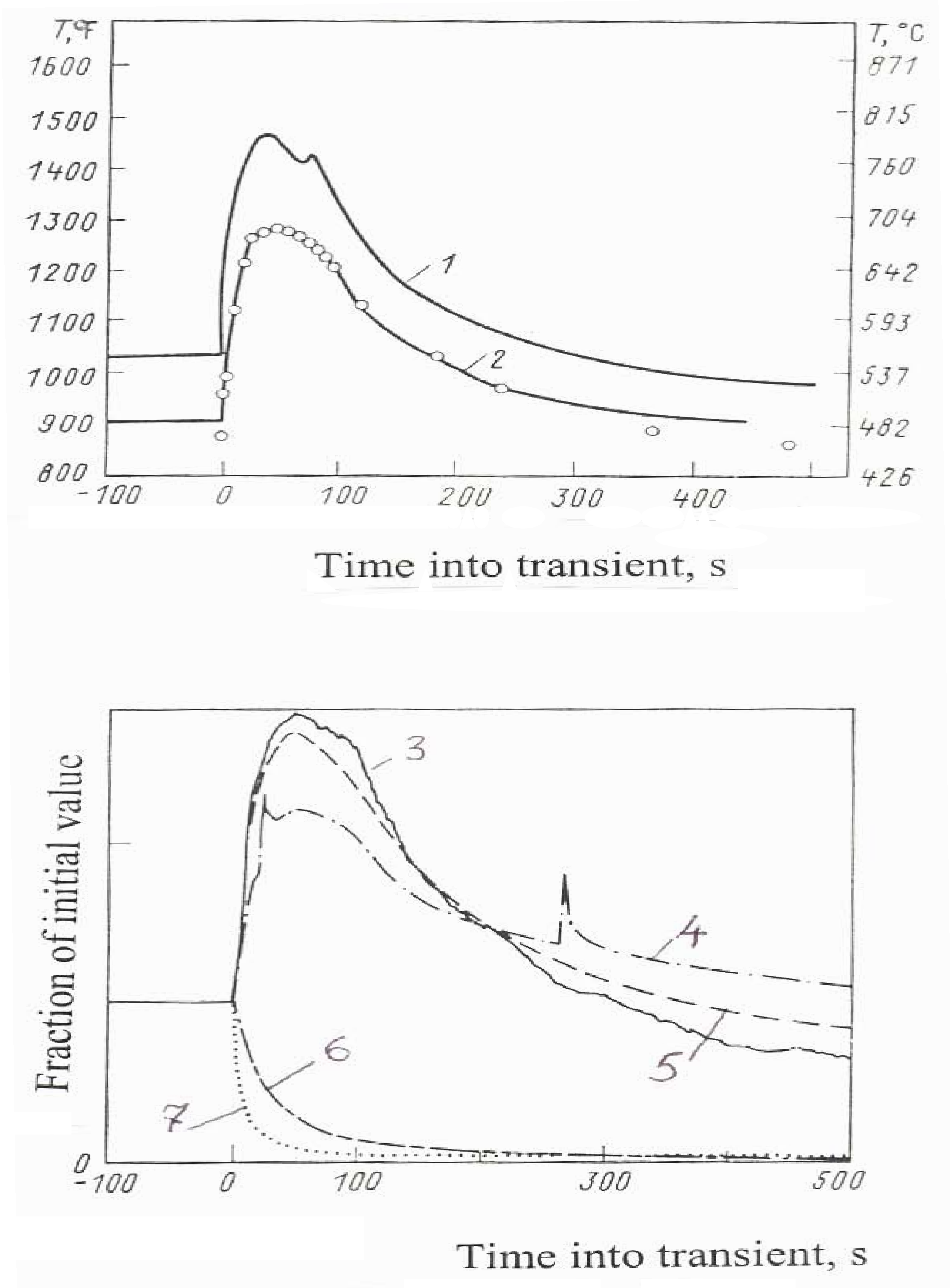




1, 2-the maximum fuel cladding temperature and the coolant outlet temperature, respectively; thermocouple reading; 3-power/flow; 4-reactivity; 5-sodium temperature rise; 6-power; 7-flow rate

FIG. 16. Characteristics of the EBR-II reactor during LOFWS transient.

Experiments simulating the loss of flow without scram (LOFWS) were conducted at full power in the presence of different pump coast-down times: active rundown, by controlling the pump speed, pump stop time 300 seconds (Fig. 16) and 100 seconds, passive coast-down, accompanied by the shutdown of the auxiliary electromagnetic sodium pump (Fig. 17).



1, 2-the maximum fuel cladding temperature and coolant outlet temperature, respectively; thermo-couple reading; 3-power/flow; 4-reactivity; 5-coolant temperature rise; 6-power; 7-flow rate

FIG. 17. Characteristics of the EBR-II reactor during LOFWS transient.

The EBR-II pump system has small inertia, leading to a fast coastdown. Therefore it was decided to use the stored energy in both the pumps and the motor-generator set, and controlling the coastdown with the magnetic clutch which couples the motor and generator.

A comparison of peak temperatures demonstrated the decisive influence exerted by the rundown time of the primary pumps on fuel element cladding and reactor coolant temperature values. Experiments relating to the loss of heat sink without scram (LOHSWS) were also conducted at full power. The reactor outlet temperature was reduced, while the reactor inlet temperature was increased.

Owing to the experimental confirmation of the fact that safety properties are intrinsically inherent in fast reactors, design work has been undertaken, the purpose of which was to ensure the creation of conditions which favour, to the greatest extent possible, the utilization of inherent factors. It is generally known that an increase in the reactor inlet temperature and the reactor coolant temperature rise present therein is accompanied by the thermal expansion of the core diagrid, subassembly bending directed outward from the center, as well as by the elongation of the control rod transfer bars, thereby resulting in the introduction of negative reactivity. In order to reduce power, it is essential that the positive reactivity effects produced by the coolant density decrease associated with an increase in both the coolant temperature and the Doppler effect, should be overcome by the negative effects generated by heating-up and the thermal expansion of the reactor inner structures, since the fuel temperature value is also necessarily reduced.

As previously mentioned, the most impressive results with respect to self-regulation were obtained on the EBR-II reactor when using metal fuel. During experiments involving pump shutdown in circuits I, II, and III together with the safety rod failure, this reactor's output was spontaneously reduced to the decay heat level and the coolant temperature value at the outlet of the hottest core subassembly was increased from 520 to 720°C (a boiling margin of 150°C).

In summary, the EBR-II experiments, which were done in April 1986, simulated both the two major heat imbalance accidents that happened (pumps failure which stopped coolant flow through the reactor - essentially the Chernobyl accident, on the one side, and failure to transfer heat from the reactor coolant to the steam system - essentially the TMI-2 accident[11], on the other side). In neither case was any operator or safety system action taken, and in neither case was the reactor or its fuel harmed in any way.

7.3.3. The FFTF reactor

7.3.3.1. Design features

The concept of a Fast Flux Test Facility (FFTF) dates back to the late 1950s. In April 1965, the USAEC authorized Battelle Pacific Northwest Laboratories to perform a conceptual design and cost study of FFTF. The objective was to establish a unique and flexible reactor plant capable of intense fast flux irradiation of fuels and structural materials in view of the US LMFBR development programme. The components, fuels, materials, and core environment of FFTF were to be as similar as practicable to those required for larger power stations. The FFTF was a 400 MW(th) sodium-cooled fast reactor specifically designed for development and testing of fast breeder reactor fuels, materials, and components (Fig. 18).

[11] Not included in this Report.

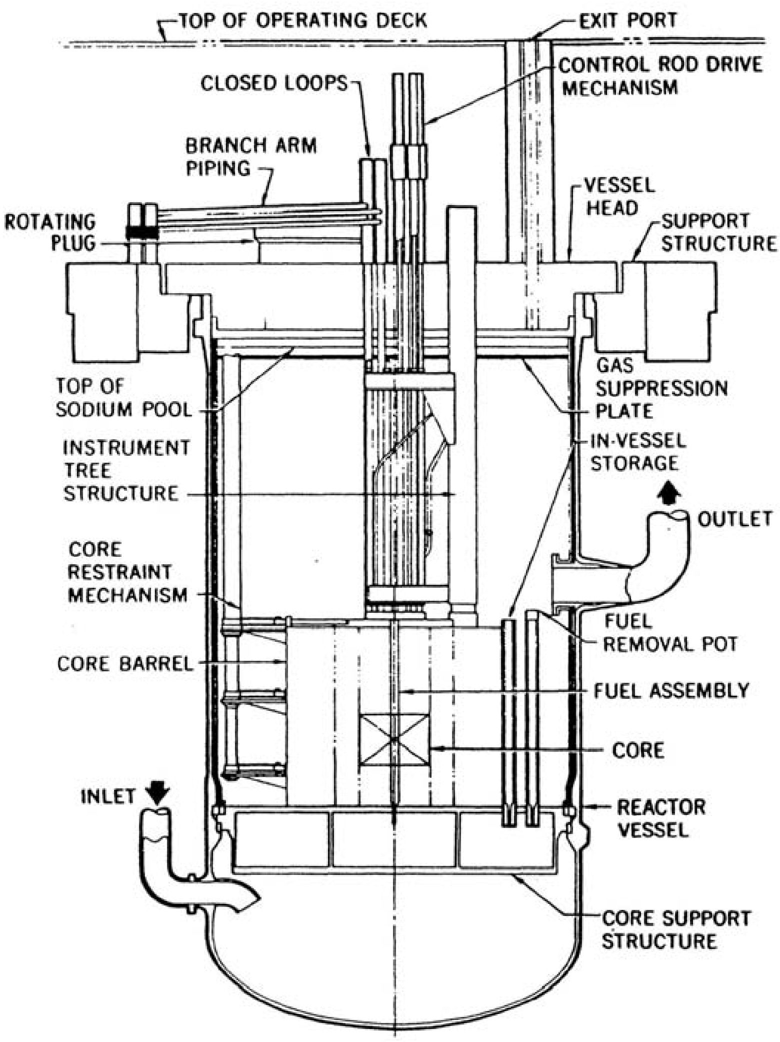


FIG. 18. FFTF reactor elevation.

The reactor was a loop-type plant with three parallel heat transport system loops. The plant has neither steam generators nor blanket assemblies for fissile breeding, consistent with its role as a test reactor. The outer three rows of core assemblies were stainless steel radial reflector assemblies which serve to enhance the neutron flux in the core interior.

The FFTF was equipped with a great deal of instrumentation. Each core assembly was provided with instruments for measurement of sodium flow rates and sodium outlet temperature. Three instrument trees, one of which serves each of the three core sectors, provide outlet instrumentation for all fuel assemblies, control and safety assemblies, and selected reflector assemblies. In addition, 8 of the 73 core positions were equipped for full in-core instrumentation. Two of these eight positions were available for closed-loop facilities.

In these closed test-loops components inserted in the reactor core, the coolant, instrumentation and heat transfer systems were completely separated from the main FFTF core, permitting the testing of fuels and materials over a wide range of temperatures in a controlled environment independent of the main reactor coolant system. The open loop test positions and integral components of the reactor core for testing large quantities of candidate fuel pins and assemblies were cooled by the reactor primary coolant system.

7.3.3.2. Operating experience

The operating histogram of FFTF is given in Fig. 19.

The FFTF began its power ascent in November 1980. In December 1980 full power of 400 MW(th) was reached. A series of natural circulation tests proved that the FFTF loop-type system could be operated safely under conditions of long-term decay heat removal by natural convection without any sodium pumps working.

The FFTF has completed basically “flawless” operation over 10 years confirming the design assumptions and material performance of mixed oxide fuel, sodium reactor systems, and the overall safety and robustness of the modern LMR.

Advanced core materials have also been developed and tested at the FFTF. The most notable material among them has been the ferritic steel alloy HT9, which has been irradiated to ultra-high neutron fluence levels with little or no neutron-induced swelling and has been selected for use in the U.S. ALMR. The FFTF also conducted materials experiments for the development of the fusion reactor. A special test zone in the FFTF core allowed continuous monitoring of tritium production occurring in fusion reactor blanket materials under a variety of irradiation and thermal conditions.

The FFTF completed Cycle 12 in March 1992, accumulating a total of 2278 effective full power days (EFPD) since the beginning of operation. During this time only one fuel pin from all of the standard driver fuel assemblies had developed a leak; this assembly was well beyond its design exposure. Irradiation of the reconstituted fusion materials open test assembly experiment was initiated in May 1991 and functioned as designed until reactor shutdown in March 1992. It has been removed from the reactor for irradiation in other facilities.

Nine core demonstration experiment for fuel assemblies, including lead tests, continued irradiation until the reactor was shut down in March 1992. A lead test assembly reached a world's best fuel assembly burnup of 238 MWd/kg. The highest burnup assembly reached a burnup of 221 MWd/kg. It was agreed to process a hot channel lead test for post irradiation examination by PNC. All nine assemblies have achieved their current exposures without operational difficulty.

The possibility of future DOE missions along with collaborative international programs for the FFTF had been evaluated. The plant was in steady state hot standby conditions for a long time and was finally shutdown in 2000.

