IAEA-TECDOC-1180

Unusual occurrences during LMFR operation

Proceedings of a Technical Committee meeting held in Vienna, 9–13 November 1998

INTERNATIONAL ATOMIC ENERGY AGENCY IAEA

October 2000

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The originating Section of this publication in the IAEA was:

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

UNUSUAL OCCURENCES DURING LMFR OPERATION
IAEA, VIENNA, 2000
IAEA-TECDOC-1180
ISSN 1011–4289

Printed by the IAEA in Austria
October 2000

FOREWORD

Design of liquid metal cooled fast reactors (LMFRs) is still in evolution, and only a small number of LMFRs are in operation around the world. Specialists operating these LMFRs have gained valuable experience from incidents, failures, and other events that took place in the reactors. These unusual occurrences, lessons learned and measures to prevent recurrences are often either not reported in literature, or reported only briefly and without sufficient detail. Hence there is a need for specialists designing and operating LMFRs to share their knowledge on unusual occurrences.

Considerable experimental and theoretical knowledge on various aspects of LMFR design construction, pre-operation testing and operation has been collected by several Member States with fast reactor programmes over the past decades.

Recently, more countries have launched their programmes on fast reactors in critical and subcritical (driven by a spallation neutron source) mode cooled by liquid metal.

The needs in generalisation, review and documentation of fundamental knowledge in liquid metal cooled reactor technology were a major consideration in the recommendation by the International Working Group on Fast Reactors (IWGFR) for the IAEA to convene this Technical Committee meeting on the subject of unusual occurrences during LMFR operation and their consequences for reactor systems.

The IAEA officer responsible for this work was A.A. Rineiskii of the Division of Nuclear Power.

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SUMMARY

1. OVERVIEW AND PURPOSE

The Technical Committee Meeting (TCM) on "Unusual Occurrences During LMFR Operation: Review of Experience and Consequences for Reactor Systems" was held on the recommendation of the International Working Group on Fast Reactors (IWGFR) at the IAEA Headquarters in Vienna from 9 to 13 November 1998. Participants from nine countries (China, France, India, Japan, Kazakhstan, the Republic of Korea, the Russian Federation, the United Kingdom and the United States of America) were in attendance.

The objectives of the TCM were:

- to review design approaches and operational experience and to identify and characterise the main design and technical problems and relevant unusual occurrences during LMFR operation and the consequences for the reactor system;
- to review findings in advanced LMFR designs with a view to avoiding unusual occurrences; and
- to provide a forum for discussion and identification of pathways to take advantage of the opportunity for international co-operation in these activities

The discussion was focused on those aspects of LMFR technology which are unique and distinctive to plant design and operation.

2. MAJOR PROBLEMS AND THEIR CAUSES

LMFRs have been under development for more than 45 years. Twenty LMFRs have been constructed and operated. Five prototype and near-commercial scale LMFRs [(BN-350 (Kazakshstan), Phenix (France), PFR (United Kingdom), BN-600 (Russian Federation) and Superphenix (France)], with electrical output of between 250 and 1200MW(e), have accumulated more than 85 reactor years of operating experience. Altogether, LMFRs have accumulated more than 280 reactor years of operating experience. In many cases, experience with these reactors has been extremely good. The reactors and special components have shown remarkable performance well in excess of design expectations.

Stable operation of the demonstration reactor BN-600 in Russia with a nominal power output of 600MW(e) for 20 years and an average load factor of ~72%, successful operation of the prototype reactors BN-350 in Kazakshstan and Phenix in France as well as the reliable operation of MOX fuel at high burnup (20% with an irradiation dose in excess of 160 displacement per atom (dpa) in the cladding) in PFR (UK) and Phenix, are milestone in the implementation of LMFR technology.

As is usual at the initial stage of development, the operation of prototype and demonstration plants revealed the weak points of the original design and mode of operation, causing some unusual events. The meeting participants discussed these unusual occurrences as well as the steps taken to rectify them. In this way the benefit of the lessons learnt can be made available to the designers and operators of reactors of similar type.

2.1. Sodium leaks

There is only one disadvantage inherent in the LMFR coolant, liquid-sodium, namely that it interacts chemically with water/steam and air. Protection against sodium fires was an important theme of the discussion at the meeting.