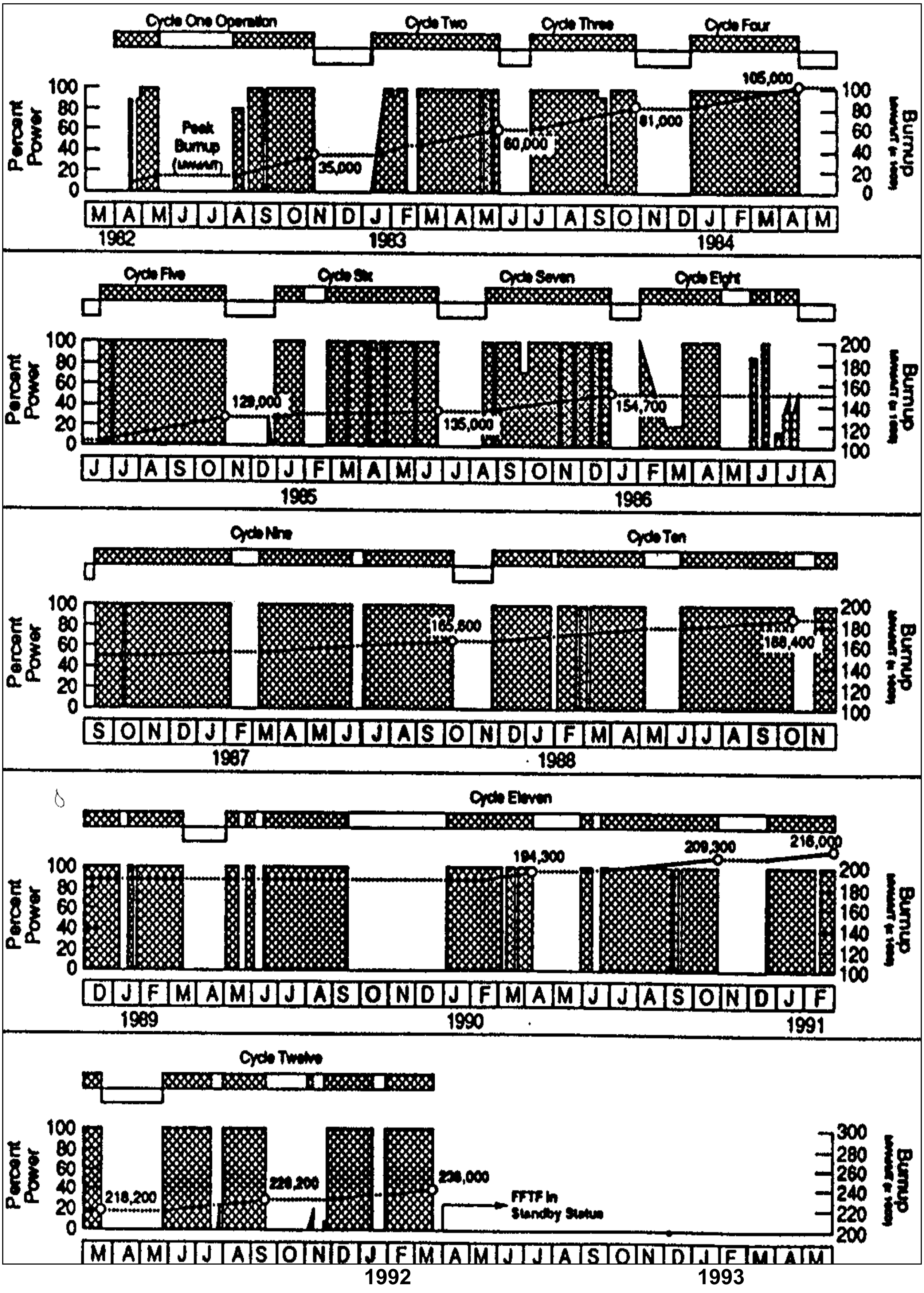


FIG. 19. FFTF operating histogram.

7.3.3.3. The FFTF reactor experiments

The experiments conducted with respect to this reactor were carried out at 50% power over a fairly low range of temperatures (the main pumps functioned at a 100% flow rate prior to shutdown). The results emanating from these investigations are presented in Fig. 20. Experiments involving pump shutdown without scram demonstrated that reactors based on oxide fuels also possess intrinsically inherent safety properties (self-controllability); however, in such reactors, these properties become readily apparent at higher temperatures, sometimes exceeding the permissible value.

Supplemental measures were considered for the passive introduction of negative reactivity in medium power and high power reactors that use oxide fuels. During the aforementioned FFTF reactor experiments, such measures included so called gas expansion modules, GEMs (hollow pipes, sealed at the top, open at the bottom, and set into the internal row of the radial blanket).

When the pumps were in operation, the coolant was pumped into the aforementioned GEMs and gas compression occurred. Following pump shutdown, the core negative reactivity was introduced simultaneously with the displacement of the sodium. This effect on the part of the FFTF is essential to the subject reactor's self-shutdown.

The experiments conducted in the United States — especially those performed on reactors that use metal fuels — are considered to be both outstanding and promising. Based on the results emanating from these experiments, the largest U.S. reactor manufacturers, with the support of government agencies (US DOE) have launched efforts aimed at the development of absolutely safe reactors, in which both the shutdown system and the heat removal system function exclusively on the basis of passive principles; i.e. their activation does not require the participation of automatic systems or an operator.

In the most serious situations, such as those exemplified by LOFWS[12], the spontaneous reduction of power to a few percentage points of the rated value was essential in facilitating the achievement of emergency heat removal from the reactor based entirely on passive principles. As previously mentioned, the simplest of the subject techniques is the cooling of the reactor vessel (guard vessel) by natural air flow. Here, the positive properties of reactors that use a liquid-metal coolant also come into play. Owing to the high boiling point of the coolant (~ 850°C), as well as to the fact that the structural materials used are stainless steels, the temperature of the structural elements (the reactor vessel, the guard vessel, etc.) can be increased to 600–650°C when low-probability one-time emergency situations arise. Relatively low vessel wall thickness values (30–50 mm), together with the great difference between the temperature of the surface being cooled and that of the free air, ensure efficient heat removal, even when the air present in the annular space around the non-insulated guard vessel in the reactor well (with respect to small reactors) or in the special sodium-air exchanger (with respect to medium and large size reactors) is circulated naturally.

[12] The FFTF primary pump was found in acceptance testing to stop in 100 to 115 s and have a halving time τ of about 6 s.

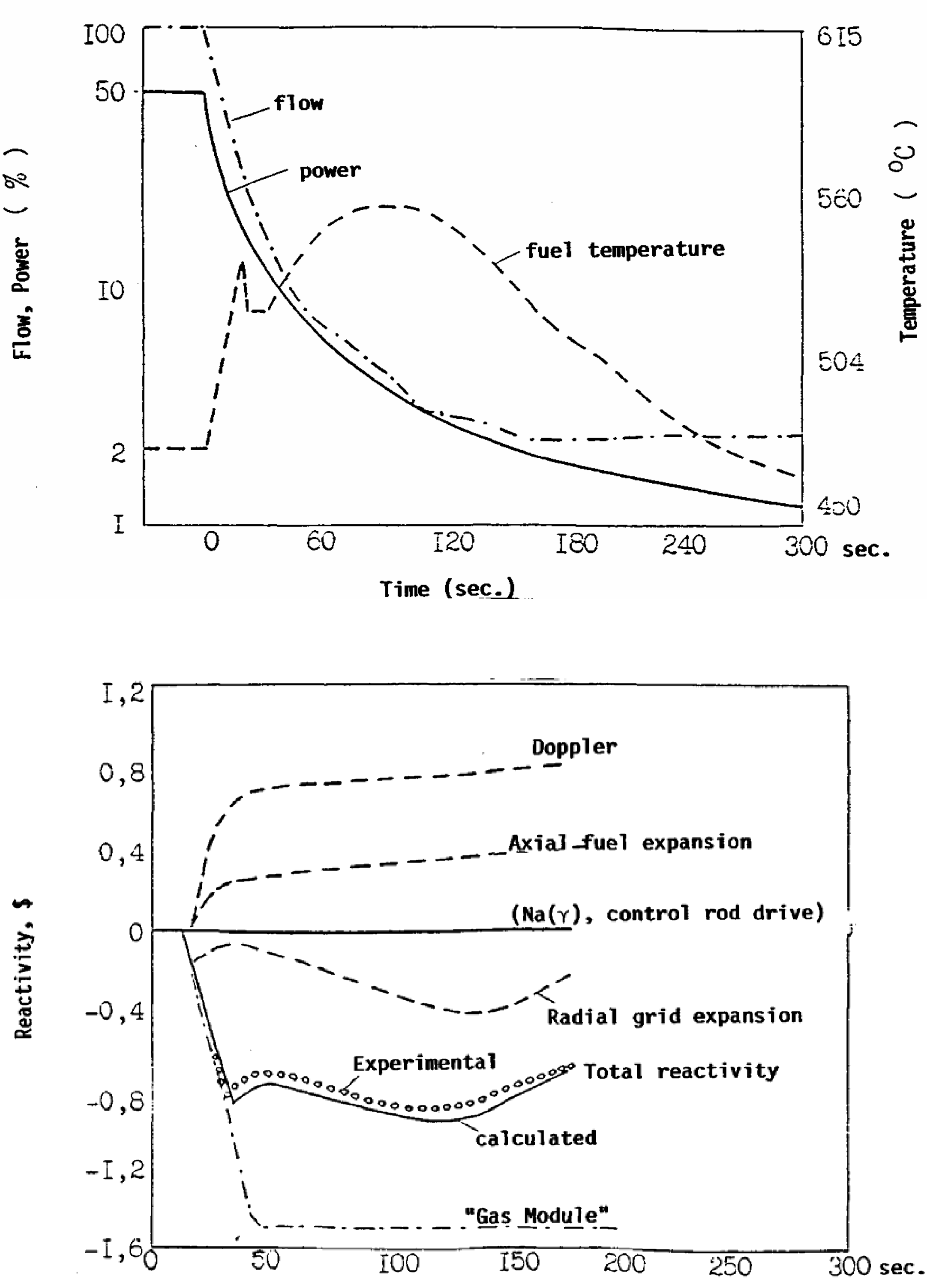


FIG. 20. Characteristics of FFTF during LOFWS.

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8. DOUNREAY FAST REACTOR: DESIGN FEATURES, EXPERIMENTS DURING THE FINAL STAGE OF OPERATION

8.1. DESIGN FEATURES AND MAIN OPERATION RESULTS

As an initial experimental stage of a programme that began in the early 1950s to ensure long-term security of the nuclear component of the UK's electricity supplies, a decision was made in 1954 to build the 60 MW(th), 15 MW(e) Dounreay Fast Reactor (DFR), which was subsequently operated from 1959 to 1977. The main design features of DFR were as follows (Figs 1-3):

- Upwards coolant flow through the core to avoid gas entrainment;
- The driver core with a vented metallic enriched uranium fuel elements in the form of hollow cylinders;
- An electromagnetic pumps in primary and secondary circuit;
- A low-rated, double-walled matrix design steam generators.

The philosophy of DFR at that time was to have the experimental part of the system only inside the reactor vessel, and in the outside zone every effort was to be made to minimize the risk of breakdown of the cooling system. This explains the unusual feature of 24 coolant loops, which results in a size of pumps and heat exchanger where experience had been accumulated in previous experimental work.

The DFR was designed primarily to confirm the feasibility of the fast reactor concept, but quickly assumed a more enduring role as a test bed for candidate fuel, clad and structural materials. After several years of successful operation, the DFR was shut down in 1967/68 for one year to locate and repair a small leak in one of the coolant outlet pipes inside the reactor vessel. Perniciously, the leak disappeared every time the reactor was shut down, making it very difficult to locate and assess. The DFR continued to operate until March 1977, when it was finally shut down. At its closure, mixed oxide fuel experiments had reached a peak burnup of over 20%. Fuel pins with leaking cladding were irradiated following failure to a further 3% burnup with little deterioration.

Until 1967 the major problem of damage to cladding materials was embrittlement. However, in 1967, evidence was firstly announced of considerable void swelling taking place in austenitic stainless steels irradiated to high fluences in the DFR.

During the final stages of normal power operation of the DFR, a series of experiments was performed with the objective of exposing bundles of typical mixed oxide fuel pins to coolant boiling for prolonged periods. The series, known as the DFR special experiments programme, were comprised of eight separate experiments; they utilized both unirradiated and previously irradiated fuel pins; and, in three experiments, included a thin steel plate simulating a local blockage in the heated section.

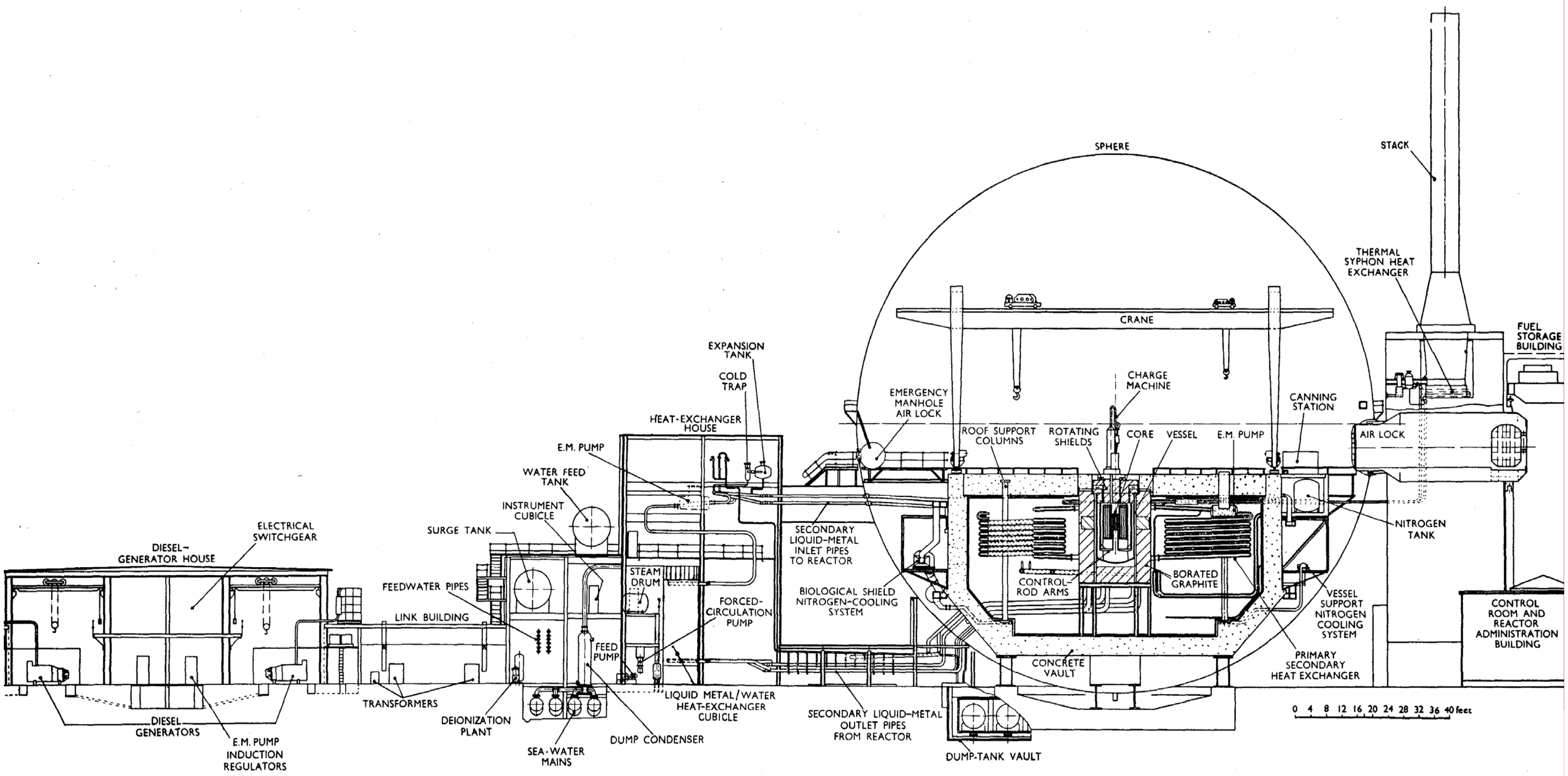

FIG. 1. DFR plant cross section.