There were significant sodium leaks and fires from the BN-600 primary (1000kg) and secondary (650kg) circuits in 1993 and 1994, respectively. In both cases the protective systems were effective, the damage was not extensive and repairs were effected quickly. Thermal striping was the cause of the leak in 1993 and staff error (pipeline cutting before sodium was frozen) of the leak in 1994. None of these leaks has caused injury to personnel.

A secondary sodium leak and fire at the Monju (Japan) plant in December 1995, caused by a temperature sensor well broken by vibration, caused a long operational delay and required plant modifications. The total mass of non-radioactive sodium which leaked was ~ 640± 42kg due to delay in sodium draining from the loop. There were no adverse effects for operating personnel or the surrounding environment.

To prevent a recurrence of the Monju secondary sodium leakage incident comprehensive design review activities were started for the purpose of checking the safety and reliability of the plant. As a result, several aspects requiring improvement were identified and improvements and countermeasures were studied. The main improvements and countermeasures are as follows:

- to enable the operators to understand and react to incidents quickly, new sodium leakage detectors (TV monitors, smoke sensors) and a new surveillance system will be installed;
- to reduce the amount of sodium leakage and damage by spilt sodium, the drain system will be remodelled to shorten the drain time;
- to extinguish a sodium fire in the secondary circuit, a nitrogen gas injection system will be installed;
- to limit the spread of aerosol, the secondary circuit will be divided into four smaller zones;
- to replace the secondary circuit thermocouple wells by a new design;
- to prepare a new design guide against flow-induced vibration.

The event at the Monju plant caused a serious interruption to reactor operation. For this reason there will be undoubtedly important efforts to reduce the incidence of sodium leaks and to improve the protection against the consequences of fires.

The number of leaks will be reduced by improvement of the design methods for sodium systems taking account of the peculiar conditions to which the materials are subjected, particularly high and fluctuating secondary stresses, high-cycle fatigue, and prolonged exposure to high temperatures which can cause degradation of the properties of some austenitic steels. Improved methods of estimating stresses and improved design codes appropriate to thin-walled vessels and pipework are to be expected. Thermal striping has been the cause of some sodium leaks in auxiliary circuits. Mixing devices should be provided to ensure that sodium streams mix with temperature differences below the established safe limits.

Additional protection will be afforded by improved methods of non-destructive in-service inspection to demonstrate the integrity of sodium vessels and pipework and to detect incipient defects so that they can be repaired before they grow to cause leaks. Ultrasonic methods for inspecting welds in austenitic steels, and transition welds between steels of different composition, have been improved and further improvement can be expected. Components will increasingly be designed to facilitate in-service inspection.

In addition there will be greater acceptance of the "leak-before-break" approach to protect against major failure and large leaks. This requires close attention to the reliable detection of small leaks at an early stage, before they can give rise to significant fire. There will also probably be developments in the field of protection against the effects of fires, by means of improved segregation and protection of essential equipment and services, faster sodium dump systems, better methods of extinguishing sodium fires, and enhanced protection from damage by sodium smoke.

Prevention, detection and mitigation of sodium leaks, improved resistance of nuclear systems to fires and choice of concrete for minimisation of interactions remain important directions for safety research. In France a new aluminous concrete which does not interact with sodium has been proposed. The EFR (European fast reactor) anchored safety vessel option was tested with this concrete. There is a need to continue the R&D on sodium-resistant concrete to minimise damage to structures in the event of sodium leaks.

2.2. Steam generator significant events

Steam generators (SGs) are generally regarded as the most critical of all sodium system components. Design, manufacture and experimental testing should be carried out with special case. It seems that all was done to install reliable SGs in prototype, demonstration and semi-commercial LMFRs. Three prototype fast reactors (BN-350, Phenix and PFR) were commissioned in the 1960s and two of them (BN-350 and PFR) had unforeseen occurrences with SG.

Problems in commissioning of BN-350 and PFR were almost entirely due to steam generators leaks. The effect of these leaks on BN-350 and particularly on PFR availability was considerable so that the highest annual load factor was only 10–20% for several years after commissioning. Leaks in the PFR SGs 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.

The tube-to-tubeplate weld leaks in the PFR evaporators necessitated the development of a repair method by fitting sleeves to bypass defective welds: 3000 sleeves were installed in each end of the 500 tubes of each of the three evaporators. The success of the method was demonstrated by the fact that no further leaks have occurred.

It was concluded that the type of direct tube-to-tube plate weld adopted initially at PFR, which could not be heat-treated after manufacture, should be avoided in future reactors. The UK specialists consider that austenitic steels are unsuitable for LMFR steam generators because of the high risk of caustic stress corrosion damage following even small leaks.

The initial period of BN-350 reactor plant operation was characterised by unreliable operation of the SGs. Numerous leaks occurred in the tubes of the evaporators. Metallographic examination of a great number of tubes showed the presence of microcracks in the tube-to-bottom weld joints. Mechanical deformation of the tube bottoms during manufacture by cold stamping is the most probable cause of the microcracks. Growth of the cracks could occur under the effect of internal stresses arising during welding the bottoms to the tubes and under cyclic thermal loads during operation. The evaporator tube bundles were replaced with machined bottoms. No further problems were experienced with the evaporators.

The early problems with the SG highlighted a need to improve non-destructive methods for detecting small flaws in relatively inaccessible welds. A major programme of development work led to major refinements, particularly in ultrasonic techniques.

The under-sodium leak in a PFR superheater demonstrated that it is possible for a large number of tubes to fail due to overheating in a few seconds, but as was pointed out at the meeting such an event is unlikely to cause significant overpressurisation damage in the secondary circuit or the intermediate heat exchanger. As to the modification of the PFR sodium-water protection system and replacement of the tube bundle, 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 rupture to 40 double-ended guillotine raptures spread over a period of 10 s.

As stated above, in the early 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 can be done without significant reducing 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. 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. Since late 1981, the SGs have been operated at the nominal power and parameters. Some evaporator modules have been replaced by new ones due to expiry of the projected lifetime.

SG operation experience shows that the vast majority of leaks occurred where the tubes are attached to the tubeplate and seldom in tube-to-tube welds. Therefore specific attention in advanced SG design is being paid to decrease the number of the tube-to-tubeplate welds, as well as avoiding welds under sodium in the length of the tubes. Progress in this direction can be achieved by minimising the number of separate units, e.g. by locating the evaporator and superheater in one unit, simplifying the SG configuration by replacing sodium-heated reheaters by steam-heated, and using long tubes.

To ensure satisfactory SG performance there must be an adequate industrial base for the structural materials (which have to be compatible with both sodium and water/steam at high temperature) and for manufacture, and a sensitive system for detection and mitigation of steam/water leaks in sodium.

Once-through SGs with high unit power have to be made of high-nickel alloy 800 or of an advanced ferritic steel 9Cr1MoVNb development of which started in the USA in 1980. To minimise or prevent the carbon dissolution from ferritic steels and to improve creep properties stabilising additives such as niobium and vanadium are incorporated. During the last two decades excellent results have been achieved in the technology for ensuring high quality of manufacture of steam generators and studies in realistic conditions of the problems of thermohydraulics, structural mechanics, etc. have ensured the reliable operation of large once-through SGs. Successful operation of the four Superphenix SGs, each with a power of 750MW(th) and with few (357) long (~95m) welded helical tubes, holds the key to reliable and compact SGs for future LMFRs.

Cost reduction and reliability improvement studies have been performed to select the steam generator concept for EFR. An integral unit of 600MW(th) has been verified as favourable based on successful operating feedback from the SPX SGs. The advanced SG for EFR is a once-through straight-tube unit, without welds in the long (33,3m) tubes with tube to the tubeplates weld at each end.

The EFR SG project is widely regarded as successful in matching the designers' intentions of simultaneously improving the reliability and economics by decreasing the number of tube-to-tubeplate welds, the structural material content and the SG building volume. This can be confirmed by comparison of the major characteristics of the BN-600 sectional-modular SGs, with three units each consisting of evaporator(ev) + superheater(sp) + reheater, with the EFR advanced integral SGs, with (ev + sp) in one unit:

Reactor Plant	SG power MW(th)	mass of SG. tons	Max. tube length (m)	Number of tubes in SG	Specific metal content tons/MW(th)	Number of tubes to tubeplate attachments (welds) per 1 MW(th)
BN-600	490	600	16	6592	1, 22	27, 0
EFR	600	150	33, 3	1386	0, 25	2,3

For the complete realisation of potentially feasible LMFRs from the point of view of cost, metal content and reliability new design solutions for SGs and other equipment may be required, differing from those being used at the present time.