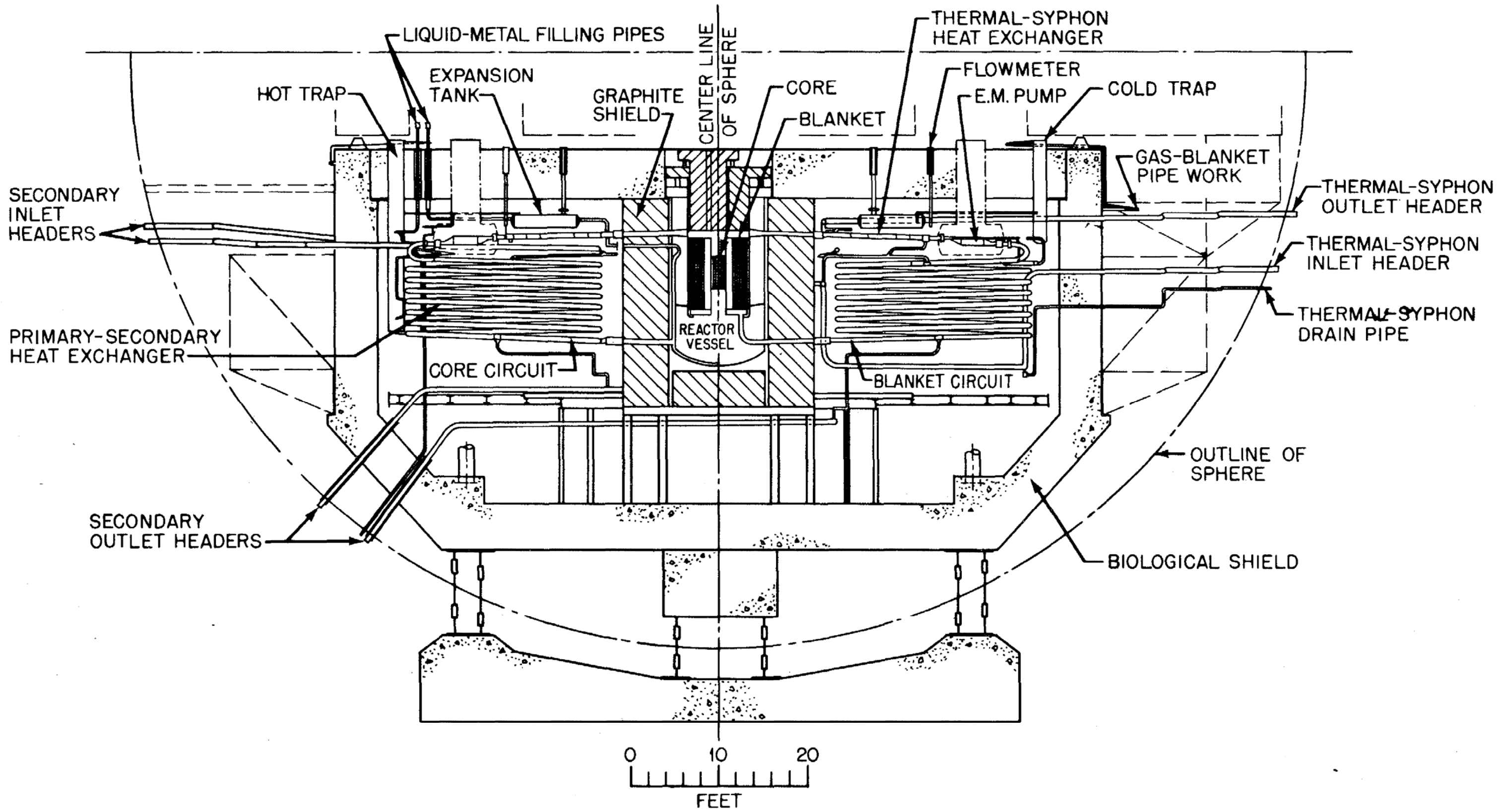

FIG. 2. DFR reactor primary circuit.

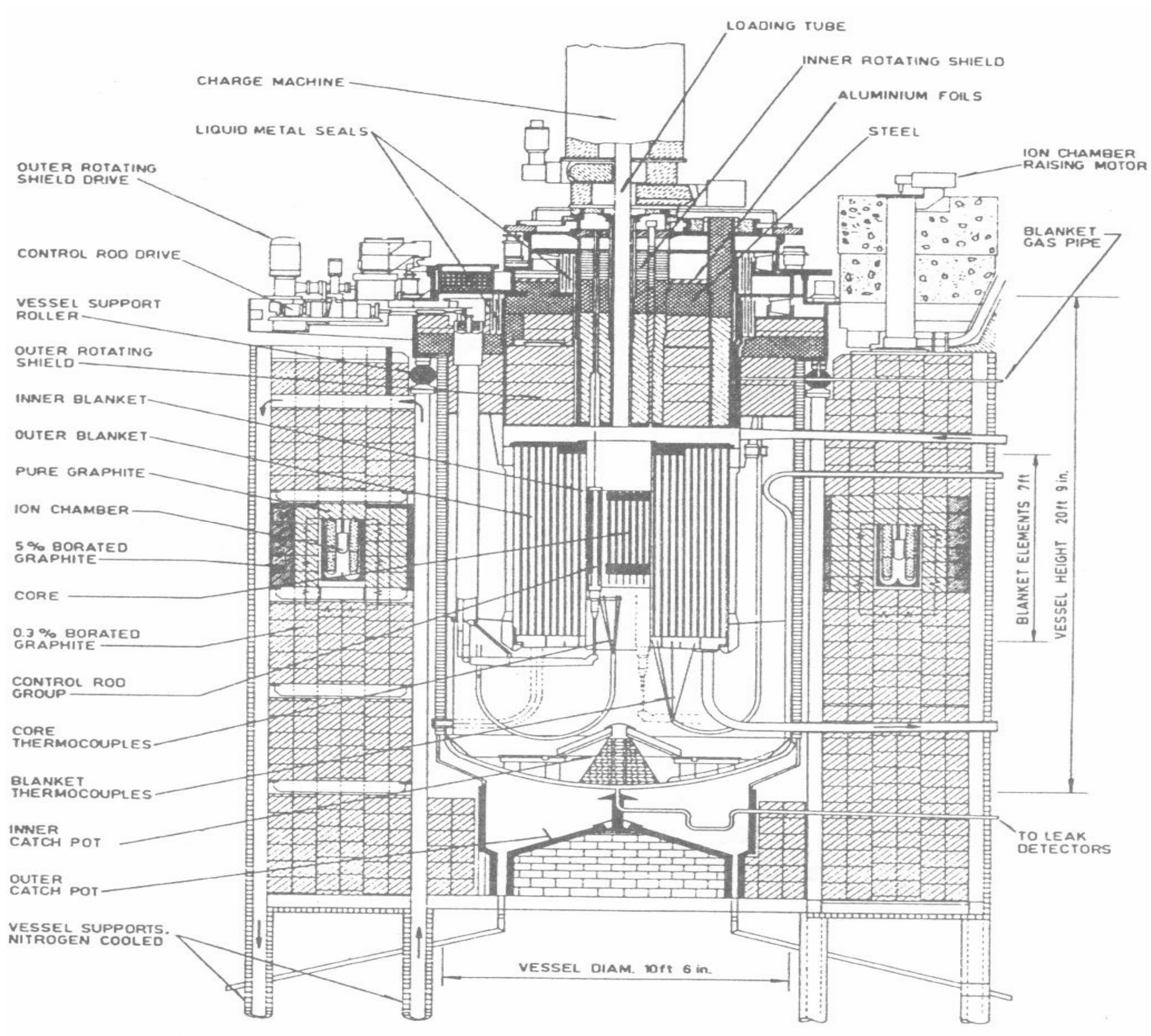

FIG. 3. DFR reactor cross section.

8.2. IN–CORE COOLANT BOILING EXPERIMENTS [1–4]

8.2.1. Introduction

This section comprises the results of DFR in-core coolant boiling experiments, including a brief description of the different rigs. It summarizes the operating experience and conclusions from post irradiation examination (PIE) for each rig and includes conclusions concerning the relevance of the programme and its results to ongoing LMFR safety studies. The work was involved a large number of people at DNE and other centers, e.g. the intrinsic thermocouple was supplied by the Nuclear Research Centre, Karlsruhe, Germany.

In each experiment, coolant boiling was initiated by reducing the coolant flow through the rig using an inlet flow valve built into the experimental vehicle and controlled from the reactor control room. The coolant flow rate was reduced to a level which was 25% below that needed to induce boiling. Each experiment contained an array of thermal and acoustic sensors, the precise complement of devices differing from rig to rig. The installed instrumentation served two purposes; firstly it provided a means of monitoring conditions within the experiment; secondly it enabled a comparison to be made between the efficacy of different types of technique for detecting the initiation and development of a subassembly incident.

In particular, the complementary and contrasting merits of the thermal and acoustic noise detection techniques were assessed at the Risley Nuclear Laboratories (RNL). Interpretation of the experiments was carried out using the subchannel boiling code CLAYMORE which was specially developed for the DFR programme. As a further aid to interpretation, a replica of one of the rigs was studied, using electrical heaters, in a water rig at BNL. The range of tests carried out included both single and two-phase behaviour, the influence of flow direction, effect of different blockage porosities, and the consequences of gas injection. The experiments were not designed to study dry-out mechanisms, so the investigation did not attribute the encouraging behaviour to these mechanisms.

The experiments produced minimal interference with normal reactor operation; they were loaded as part of the normal reactor inventory and, with one exception, not unloaded until the scheduled end of the run, some 65 days later; therefore in most instances the fuel was subjected to significant periods of irradiation following exposure to boiling conditions. PIE of the fuel pins provided insight into how fuel might behave in incident situations analogous to those modeled in this programme. Despite exposure to severe boiling conditions, the majority of the fuel pins examined to date showed either no damage or minor damage such as clad splitting; in one blockage experiment where conditions were severe enough to lead to clad melting there was no evidence of fuel melting.

8.2.2. Objectives of the experiment

In broad terms the experimental objectives were to investigate the consequences of local and inlet blockages in fuel pin bundles under reactor conditions, hopefully to demonstrate the pessimism inherent in safety analyses. More specifically, the results were expected to contribute to a more realistic understanding of the following aspects of LMFBR safety arguments:

- Fuel failure characteristics and propagation under abnormal service conditions;
- The conditions for the initiation and stabilization of local boiling;
- The possibility of blockage growth through instability of the pin cluster under conditions of high temperature and high temperature gradient;

— Heat removal under bulk boiling conditions in a multipin geometry;
— The effectiveness of different instruments in detecting and distinguishing specific fault conditions;
— The timescales available for detection and for remedial action.

In addition, it may be possible to make qualitative comments upon:

(i) Fuel redistribution under escalating local incident conditions;
(ii) The dynamics and extent of fuel-coolant chemical reactions.

The special experiments programme involved eight separate in-pile experiments using a total of ninety-nine fuel pins. The coolant flow conditions were different in magnitude and direction from those in current LMFRs such as the prototype fast reactor (PFR) but it is believed that these atypical conditions make the DFR experiments pessimistic and that the conditions experienced by the fuel were more severe than for the corresponding stages in a similar incident in a commercial LMFR. These experiments explored fresh ground by reducing the areas of speculation and by providing quantitative data, both supported the development of theoretical methods used in safety analysis and provided pointers to the most appropriate areas for further development.

8.2.3. Design of the instrumented vehicles

The experiments were mounted in two different designs of instrumented vehicle; five experiments used a vehicle called a mini-subassembly containing 18 fuel pins and an unheated instrument probe; three experiments used a smaller vehicle involving only three fuel pins and called a trefoil. The coolant flow through the eight experiments was determined by a valve sited upstream of the fuel bundle in each vehicle and operated remotely from the reactor control room. A typical mini-subassembly is shown schematically in Fig. 4.

It contained 18 fuel pins within a pair of concentric cluster walls sitting inside a massive steel carrier. The cluster walls confined an argon gas layer which insulates the coolant flow in the rig from the comparatively cold bulk reactor flow. The five mini-subassemblies are divisible into two categories, the 523 series for bulk boiling and the 539 series for local blockage studies.

The fuel was mixed oxide, either pelleted or vibrocompacted, with burnups ranging from zero to 10%, and clad in cold worked M316 stainless steel. The nominal pin outside diameter (OD) was 5.3 mm and the pitch spacing similar to the PFR; the pins were spaced by honeycomb grids axially separated by approximately 100 mm. The fuelled length was comparatively short, being 505 mm in the 523 series and between 286 and 375 mm, respectively, in the 539 experiments. Because of the low inlet temperature of 230°C, and short heated length, the axial temperature gradient under boiling conditions was between three and four times the gradient for a comparable incident in current LMFR designs. Typical peak linear rating for the fuel pins was 32 kW/m (280 W/g).

The local blockage in the 539 series was a thin steel plate mounted eccentrically and covering approximately 70% of the available flow area (Fig. 5).