2.3. Pump oil leaks into the sodium circuit

Oil ingress into the primary sodium circuits of an LMFR is undesirable because of the potential release of methane gas through the reactor core, causing positive reactivity effects, and possible blockage of the fuel subassemblies by solid carbon debris. All large LMFR sodium pumps have oil-lubricated bearing and seals at the upper end of the shaft (above the sodium level in the pump casing). There have been several oil leaks into sodium due to non-optimal pump design and/or operator errors. A major oil leak (possible up to 35 litres) into the primary sodium circuit took place in PFR. This was a result of a blocked overflow pipe causing a pressure differential between the gas blanket in the reactor vessels and the pump casing. Work to recover from the oil ingress problem, most particularly work to install new primary pump filters, occupied almost a whole year. The length of the outage indicates the importance of designing to avoid any possibility of spillage of oil from pump bearings and seals into the sodium coolant 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.

The EFR design was changed following the PFR oil ingress incident by the introduction of the innovative features of magnetic bearings and ferro-fluidic seals to eliminate oil completely and remove the potential hazard of its ingress into the sodium.

2.4. Air ingress into the argon of the Superphenix reactor cover gas.

A partial tear of a compressor membrane in a cover gas activity measurement line resulted in the admission of an argon and air mixture into the reactor, which was at nominal power. The total pollution was estimated at between 300 and 350kg of sodium oxides. There was no system to measure the purity of the reactor sodium argon cover gas to provide an alarm for the operator in case of pollution, and the reactor operated outside the operation specification range for three days before shutdown. An unplanned reactor outage of over 10 months was needed to remove the impurities from the primary sodium.

The fundamental causes of the incident were identified as:

- incomplete analysis during design of the parameters, including gas pressures, in the argon gas circuit, and lack of assessment of the risks and prevention of air ingress and leaks in general, particularly with respect to the activity measurements circulation pumps;
- absence of primary argon on-line monitoring devices in spite of such a system having been installed in Phenix, PFR, KNK, SNR 300 and other plants;
- incomplete referencing of equipment and preventive maintenance programmes;
- incorrect interpretation of the plugging indicator recordings;
- inappropriate operating specifications both in terms of clarity and applicability.

Further actions were taken to draw all the relevant information from the incident and to implement the resulting measures before the unit was restarted.

2.5. Secondary sodium leaks in Phenix IHX

Several sodium leaks from the heads of the intermediate heat exchangers (IHXs) took place during reactor operation. The damage was due to a design error leading to deformations and stresses because of differential expansion of the external and internal rings at the secondary sodium exit. The cause was found to be incomplete mixing of the secondary sodium at the outlet of the tube bundle, resulting in a significant radial temperature gradient between the outer and the inner sodium flows, the former being hotter. As a result, mechanical constraints led to overstressing of welds in parts of the secondary sodium duct.

Serious loss of generation and increased expenses were caused by the need to repair all the IHXs and then replace some of them. The removal, washing and decontamination of an IHX was a very difficult maintenance task. It was pointed out by the TCM participants that IHX operational experience, except for Phenix, is excellent.

2.6. Spent fuel assemblies handling system

Spent fuel assemblies from all existing prototype and demonstration LMFRs are taken out of the reactor and transferred to an intermediate fuel storage drum cooled by sodium. They remain there until their decay heat has sufficiently decreased for further cleaning and transport. The drum plays the role of a storage buffer between the reactor and the water storage pool: it also allows loading and unloading times, and hence the duration of reactor outages, to be reduced to a minimum.

The fuel assembly storage drum is generally regarded as the most critical of all reactor auxiliary system components. Highly reliable cooling systems are needed to remove heat from the irradiated fuel assemblies. All technical means of attaining this ambitious goal have to be included into the design and thoroughly evaluated. Diversity, degree of independence from energy supply (passivity) and simplicity have to be taken into account to prevent accidental disruption of the irradiated fuel assemblies. Two (BN-350 and Superphenix) of the six commissioned prototype and demonstration fast reactors have had difficulties in the spent fuel assembly handling systems due to failure of the storage drum at the beginning of operation. These failures are believed to have been due to less than optimal structural material specification.

The leak in the Superphenix fuel storage vessel had a great impact in suspending plant operation for 20 months, although the reactor itself was unaffected. Investigation indicated that cracking in the inner vessel wall, made of 15Mo3 ferritic steel, was caused by embrittlement of welds due to a combination of stress and a hydrogen-rich environment prior to power raising. The ferritic steel sodium storage tanks in SNR-300 fabricated in a similar material suffered weld cracks which were also attributed to hydrogen embrittlement.