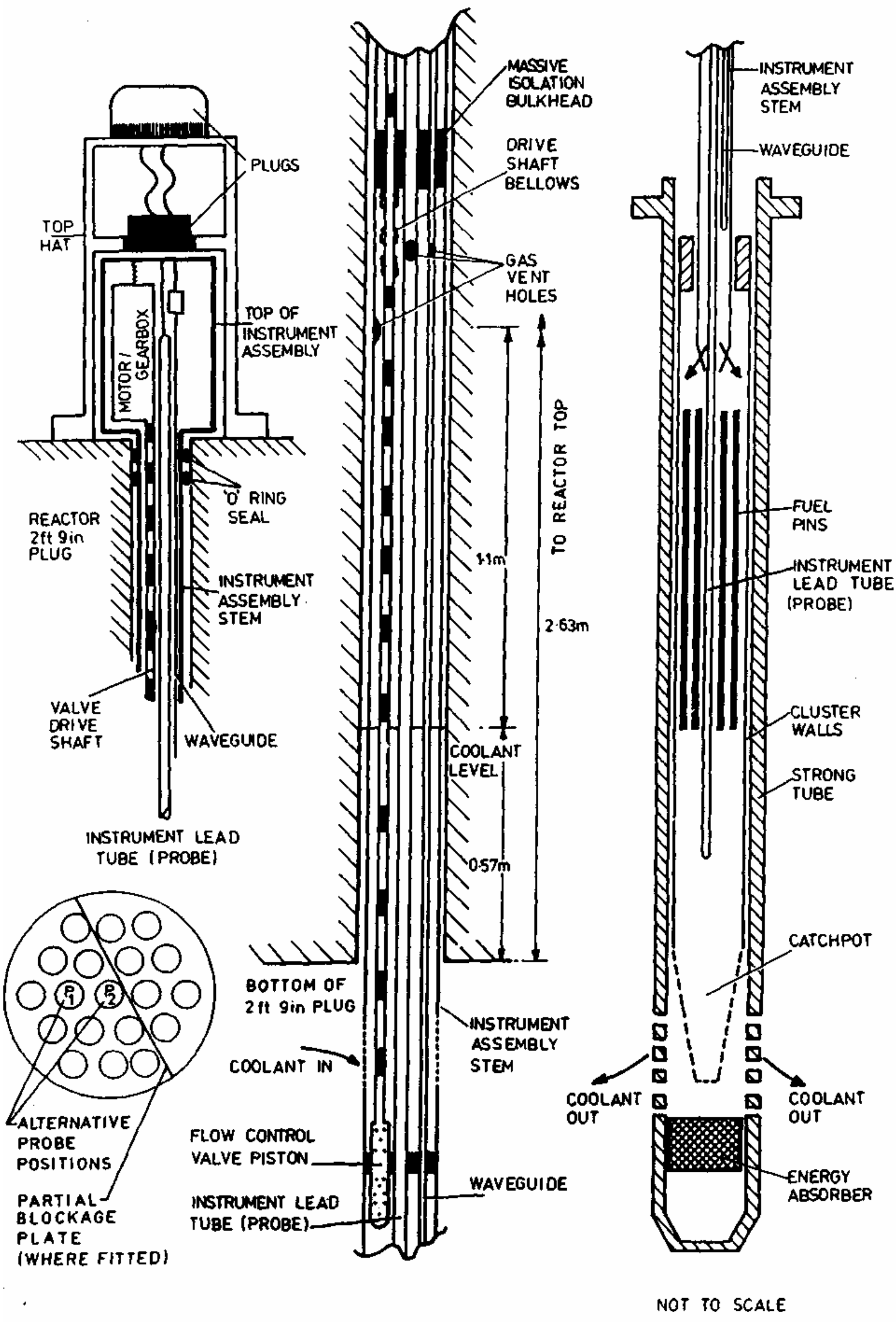

FIG. 4. Simplified sketch of mini-subassembly.

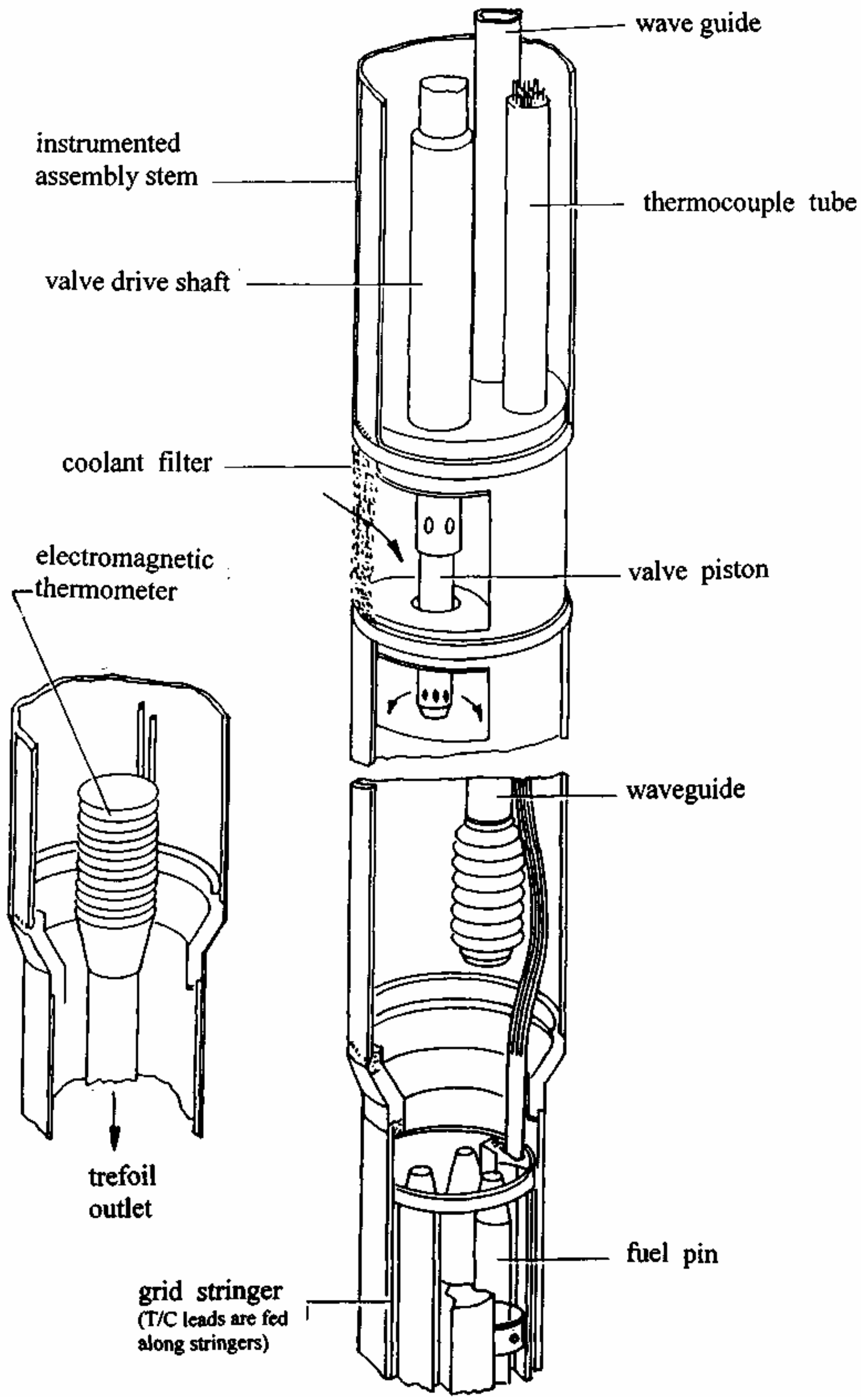

FIG. 5. Simplified sketch of trefoil.

The plate was welded to the downstream edge of a honeycomb grid and sited approximately midway along the heated length. There was a narrow, internally shaped, annular gap between the blockage plate and each of the fuel pins which passed through the plate; the effect of the resulting small leakage flow upon the wake has been examined in out-of-pile water tests.

The trefoil vehicle in shared many features with the mini-subassemblies, e.g. the flow control valve design; the most distinct difference, apart from the number of fuel pins, was the introduction of instrumentation directly into and below the fuel cluster without the use of a separate instrument probe; this was made possible by the use of unirradiated fuel allowing the rigs to be assembled in one piece in the laboratory (Fig. 5). Thus trefoil experiments possess thermocouple hot junctions sited in the main coolant flow, i.e. proud of any supporting structure, and, in the final trefoil experiment, an acoustic sensor positioned close to the boiling region. All three trefoils contained unirradiated mixed oxide fuel; for two trefoil experiments, 522 and 540, the specification was similar to that used for the 523 series mini-subassemblies; for the remaining trefoil, 536, the fuel pins were 6.68 mm outside diameter, spirally wrapped and contained vibrocompacted fuel at 80% theoretical density (TD) over a height of 569 mm.

In general, acoustic sensors were situated at upstream of the fuel bundle and thermal sensors were positioned at inlet, midplane, outlet, and, when appropriate, in the wake region downstream of the blockage. The instrument probe carrying the thermal sensors into the mini-subassembly fuel bundles passed centrally in the 523 series but was displaced to an off central position in the local blockage experiments so as to provide a better guide to the wake temperatures (Fig. 4).

Three types of thermocouple were used in the experiments; mineral insulated stainless steel sheath bifilar chromel/alumel 1 mm diameter conventional thermocouples; fast-response thermocouples consisting of filaments of chromel and alumel butt welded together inside mineral insulated stainless steel sheaths to form a single core coaxial cable, OD from 0.5 to 1 mm [2]; and, in the last of the 528 mini-subassemblies only, an 'intrinsic thermocouple' based upon the principle described in [3] and allowing a very high frequency response. Experience with all three types of sensor was excellent throughout the programme and of the many thermocouples used only one gave any sign of failure.

Three types of acoustic sensor were employed in the experiments, a conventional waveguide/accelerometer, a RNL high temperature microphone, and a miniature Capacitance microphone. Apart from the final trefoil experiment all the acoustic sensors were sited approximately 300 mm upstream from the start of the fuel bundle. The RNL microphone was based around a lithium niobate crystal and was reduced in size especially for the DFR programme to an outside diameter of 11 mm. The DNE miniature microphone was built from 3 mm MISS single core cable and has a flattened 'sensitive' end of approximately 35 mm×5 mm×2 mm; because of its small size it can be inserted into otherwise inaccessible locations. In the final trefoil experiment a miniature microphone survived a temperature of nearly 1 000°C for many hours. In addition to the thermal and acoustic sensors described above selected rigs also incorporated void detectors and electromagnetic devices designed to detect small rapid changes in local coolant resistivity caused by temperature fluctuations or voids.

During the whole of each experiment's in-pile life the signals from the installed sensors were recorded visually on paper charts and, as a permanent record, on magnetic tape; on-line processing of thermal and acoustic noise data was provided by RNL during the majority of boiling runs. All these data have now been edited and a series of reference 'archive' magnetic tapes produced. The raw analogue data from the magnetic tapes has been normalized, to account for the amplification in the recording system, and digitized to provide a computer data bank of temperatures for each boiling run of each rig.

8.2.4. Operation

The experiments were irradiated during the last three power runs of the DFR, i.e. in the period December 1975 to March 1977; the experiments were loaded and discharged, with one exception, as part of the normal reactor loading schedule and were subject to the same stringent safety criteria as other rigs, both with regard to the safety of the core and to possible interference with other experiments.

To complement out-of-pile hydraulic tests, in-pile calibrations of the throttled rigs, i.e. curves of valve position against corresponding coolant temperature rise across the rig, were carried out during the period of low-power operation at the beginning of each reactor run.

Additional calibrations were also carried out, if sufficient time was available, at the end of a reactor run and following any reduction in reactor power in an attempt to identify changes in the flow characteristics of the rigs as a result of boiling. Low power calibrations could be scaled to higher reactor powers and deviations between such scaled calibrations and data measured at the higher power attributed, in principle, to the presence of coolant boiling.

The simple intention behind the operation of each rig was to expose the fuel pins to periods of prolonged boiling; the mode of operation was straightforward. Because there was little previous experience to act as a guide, coolant boiling was approached slowly in the early experiments and the periods of boiling kept relatively short; conditions were made more rigorous as confidence was gained. Thus in the lead experiment, DFR 522, the outlet temperature was raised from 620 to 900°C, at which sub-cooled boiling was first detected in slightly more than 3 hours and boiling conditions were maintained for less than one hour. For the final boiling run of an exactly similar rig, the DFR 540, at the end of the programme, outlet temperatures in excess of 900°C were reached in less than an hour and boiling maintained for approximately 3.5 hours.

The coolant flow rate was reduced to a level which was 25% below that needed to induce boiling. A typical boiling run proceeded by reducing flow through the rig using the control valve until coolant boiling was detected; the flow was reduced by a further 10 to ~ 30% and the rig left for a predetermined period of time; finally the valve was opened and the rig left until the next approach to boiling or the end of the reactor run. During their in-pile lifetime the rigs were subjected to several boiling runs, e.g. in the local blockage experiments DFR 539/2 and 539/3 they boiled 5 and 11 times, respectively. A summary of the overall programme is provided in Table 1 and represents a total of 600 pin boiling hours; the longest total boiling time, 24 hours, was experienced by 13 pins in DFR 539/3.

Three different indicators of boiling were recognized during the programme. Bulk boiling at the rig outlet was readily detected by the measured outlet temperature reaching the saturation temperature and by noting the invariance of the outlet temperature against a further decrease in coolant flow, as measured by the increase in midplane temperature.

Analysis of the lead experiment, DFR 522, showed that the inception of sub-cooled boiling within the bundle was accompanied by an increase in the thermal noise signal at the outlet thermocouples; the signal increased with further decrease in flow (i.e. increasing severity of boiling) until it reached a peak and thereafter decreased to a low value as bulk boiling was approached. For all subsequent rigs thermal noise was monitored on line by a team of experts from RNL; details of both this and the acoustic technique referred to in the next paragraph will be published elsewhere and it is sufficient to say here that consistent and apparently reliable indications of the onset and development of boiling were made. Water tests in an electrically heated replica of the local blockage experiment confirmed that the thermal noise signal at the outlet thermocouple increased following the inception of fairly extensive nucleate boiling and increased further as the boiling developed.

In these experiments it was found that on-line examination of the RMS level and spectral content of the signals from the acoustic sensors sited upstream of the bundle was unable to provide any reliable indication of boiling inception.

A pulse technique developed by RNL and based around two separate detectors provided the only reliable acoustic indications of boiling and for the majority of rigs gave results consistent with those from the thermal noise analysis. In one rig an acoustic detector was sited close to the boiling region and provided signals which have been interpreted as indicating sub-cooled nucleate boiling well before the outlet thermocouple showed any significant rise in thermal noise signal; this was consistent with signals from an electromagnetic sensor at the rig outlet and with observations from the water rig.

8.2.5. Analysis of the results and conclusions

The most important feature of the experiments is the behaviour of the fuel pins and the first objective of the supporting analytical programme was to assist and complement the Post Irradiation Examination (PIE) of the fuel pins by specifying the thermalhydraulic conditions experienced by each set of pins. A subsidiary objective is the determination of timescales at which significant points- in the rigs' histories were reached, such as fuel failure, from changes in the measured thermalhydraulic data.

The CLAYMORE [4] code was developed to help reach these objectives. CLAYMORE is a thermal hydraulics sub-channel code capable of describing local and bulk boiling, it includes a homogeneous two-phase flow model with slip, and reverse flow, and i.e. it can represent the wake downstream of a local blockage. It is necessary to substantiate predictions from a code such as CLAYMORE and to this end comparisons were made with:

- Measurements made in a water-cooled replica of the local blockage rigs using electrically heated pins; and
- With data from single pin and multi pin out of pile sodium experiments.

The results of such comparisons showed there to be a broad measure of agreement between prediction and measurement, in particular the description of the development of local boiling appears to be qualitatively correct.

When applied to the DFR experiments the code has adequately predicted the inception of boiling and such gross effects as the two-phase pressure drop/flow curve but detailed comparisons show that there are aspects of the rigs' behaviour which remained unexplained.

8.2.5.1. Post irradiation examination (PIE)

The results with regard to PIE are summarized in Table 1.

TABLE 1. SUMMARY OF THE DFR INCORE COOLANT BOILING EXPERIMENTS [1]

Rig	Brief rig description	Boiling history	Boiling duration	Comments
T522	3 unirradiated prepressurized mixed oxide fuel pins. Max. rating, 315 W/cm per pin.	One boiling run at end of run 79. This constituted a lead experiment for the series.	Approx. 1 hour	Boiling was clearly detected at the time by the constant boiling temperature limiting at saturation temperature and, retrospectively, by thermal noise analysis; examination of fuel pins showed no sign of damage.
M(b)539/l	18 reference design fuel pins, burnups in range 0-10%, vibro and pelleted fuel, different clad types. Blockage plate covering approx. 70% of flow area.	One boiling run, but experiment terminated because of severe gas entrainment problems leading to extensive clad failure in the wake region and self induced boiling.		Despite the severity of the conditions to which the rig was subjected (failed pins exposed to hot coolant for the order of 10 days) there was no sign of fuel melting; nor was there any sign of rapid escalation of the fault condition.
T536	3 fat unirradiated spiral wrap mixed oxide pins. Max. rating 416 W/cm per pin.	Three boiling runs of approx. one hour duration each. On two occasions a self limiting temp., excursion was observed. First and second boiling runs approx. 1 week apart.	Approx. 4 hours	Boiling detected by thermal noise and acoustics on line. Examination of fuel pins showed no sign of damage.
M528/1	18 reference design fuel pins at 0.3% and 6% B.U. One pin contains an artificial defect at its hot end.	Two boiling runs, each run showed similar, steady and repeatable behaviour.	Approx. 2.5 hours	Boiling detected by both thermal and acoustic noise techniques on line at similar rig temperatures. Examination of bundle showed bowing of pins at the hot end and one pin had failed - no signs of clad melting or of any fuel loss.
M(b)539/2[13]	As for 539/1.	Five boiling runs spread over 4 weeks. In the intervals between the first four boiling runs the rig was left at approx. 100°C below saturation at hottest point.	Approx. 4 hours	Boiling detected by thermal noise measurements on line. Examination of rig showed marked bowing of pins at hot end. Six pins had failed with numerous longitudinal cracks. One also had a small zone of molten clad. No observable fuel loss.

[13] M represents mini subassembly, M(b) represents mini-subassembly with plate blockage, T represents trefoil

TABLE 1 - SUMMARY OF THE DFR INCORE COOLANT BOILING EXPERIMENTS [1]

Cont'd

Rig	Brief rig description	Boiling history	Boiling duration	Comments
M(b)539/2[14]	As for 539/1.	Five boiling runs spread over 4 weeks. In the intervals between the first four boiling runs the rig was left at approx. 100°C below saturation at hottest point.	Approx. 4 hours	Boiling detected by thermal noise measurements on line. Examination of rig showed marked bowing of pins at hot end. Six pins had failed with numerous longitudinal cracks. One also had a small zone of molten clad. No observable fuel loss.
M528/2	As for 523/1, proportion of pins at differing burnups changed.	Two boiling runs of approx. 3 hours duration ten days apart.	Approx. 6 hours	Boiling clearly detected with thermal noise measurements and consistent results from acoustic noise. Evidence of coolant chugging for outlet temp, exceeding 900°C. Nondestructive examination showed the pin array to be regular with no obvious pin failures. Swelling and distortion at hot end.
M(b)539/3	As for 539/1.	11 approaches to boiling spread over period of a month. For most runs the period of boiling lay between 2 and 3 hours. A total of more than 7 hours was spent with the rig outlet temp exceeding 900°C with a significant fraction above 960°C.	Approx. 24 hours	Increase in thermal noise signal with boiling less pronounced although qualitatively the same as in previous rigs. Radiographs of the rig showed obvious signs of pin failures with possible fuel loss. During some of the later boiling runs flows were reduced to 75% of their value at boiling inception.
T540	Similar to 522, but pins unpressurised and fuel in pelletted form.	Two boiling runs of approx. 1.5 and 3.5 hours duration. Saturation temp., was maintained steady at outlet and reached at the midplane.	Approx. 5 hours	Thermal noise and acoustic noise both indicated boiling. Examination of pins showed distortion but no significant diameter change - no sign of failure of any description.

[14] M represents mini subassembly, M(b) represents mini-subassembly with plate blockage, T represents trefoil

8.2.5.2. Trefoil behavior

PIE shows no signs of failure in any of the trefoil (previously unirradiated) fuel pins. Measured diameter changes are small showing no apparent effect caused by prolonged high temperature operation: the DFR 540 coolant outlet temperature was above 800°C for more than 6 hours, above 900°C for more than 5.6 hours, and at a steady 960°C for just over 4 hours.

8.2.5.3. Mini-subassemblies without local blockage

The first of the DFR 528 series has been radiographed and examined visually, the second experiment has only been radiographed. Both bundles show significant bowing of the pins, and some swelling over the hottest region, otherwise the pin arrays are regular. DFR 523/1 contained only one failure; there was either sign neither of fuel loss nor of clad or fuel melting.

8.2.5.4. Mini-subassemblies with local blockade

The first local blockage experiment inadvertently entrained gas into the wake region causing fuel failure, including clad melting, soon after the first rise to full power. Boiling of the rig occurred after a further 10 days irradiation and led to a further region of fuel failure downstream of the wake. Detailed Post Irradiation Examination has been performed on the rig; there is no evidence of any fuel melting despite the severity of the incident; there are several examples of sintered fuel remaining intact despite removal of clad from the failed region; there was no sign of any secondary blockage formation at the downstream grids despite the loss of complete sections of some fuel pins. The mechanism of gas entrainment into the wake and subsequent fuel failure has been demonstrated using the heated water rig.

The second blockage experiment has been examined visually. Six pin failures were found, including small regions of clad melting. There was some small observable fuel loss but no sign of steady blockage growth. The final blockage experiment has only been examined radiographically but there were obvious signs that at least three pins have failed extensively; however the rig was boiled vigorously for nearly 24 hours. In the experiments 600 pin-hours of detectable boiling were accumulated. No failures occurred in the pins with 3% burnup, seven failures arose amongst the seventeen pins with greater than 6% burnup. From the results obtained it can be concluded that:

— DFR fuel pins were extremely durable, even when subjected to severe incident conditions;
— There was no sign that fuel failure, even when subjected to continuing irradiation and boiling, leads to a rapidly developing subassembly incident;
— The time available for the detection of the initial incident is long compared to the response times of all potential reactor monitoring systems;
— Thermal noise was an extremely effective and reliable method for detecting coolant boiling.

8.2.5.5. Relevance to safety studies

Because of the low, downward flow in these experiments the effect of buoyancy upon boiling and entrained gas behaviour ensured that the experiments represent conditions which are more severe than the corresponding situation in a modern LMFR with high, upward coolant flow. In particular the results have shown that:

— Boiling was detected at an early stage;
— Prolonged boiling was stable;
— Boiling did not necessarily lead to fuel failure;
— Fuel failure did not necessarily lead to fuel release;
— Fuel release did not lead to the formation of a local blockage;
— A local blockage did not lead to fuel melting;
— The combination of gas entrainment and a local blockage with consequent hold-up of gas in the downstream wake did not lead to fuel melting.

Experimental studies on heat removal from the core by the boiling coolant carried out at the IPPE (fuel element simulators were tested in sodium-potassium coolant) shown possibility of the long term heat removal from the fuel elements by the boiling coolant under conditions of natural flow and heat flux up to 270 kW/m^2 (without fuel element simulator damage).

8.2.5.6. Relevance to sodium voiding reactivity

As pointed out in Ref. [5], historically, the desire to lower the amplitude of the positive sodium void worth in LMFR was driven by a concern over extremely high rates of sodium voiding. This concern had several origins:

— First, an observation in laboratory experiments showed that the onset of boiling in liquid sodium can occur at temperatures significantly higher than the boiling point before any vapor was formed. This would imply that once vapor began forming, the entire superheated coolant inventory in the core could void almost instantaneously without further heat addition;
— Second, there was the potential for vapor explosions wherein extremely high temperature fuel suddenly breached through cladding and mixed intimately with liquid sodium [6]. Finally, any case of rapid voiding could result in high positive reactivity ramp rates which might act faster than the negative feedbacks could respond and would take the core into the super prompt critical range.

However, experimental data on boiling in tube bundles shown that the in-core sodium boiling process in fact does not reach high superheat, but rather comprises a series of local pressurization and flow reversals which voids part of an assembly for a short period of time. Detailed analyses have shown substantial spatial and temporal incoherence in the boiling process, with incoherent chugging and a few assemblies “leading” the rest of core [6].

8.2.5.7. Relevance to reactor core acoustic instrumentation

Acoustic devices have been used successfully to detect boiling at DFR. Therefore, an acoustic detection of boiling can play an important part in the plant protective system, because it affords a means of detecting local overheating and allows protective action to be taken before damage spreads to otherwise unaffected parts of the structure.

The objective of the protective system is to intervene by shutting down the reactor, preferably soon enough to prevent fuel melting, certainly soon enough to prevent the propagation of damage beyond the affected subassembly. An acoustic instrumentation has an important role in providing a signal, indicating that something is amiss, to initiate the protective action. Its importance lies in the fact that it may afford the only way of knowing promptly that the coolant is boiling.

As an example of the type of local fault against which acoustic boiling noise detection (ABND) provides protection, it is common to consider a complete instantaneous blockage of the coolant flow at inlet to a core subassembly while the reactor is operating at full power.

The performance of ABND systems was the subject of an IAEA Coordinated Research Programme [7], aimed at determining whether boiling could be detected reliably, in the presence of background noise, with an acceptably low spurious detection rate. Background noise from the dummy cores of PFR and Super-Phénix, and from the operating cores of PFR and KNK-2, was recorded, and from various out-of-pile test rigs, was recorded.

Various software methods for detecting boiling noise in the presence of background have been investigated [8]. The use of filtration, pattern recognition techniques, or a combination of both, allows boiling to be detected in the presence of a background which is noisier than the boiling source.

The IAEA programme showed that with a signal-to-noise ration of -12dB [i.e. with the root mean square method (RMS) boiling signal power about a factor of 16 lower than the RMS background signal power], boiling can be detected with a reliability of less than one error in 10^6, and with less than one spurious indication in 10^6years.

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9. SODIUM COOLED FAST REACTORS DEVELOPMENT AND OPERATING EXPERIENCE: LESSONS LEARNED IN THE PAST AND CHALLENGES FOR THE FUTURE

9.1. MASTERING OF SODIUM COOLANT TECHNOLOGY

Excellent thermophysical properties of sodium were demonstrated at the first test facilities as early as in the 1950s, and promised its large-scale application in the power area.

During the period 1958-1963, design studies were made on the use of sodium as a coolant of traditional boilers: sodium heated by burning fossil fuel in the furnace was used to transport heat to the steam generator (SG), located in the vicinity of the turbine. This resulted in considerable decrease the length of steam/water pipelines. However, this option was abandoned because of complicated sodium handling for traditional power engineering.

At the same period, the sodium-graphite thermal neutron power reactor Hallam [240 MW(th), 75 MW(e)] was constructed and put into operation in the United States in 1962. Sodium was used to cool the reactor core and to transport heat to the double wall SG; graphite blocks with steel lining being used as moderator. In case of lining integrity loss, sodium-graphite interaction occurred in the blocks resulted in graphite swelling and lining rupture. This circumstance along with the complexity of sodium technology in comparison with water caused premature closing down of the reactor in 1966, and abandonment of this concept.

A large size solar facility [500 kW(e)] at Almeria in Spain with 70 tons of sodium used to transfer the solar heat energy to the boiler, had faced technological and safety problems. In order to work on the sodium circuit, the operators should have cooled the piping to freeze the sodium at ambient temperature, then they could cut the circuit. But a problem occurred with the formation of the solid sodium plug; pressurised sodium spewed through the cut made, splashing off the nearby steel structures and causing a fire in the hall. Thus, non-observation of special requirements to maintenance of sodium components resulted in pouring out large amounts of sodium (14 tons at 225°C). The sodium burnt in the atmosphere and the temperature reached an estimated 1200°C during approximately 15 minutes. It caused considerable damage (the metallic beams distorted and a hole was blown in the roof), and the facility was closed down in 1986 [1].

Ten experimental and six prototype and commercial size sodium cooled fast reactors have been constructed and operated. The worldwide investment already made in the development of this unique technology exceeds US$ 50 billion [2, 3]. Robustness of the sodium systems (active zone cooling and heat transfer to SG or sodium/air heat exchanger) as a whole was mastered by operation of fast reactors in various countries.

The sodium production meeting reactor standard requirements was developed and mastered at plants; and measuring instruments and devises were developed for controlling sodium quality and parameters required for reactor's systems. For the most vulnerable element of the nuclear power plant, i.e. the SG heated by sodium, special systems were developed for detection of water-to-sodium leaks and protection of the SG against consequences of sodium-water reaction. Problems related to the design of pumps with bearings lubricated by the pumped liquid (sodium) were successfully solved.

Appropriate steels (mainly standard stainless steels) were chosen out of the available range of structural materials on the basis of their irradiation in research and then in demonstration reactors. The reactors themselves and, more frequently, components have showed remarkable performance well in excess of design expectations. The overall experience with sodium cooled fast reactors in many cases has been extremely good, in spite of some setbacks.

Incidents occurred in the process of removal of sodium residues from the drain tanks using heavy alcohol resulted in the extensive local damage of the auxiliary building structures at two test facilities in Europe. Appropriate studies to the use of alcohol together with sodium had been performed. It was concluded that under certain circumstances (e.g. closed geometries, sodium puddles) the use of alcohol to clean components or to destroy sodium can be dangerous.

The Japanese prototype fast reactor MONJU of 280 MW(e) power was shut down in December 1995 during the 40% power pre-operational testing phase due to a leak and fire in the non-radioactive secondary sodium heat transport system. The cause of the failure of the well tube is considered to be high-cycle fatigue due to flow-induced vibration.

These events gave rise not only to perform design modifications aiming at the confinement of possible leaks and the protection of structures against sodium fire, but also to accelerate R&D and code development with respect to sodium spray fires.

The above mentioned issues, as well as fast reactor development declined, have forced researchers to complete the knowledge on fast reactor technology with alternative coolants: gas (He and CO_2), steam and heavy liquid metals[15] (lead-bismuth and lead). Less imminent requirements to the breeding capability have made it possible to expand the range of coolants under study, since one of advantages achieved by sodium, i.e. high power densities of cores assuring effective fuel breeding was no longer so important. Of course, some sodium cooled fast reactors had faced operating and technological issues, but it should be noted that this would be true for any other reactor line at the initial stage of development, as this type of reactors is. Full industrial development of fast reactors has not been completed yet. It is simply too early at the prototype stage of development to more general view of sodium cooled fast reactor technology. Other reactor technologies, including water cooled reactors, achieved high reliability when their respective large scale introduction had taken place. We cannot say that this will not happen in the case of LMFR. In this context, sodium's attractive properties should be recalled, namely: compatibility with traditional structural materials and all fuel compounds up to high temperatures owing to its corrosion inertness and pressure close to atmospheric value, excellent thermohydraulic characteristics, thus assuring effective heat removal under conditions of either nominal flow rate or natural circulation flow rate at the reactor. Therefore, on condition of proper design and manufacture of components, there are no physical factors provoking failures. Consequently, in most countries involved in fast reactors R&D activities, sodium cooled reactor line is considered showing promise.

Sodium is relatively cheap compared to other liquid coolants, and its resource is vast. In the nineties of the last century, the technology of sodium as fast reactor coolant was mastered and brought to a demonstration commercial level experts believe, that sodium could even more consolidate its stand as fast reactor coolant, if the designers used vast experience gained in this area in order to develop advanced approach as to the design of nuclear steam supply systems, for its simplification and economics improvement.

[15] Although, on the basis of experience, it is fair to assert that problems and risks from recognized shortcomings (e.g. sodium fire) are less worrisome than those due to hidden effects chronically present in the system (e.g. corrosion effects by heavy coolant).

Three fast reactors, which are now under construction: the experimental CEFR (China), the prototype PFBR (India), and the commercial BN-800 (Russian Federation) use sodium as coolant. Out of the selected reactor systems for Generation IV, the sodium cooled fast reactor received international interest and support as reported at the Policy Group Meeting of the Generation IV International Forum (GIF), Cape Town, South Africa, March 2003.

There is no doubt that experience gained in the world could serve as a basis for the development of sodium cooled fast reactors using so called traditional approaches. However, there is a great need to continue design studies to decrease the cost of construction, operating and decommissioning of the reactor, and to improve the robustness of the sodium cooled fast reactors technology in general. The points causing higher costs compared to that of LWR are the large mass of structural material (mainly steel) in the reactor system as well as the comparatively complicated reactor and balance of power design. It is an important achievement in the fast reactor area of nuclear power that two demonstration commercial size pool type fast reactors: BN-600 and Super-Phénix have been constructed and operated (Fig. 1).

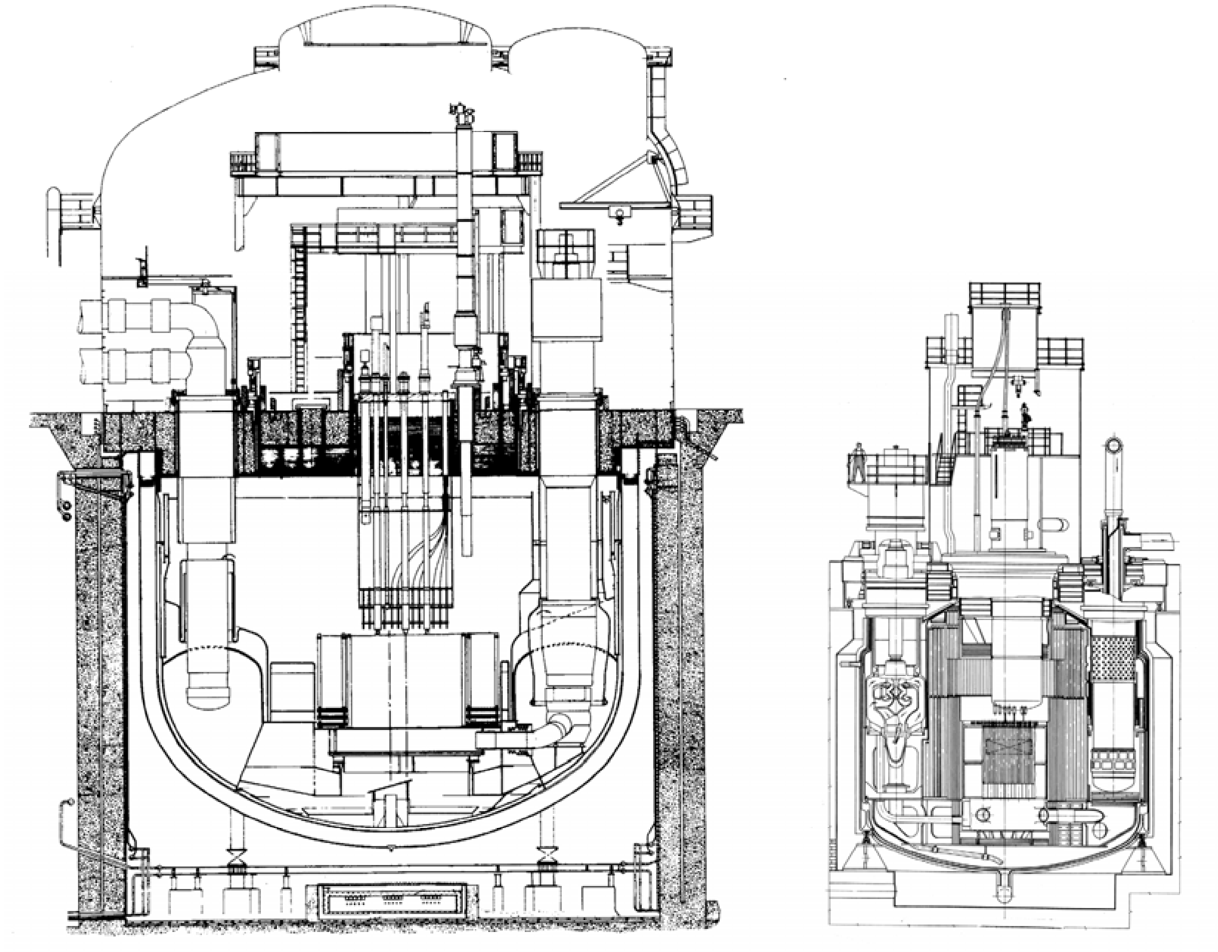


FIG. 1. Cross-section of the primary circuits: Super-Phénix - on the left, BN-600 - on the right in the same scale.

9.2. DEMONSTRATION COMMERCIAL FAST REACTORS OPERATING EXPERIENCE: ACHIEVEMENTS AND SETBACKS

9.2.1. BN-600

The power plant BN-600 was connected to the grid in April 1980. Since to reach the steady state of the core three refuelling cycles were needed, full power was reached in October 1981. Reactor operation is stable and the average load factor (φ) is about 75% (Fig. 2).

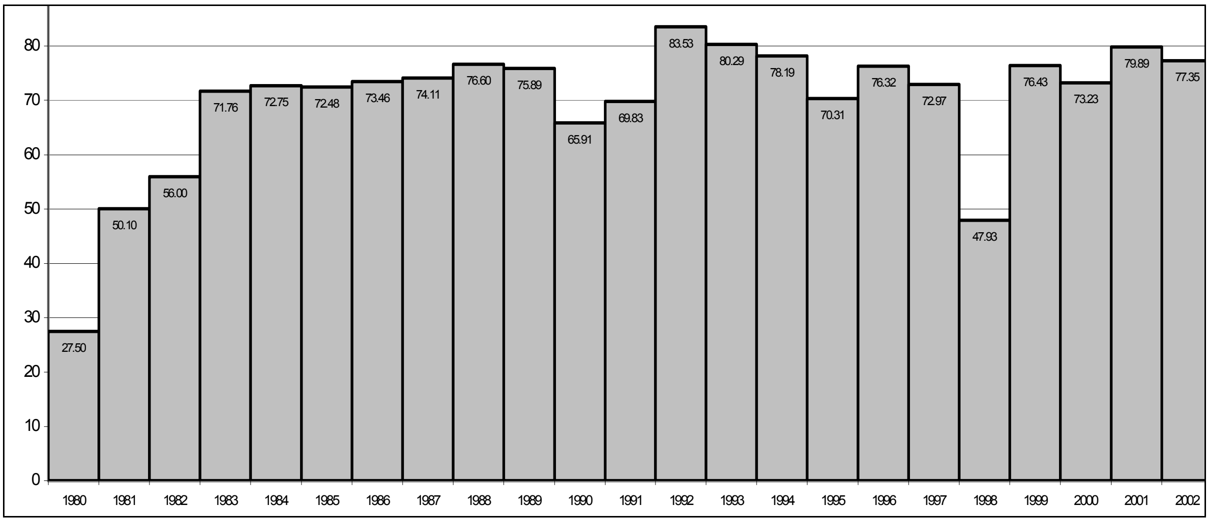


FIG. 2. BN-600: annual load factors[16],%.

Routine events, including four small (0.1-3.0 L) leaks in the primary auxiliary sodium and gas systems and more leaks in the secondary auxiliary system occurred mainly during the pre-operational testing phase (SG's valve sealings – 4 leaks; drain and blow-of lines – 11 leaks; sodium reception and storage – 7 leaks). However, almost no events exceeding normal operating conditions have occurred except one: the large (800 kg) sodium leak in October 1993 from the external primary sodium purification system rated at level 0-1 of the International Nuclear Events Scale (INES) [4]. Due to the fire fighting and protective systems operated in a proper way, the local damage was insignificant and repair was affected quickly. Of the twelve cases of steam into sodium leak occurred mainly in the first year of operation in the superheater and reheater modules owing to develop of latent manufacturing defects; in only two cases was it necessary to shut down the corresponding steam generator (SG), and in one case the reactor. In the other cases, since there is some excess of heat transfer surface of the modular SG and the presence of valves on steam-water and sodium side, the appropriate section was disconnected and the SG as a whole continued to operate without reducing the reactor power. That was the outstanding operating advantage of the modular SG, but not the cost advantage of the complicated and steel-intensive design.

The overall experience with BN-600 power plant has been rather successful, reactor themselves as well as, components with sodium coolant: intermediate heat exchangers, pumps, reactor refuelling systems, reactor cooling and heat transfer systems showing remarkable performances well in accordance with the design expectations. The majority of the BN-600 reactor components were tested at the sodium facilities simulating the design operating conditions; larger R&D work programs ($ 12 billion) [4] led to success.

[16] Reactor is being shutdown two times per year for refuelling: in 1998 - for the rotating plug repair.

9.2.2. Super-Phénix (SPX)

The largest LMFR in the world SPX was successfully constructed and connected to the grid in January 1986. The SPX plant reached its full power was in December 1986. However, these have been a difficulty in operation: the operation was interrupted by tenth events and incidents nine of which rated at level 1-2 of INES [5, 6]. Figure 3 displays an operating and administrative history of the SPX power plant.

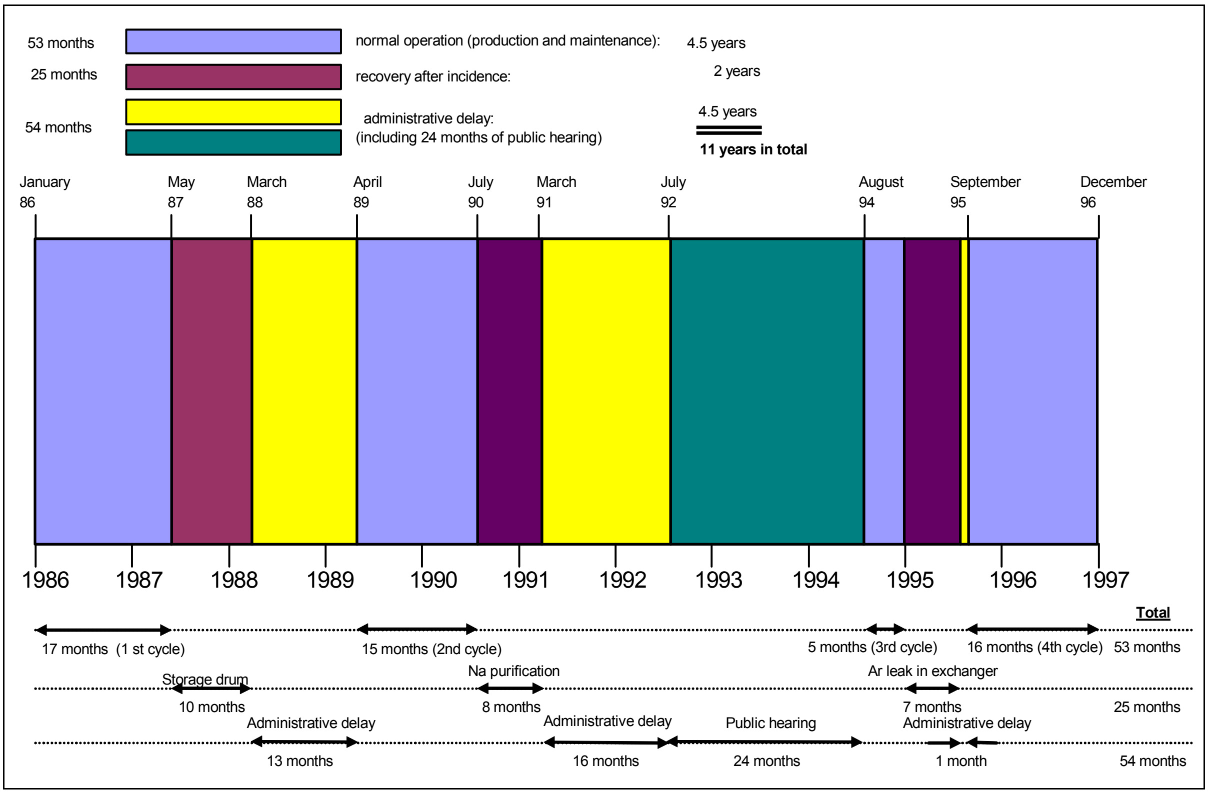


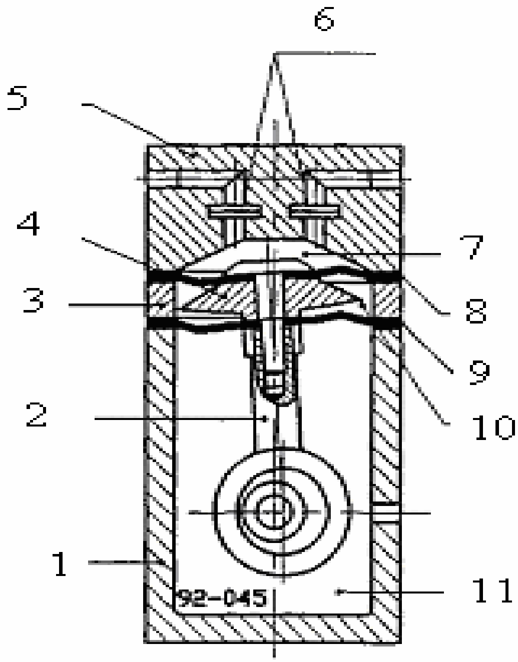
FIG. 3. The operating and administrative history of SPX.

During 11 years of the SPX power plant existence (1986-1997, Fig. 3), it was in operation for 4.5 years. Repair procedures for elimination of consequences of incidents took two years; administrative and social procedures concerning license issue for reactor start-up took more than 4 years. During 4 cycles (53 months, ~ 4.5 years, Fig. 3), the SPX produced 7.9 billion kWh [6]. The outstanding success of the SPX operation has undoubtedly been the demonstration of reliable (without leaks) operating of SGs with high self power [750 MW(th)]. Three incidents marred the commissioning procedure and caused lengthy delays:

— Cracks and a leak in the space between the storage drum vessels on 8 March 1987 were caused by inappropriate use fuel storage drum vessel (FSD) of carbon steel instead of stainless steel. The repair and other procedures (a 23-month reactor shutdown) required dismantling the FSD and conversion of 700 tones of the residual sodium into sodium carbonate through a controlled additional of water vapor and carbon dioxide gas.
— Lack of instruments for impurities control in cover gas of the SPX reactor and practically atmospheric cover gas pressure caused uncontrolled air inflow into the reactor in July during 3 weeks 1992 through a failed compressor membrane of the gas

radiometer that caused oxidation of a large sodium fraction (Fig. 4) [7]. Purification of reactor coolant from oxides (~ 400 kg) required installation of additional cold traps into the reactor. The result of the air ingress into the reactor was a 24-month reactor shutdown.

— Gas (argon) leakage to sodium through the crack in the weld of piping of the argon supply to the sealing bell surrounding the IHX in January 1995 (8-month reactor shutdown).



1-compressor bode, 2-main rod, 3-intermediate ring, 4-top of connecting rod, 5-head, 6-closures, 7-compression chamber, 8-neoprene working diaphragm, 9-neoprene safety diaphragm, 10-safety chamber, 11-control unit

FIG. 4. Scheme of the radiometer compression.

These events as well as a very strong anti-nuclear social movement in the post Chernobyl period, and a sodium fire at the Almeria facility that coincided with SPX reactor start-up tests and power increase, had considerable political repercussions, caused long reactor outage (Fig. 3) and provided one of the reasons for which the SPX reactor was prematurely closed down by the left French Government. It should be noted, that the experience gained in the development, construction and operation of the BN-600, SPX and other demonstration and prototype reactors are invaluable for the development of fast reactor technology, in particular, for the design of the commercial size advanced reactor projects, e.g. EFR, BN-800 and other, having much better technical characteristics and indicating that the goal of robustness fast rectors is within reach.

9.3. CHALLENGES FOR THE FUTURE

9.3.1. Increase of safety margins

There is a widespread opinion that the liquid metal cooled fast reactor advanced projects can meet present safety standards quite adequately, and are expected to do so for the foreseeable future. Nevertheless it is essential to continue the search for greater safety because:

— No opportunity to improve technology should be ignored;
— Safety standards may be raised by the regulatory authorities;
— Reactors suitable for countries with less developed infrastructure will be needed; and
— More cost-effective ways of meeting existing safety standards may be found.

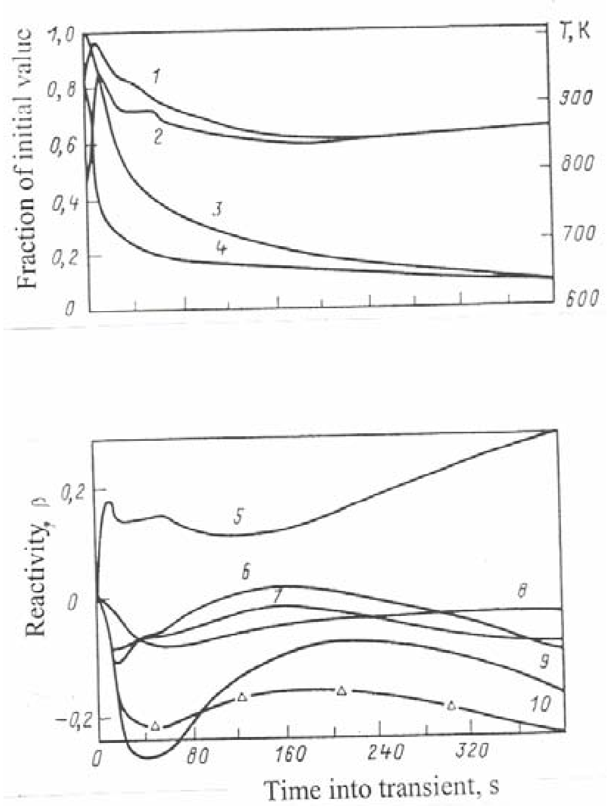
A fast reactor with the internal breeding ratio ~ 1.0 and sodium coolant, on principle, provides a potential to design a nuclear system in which the power would eventually passively adjust itself due to self-regulated heat production and heat removal [8] owing to:

— Near zero burnup reactivity swing;
— Strong negative temperature reactivity effects;
— Sufficiently small hot to cold reactivity swing;
— Possibility of core coolant temperature rise changing in wide range,
— Impossibility of local criticality formation under conditions of any perturbation of the neutron field;
— Absence of poisoning effect (such as xenon poisoning) followed by positive reactivity insertion;
— A considerable portion of radioactive fission products is held by sodium and caught by the cold traps in case of their release from the fuel elements to the coolant;
— High vaporization heat of sodium;
— System operated at low pressure and having large margins to boiling temperature;
— Negligible thermal energy stored in the coolant available for release in the event of a leak.

Figures 5 and 6 depict variations in the principal parameters of the PRISM (power reactor innovative small module) reactor during two types of emergency situations [9]:

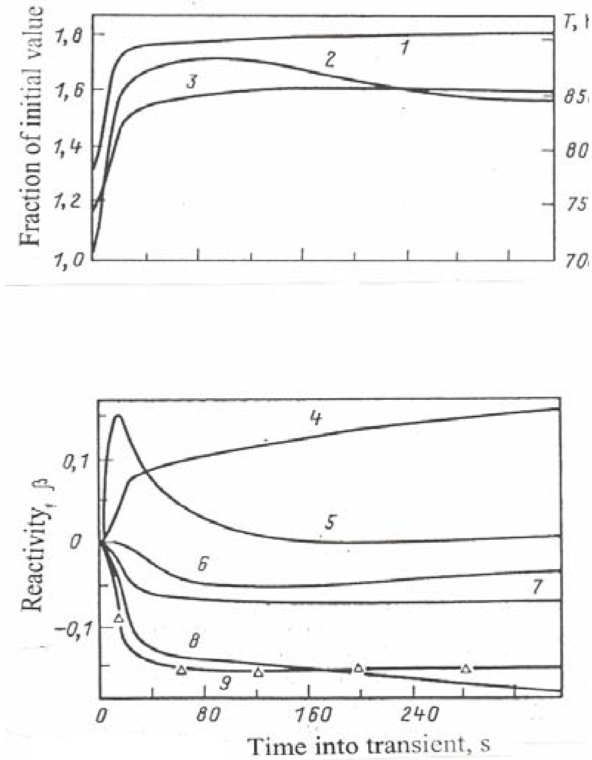
— Pump shutdown in circuits I, II, and III, accompanied by the non operation of the safety rods – an unprotected loss of flow (ULOF), Fig. 5; and
— The erroneous withdrawal of the control rods accompanied by the non operation of the safety rods – an unprotected transient overpower (UTOP), Fig. 6.

It follows from the parameters presented in Fig. 5 that, despite the use of low-inertia electromagnetic pumps, a quite gradual flow rate decrease is achieved in the primary circuit of the PRISM reactor — apparently as a result of the presence in this reactor's power supply system of synchronous electric motors with large moments of inertia. The subject reactor possesses a low pressure drop core, which ultimately ensures a high natural circulation flow rate (7-8%). The first reaction to the flow rate decrease is a fuel temperature increase, followed by a coolant temperature increase. A positive sodium density reactivity effect is achieved during the first few seconds of this process. However, negative components — the Doppler effect, the axial expansion of the fuel, and the radial expansion of the core diagrid — combine to form a resultant negative reactivity that also provides a power reduction. During the ensuing reduction of power, the fuel is cooled and the Doppler effect takes on a positive value at a particular moment in time. However, due to the use of a metal fuel with sodium filled gap that has a high degree of thermal conductivity, the difference in temperatures between the fuel and the coolant is small, which diminishes the positive component of this effect. The effect that results from the axial expansion of the fuel behaves in a similar fashion.



1-fuel temperature, 2-central fuel subassembly (FSA) coolant outlet temperature, 3-reactor power, 4-coolant flow rate, 5-sodium density, 6-Doppler, 7-axial fuel expansion, 8-control rod drive expansion, 9-total reactivity, 10-radial grid expansion

FIG. 5. PRISM parameters: ULOF.



1-central FSA outlet temperature, 2-reactor power, 3-core outlet temperature, 4-sodium density, 5-total reactivity, 6-control rod drive lines expansion, 7-axial fuel expansion, 8-radial grid expansion, 9-Doppler

FIG. 6. PRISM parameters: UTOP.

It is apparent from that these two effects - the positive sodium density effect and the negative effect – owing to the radial thermal expansions of the core diagrid are antagonistic toward one another (Fig. 5). The favourable "behaviuor" of the Doppler effect and the high sensitivity of reactivity to core geometry are dictated by the high energy neutron spectrum of a reactor which has relatively small dimensions and a metal fuel heterogeneous core. The temperature of the coolant reached a peak of 680°C at the 10-s mark, and then dropped to a constant value of ~ 600°C (Fig. 5).

Figure 6 depicts variations in the principal parameters of the PRISM reactor during the most dangerous reactor incident - the erroneous withdrawal of the control rods accompanied by the non operation of the safety rods – an unprotected transient overpower (UTOP). It was assumed over the course of the calculations performed that all six control and safety rods began to be withdrawn from the core, thereby introducing positive reactivity at a rate of 0.02 β/s. The complete withdrawal of these rods introduced a positive reactivity of 0.35 β. The power increase occurring at this moment in time was moderated by the negative reactivity effects: the Doppler effect -0.14 β; the radial expansion of the core diagrid 0.13 β, and; the axial expansion of the fuel element, 0.06 β. The elongation of the control rod drive lines set in after a slight delay, with the reactivity introduced thereby coming to -0.04 β at the 80-second mark. As a result, total reactivity began to decline and the effect produced by the withdrawal of the control rods had been neutralized entirely at the 200-second mark, at which point reactivity came to ~ 0. Despite the considerable increase in power — by 1.7 times — the temperatures in the core and throughout the reactor did not exceed the maximum permissible values [9]:

— The coolant temperature at the outlet of the most hot subassembly rose from 500 to 620°C;
— The mean coolant temperature at the core outlet was increased from 465 to 575°C;
— The temperature of the fuel reached a peak at 920°C, which is 200°C below the melting point of a metal (U-Pu-Zr) fuel.

It should be noted that the relatively favorable course of the emergency process under consideration is due to the diminished nominal temperature values used in this PRISM's design (300/465°C). Thus, in the right design, the reactor with above mentioned features would be passively protected against nearly all control system failures and operator errors. Implementation of a fast reactor design with safety based on self-regulation principles (physical laws) requires the core and its support and surrounding structures to be designed in such a way that negative reactivity insertion is provided in case of their thermal expansion and bowing in the “right direction”, that gives net reactivity sufficient for reactor power decrease down to the level at which heat removal is possible by the natural circulation coolant flow.

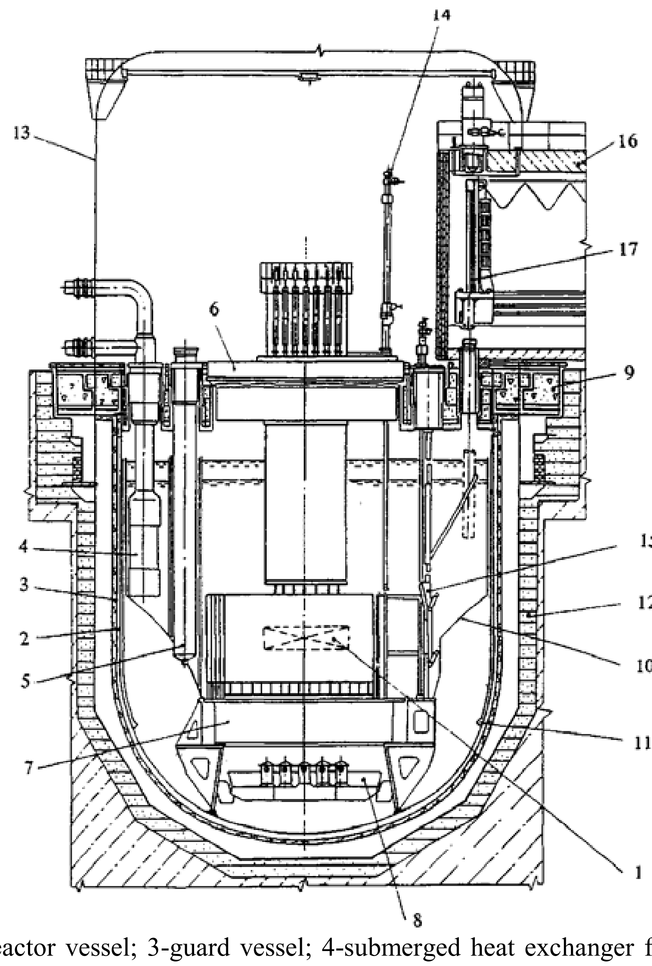
The importance of these approaches has confirmed by the results of studies, development of PRISM/ALMR reactors and during the final stage of operation of the test fast reactors Rapsodie, EBR-II and FFTF [10-12]. In these tests, the reactor pumps were switched off and reactor safety system failure was simulated with absorber rods prevented from falling down by the operator, core power decreased abruptly owing to negative reactivity effects, and heat was removed by sodium natural circulation flow. It should be pointed out, that the LMFR could become more attractive for the investors and utilities, if the designers managed to make the most of the physical and thermohydraulic advantages of this fast reactor. The complete development of the self-control and self-protection design of the sodium cooled fast breeder reactor against probable accidents including the most severe accidents, could compensate for the complication of the LMFR technology. Designers are making an effort to assure LMFR safety in the system of Generation IV reactors exclusively on a self-regulation basis. It is just because of its inherent safety features that LMFR is included in the Generation IV program. Some institutes and design organizations having know-how on this issue, are capable of solving this problem by giving a self-control capability to the sodium cooled fast reactor, which is not a characteristic for any other type of reactor, and this would reliably compensate for some drawbacks of sodium coolant.

9.3.2. Simplification of NSSS structure and operation & maintenance (O&M)

In existing nuclear power plants, the reactor and heat removal/transport systems are “enmeshed” by over half a dozen external auxiliary systems with their elements spread over various rooms belonging mainly to the “nuclear island”. The main systems and elements are as follows:

— Fuel storage drum (FSD) with related auxiliary sodium systems;
— External system for oxides removal from radioactive sodium;
— Emergency systems for decay heat removal (including heat exchangers, pumps, etc.) connected to the reactor or secondary circuit;
— Sodium heated steam reheater;
— Lubrication systems of the main sodium pumps (for the upper bearing and sealing units).

The potential of pool type fast reactors has not been exhausted regarding the integration of the main radioactive circuit and the rejection of external auxiliary sodium systems, e.g. sodium purification and emergency reactor cooling. There are some elements of sodium pipelines with T-joints in this system, where sodium flows having different temperatures are mixed, causing temperature fluctuations and, hence thermal stresses that resulted in crack formation. As was pointed out before, a crack due to the temperature fluctuations in the coolant close to a structure appeared in the T-joints of the BN-600 piping of the primary sodium purification. A design option locating cold traps and their related systems inside the reactor vessel has been realized in the SPX reactor and studies have been made of to apply advanced design option to the BN-1600 reactor project (Fig. 7). A refuelling option has been developed for the BN-350 reactor without use of ex-vessel fuel storage drum (FSD) following its failure. Similarly to SPX, designer and operator proposed a solution on the reactor refuelling without use of the FSD (there was also an incident causing the FSD failure).The latter demonstrates that this component can be avoided by means of holding spent fuel subassemblies (FSAs) in the in-vessel storage during reactor operation in order to decrease their decay heat. An additional reduction in the residual power of unloaded FSAs could be reached by incorporating one or two rows of B4C-filled assemblies between the reactor core and the store (Fig. 7).



1-reactor core; 2-main reactor vessel; 3-guard vessel; 4-submerged heat exchanger for decay heat removal; 5-cold trap; 6-rotating plug; 7-core diagrid; 8-core catcher; 9-stationary shield; 10-hot sodium collector; 11-baffle; 12-well liner; 13-containment; 14-refuelling mechanism; 15-elevator (provides removing FSAs from in-vessel storage); 16-refuelling cell; 17-fuel transfer mechanism.

FIG. 7. BN-1600 pool type reactor (project) with in-vessel fuel storage and cold trap.

The complicated and labor-consuming operation and maintenance (O&M) technology for the repair of sodium components and the absence of a guaranty that components will function after removal of sodium residues (washing) and decontamination necessitate carrying make out tests of forerunners or representative models of reactor mechanisms and components on sodium facilities under designed operating conditions. All components manufactured without defects and tested, have been operated in LMFRs for decades owing to the corrosion inertness of sodium and the low pressure.

Serious incidents occurred during the cleaning of components from sodium residues because complete sodium draining has not been enabled at the design stage. This approach was justified on the initial stage of mastering fast reactors in order to eliminate the probability of coolant leakage from the drain tanks and component's vessels. The presence of non-drained cavities in the components can be explained by insufficient understanding of the problem at that stage. Accumulated technological experience provides the basis for studying the possibility of draining the tanks and the vessels of advanced reactor facilities with deterministic elimination of unauthorized draining by design.

Oil ingress into the primary circuits of an LMFR is undesirable because of the potential release of methane gas through the core. This could cause positive reactivity effects and possible blockage of the fuel subassemblies by solid carbon debris. In future project oil bearings' are probably best avoided. Some advanced pump design was changed following the PFR and other reactors oil ingress incident by introducing innovative magnetic bearings and ferro-fluid seals [13] to eliminate oil completely and remove the potential hazard of its ingress into sodium.

9.3.3. Steels stabilized by titanium

The hot leg pipes around the steam generator up to including the buffer tanks, steam generator sodium headers and modules were made of 321 steel stabilized by titanium to improve the mechanical strength at high temperature.

The cracks in the secondary sodium piping elements with this steel (type 321) were observed at the Phénix reactor. The French expert evaluations showed that the cause of the delayed reheat crack in the thermally affected (during the welding) zones is the hardening of the steel due to tine precipitations of titanium carbide inside the crystalline structure resulted the plastic deformation capacity the periphery of the grains that provoked the crack. This phenomenon particularly has been observed in the complex geometry parts (T-junction welds, welds connections with different thickness and walls deviation) [15]. This discussed events were not observed/examined at other reactors (PFR, BN-600, BN-350), possibly, owing to lower temperature and thicker wall of the piping (Table 1) as well as using steel 18Cr 9Ni type without adding Ti (BN-350, BN-600).

TABLE 1. THE LMFR'S HOT LEG SECONDARY SODIUM PIPING DATA [14, 15]

Reactor	Steel type	Temperature, °C	Pipe diameter, mm	Pipe wall thick, mm
Phénix	321 (Ti)	550[17]	510	6.0
PFR	321 (Ti)	530	2×360	10.0
BN-350	18Cr 9Ni	415	529	12.0
BN-600	18Cr 9Ni	515	630	13.0

[17] "The higher the operating temperatures, the more crackswere found" [15]

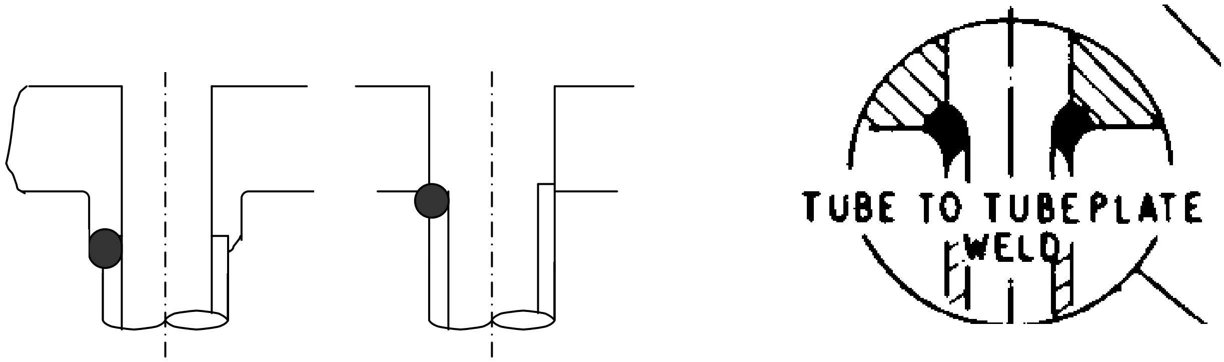
In the future sodium cooled reactor equipment designs the using of the 321 steel stabilized by titanium are probably best avoided.

It should be pointed out that an invaluable knowledge for future LMFR designs has been gained on the structural material behavior and component options thanks to Phénix's long time operation with a high level of temperatures and NPP thermal efficiency.

9.3.4. Steam generators principal design and structure materials

A total of 37 leaks were experienced in the UK prototype fast reactor (PFR) steam generator units in the period 1974 to 1984. All leaks originated at the 'butt/fillet' welds between the tubes and the tubeplates (Fig. 8c) [15].

For the first ten years (1974-1983), electrical output of the PFR plant was limited, mainly because of a series of leaks in the steam generator units, and the highest load factor in any year was 12% [15].



(a) 'stub/butt (b) 'butt/fillet' (c) PFR SG: 'butt/fillet' weld

FIG. 8. Steam generator tube/tubeplate weld joints.

The stub/butt weld (Figs 8 a and 9), permitted heat treating after manufacture, radiographic and other inspection of an individual tube to tubeplate joints, should be envisaged to avoid the risk of stress corrosion cracking and reheat cracking. Austenitic steels are in question for LMFR steam generators because of the high risk of caustic stress corrosion damage following even small leaks [15].

Two types of tube/tubeplate weld for the UK PFR SG: typical 'stub/butt' and new 'butt/fillet' welds were selected for detail consideration (Fig. 8 a-b). The 'butt/fillet' weld was used for the design (Fig. 8b). It was found that the machining costs for the 'stub/fillet' weld type of joint were considerably cheaper than for the 'stub/butt' and thus additionally there was no possible weakness due to the cross-grain of the machined stub**.**

The tube to tubeplate weld has been recognised as the most critical item in the robustness of LMFR steam generators. It was concluded, that the type of direct tube-to-tubeplate weld (the butt/fillet weld, Fig. 8c) adopted at PFR, which could not be heat treated after manufacture, should be avoided in future SG [16].

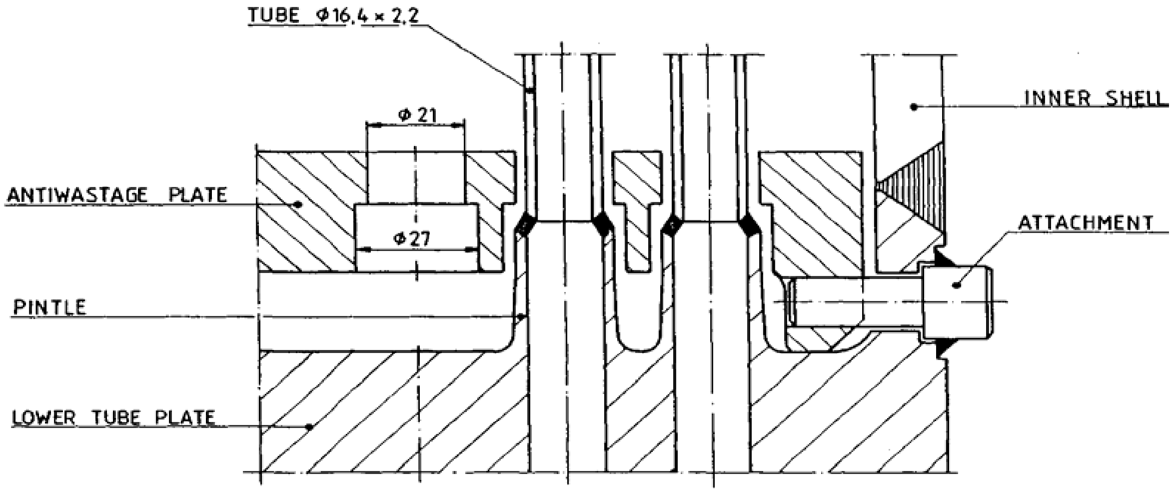


FIG. 9. A typical straight-tube steam generator tube/tubeplate weld joints (EFR) [17].

The design of sodium heated steam generators has been largely changed during the development of fast reactor technology. Studies were aimed at the creation of a reliable, low-cost design, which is easily inspected (diagnosed) in the process of operation after a steam generator unit switch-off. The experience gained during development and operation has shown that neither micro modular nor macro modular, neither double wall modular steam generator meets completely these criteria; they are complicated and metal-intensive. Consideration of capital cost suggests that the units should be few in number and as large as possible.

Upon experimental confirmation of the long-term resistance of steels such as mono-metallic modified 9Cr-lMo (Grade 91) steel in sodium, water and steam including wet steam, a possibility of assuring once-through process in the tube (water heating, boiling, evaporation, and steam superheating) has appeared. This result, along with sodium replacement with steam in the reheater, has made it possible to use a single-vessel steam generator, which was first applied at the Super-Phénix reactor. A steam generator unit of 750 MW(th) power, having welded coil tubes, operated reliably at Super-Phénix. However, it turned out to be comparatively expensive and complicated as far as the tube coil diagnosis under operating conditions (after a steam generator unit switch-off) is concerned. It was possible to eliminate these drawbacks in the single-vessel steam generator [600 MW(th)] with straight long tubes made of 9Cr-lMo (Grade 91) steel, which was designed for the EFR reactor.

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ABBREVIATIONS

AHX	Air heat exchanger
ALMR	Advanced liquid metal reactor
BN-350	Bystrie neytrony (Fast neutrons)
BN-600	Bystrie neytrony (Fast neutrons)
CDA	Core disruptive accident
CEFR	China experimental fast reactor
CR	Conversion ratio
CRC	Central rotating column
CRP	Central rotating plug
DFBR	Demonstration fast breeder reactor
DFR	Dounreay fast reactor
EBR-II	Experimental breeder reactor-II
EC	Emergency condenser
EFPD	Effective power day(s)
EFR	Experimental fast reactor
FA	Fuel assembly
FE	Fuel element
FEM	Finite elements method
FFTF	Fast flux test facility
FSD	Fuel storage drum
IFC	Irradiated fuel cave
IHX	Intermediate heat exchanger
IFR	Integral fast reactor
JSFR	Japan sodium-cooled fast reactor
KAEC	Kazakh atomic energy committee
KALIMER	Korean advanced liquid metal reactor
KNK-II	Kompakte Natriumgekuhlte Kernreaktoranlage
LMFR	Liquid metal cooled fast reactor
LRP	Leakage re-injection pump
MINATOM	Ministry of Atomic Energy
MPC	Maximum permissible concentration
NIV	Neutron-induced voidage
OD	Outside diameter
PIW	Post irradiation examination

PFR	Prototype fast reactor
RI	Reactor installation
RNL	Risley Nuclear Laboratories
SBB	Sodium boiler building
SG	Steam generator
PSP	Primary sodium pump
TC	Tight compartment
TD	Theoretical density
ULOF	Unprotected loss-of-flow
WCR	Water-chemical regime

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