It was concluded that the use of ferritic steels of this type for structures in permanent contact with sodium should be banned for the future LMFR, austenitic steels being preferred.

The problem of spent fuel assembly handling at two reactors (BN-350 and Superphenix) posed by the occurrence of leaks was settled by abandoning the decay storage function of the drum (this function being taken by the reactor itself) and maintaining only the function of sub-assembly transfer. It compelled the operator to wait for the decay power to subside before unloading the core, resulting in long outages.

The consequence of these incidents for future LMFR projects was the investigation of spent fuel handling technology aimed at eliminating the intermediate storage drum and transferring spent fuel assemblies directly from the reactor to the water storage pool. This has been realised in the large advanced fast reactor projects BN-1600M and EFR. Incorporation of two or three rows of B4C-filled assemblies between the core and an in-reactor irradiated fuel store allows the residual power of fuel assemblies to be unloaded to be reduced.

The loading and unloading operations can be combined with shutdown of the NPP for maintenance and repair, so that elimination of the storage drum causes insignificant additional plant outage. For the above mentioned reactors refuelling takes place during scheduled reactor shutdowns which occur approximately annually. Fuel subassemblies which have been retained in the in-vessel store during the previous reactor run, and therefore have a lower decay heat rating, are transferred directly from the reactor to the water storage pool.

2.7. Behaviour of some austenitic steels with sodium under load and high temperature

A circumferencial crack was detected by beaded wires on a moulded 321 steel tee junction in the Phenix secondary sodium circuit after 90,000h of operation. The crack was located in the heat-affected zone of the weld between the tee and the adjacent piping. It was shown that the crack occurred because of high stresses induced by an important misalignment.

On three occasions in 1987, 1988 and 1990, small leaks occurred through cracks in the circumferential welds in the PFR reheater vessels, which were manufactured in type 321 stainless steel.

Stainless steel type 321 with addition of titanium has been used to enhance the mechanical performance of the structural material at high temperature (520–550 C). The intensive French programme of non-destructive tests since 1989 has shown that 321 stainless steel has not performed satisfactorily over time, with cracks in welded joints occurring under load and high temperature. The sections of Phenix secondary pipework made in this steel have been replaced by 316SPH steel.

2.8. Influence of low dose irradiation on the reactor fixed internals.

Studies carried out in some Member States showed that even low neutron irradiation doses (of about 2 displacements per atom) has significant effects on near-core structural material at low and high temperature. It was observed that near core structures — permanent for 40–50 years of service — have exhibited degradation due to the neutron environment.

For components operating at the lower temperature range, the main cause of concern is degradation of fracture toughness due to atomic displacements. For components operating at higher temperature range, the main cause of concern is grain boundary embrittlement, due to the formation of helium or irradiation-induced segregation resulting in a reduction in creep and creep-fatigue resistance. This degradation in properties requires both an optimisation of materials for life extension consideration and quantitative description for the detailed design. As a result of the latter work the specification of structural materials has been modified.

Both in Europe and Japan low carbon stainless steel of type 316L with controlled addition of nitrogen is recommended for use even at low temperature. At doses above about 15 displacements per atom the degree of embrittlement and the level of swelling in all materials tested are probably unacceptable for use in critical fixed reactor components.

3. CONCLUSIONS

The accumulated knowledge on materials, thermohydraulics and mechanical science indicates that a substantial decrease in investment costs, together with better assurance that safety margins are effectively maintained, might be derived from a lesser emphasis on "umbrella" transients, compensated by an accurate analysis of actual transients and operating experience.

Perhaps there is only one disadvantage inherent in the LMFR coolant-liquid sodium, namely that it interacts chemically with air and water/steam. Therefore, providing integrity of the sodium circuits is the most important requirement to observe on the LMFR design, construction and operation tests.

The solution of the problem of the reliable elimination of coolant leaks is determined by the application of experimentally and analytically proved design, structural materials, manufacture and installation, as well as quality control at all stages of LMFR components manufacture. Technological procedures and approaches, as well as quality criteria, should strictly correspond to the related regulatory documents.

Efficient criteria and rules and effective LMFR technology, as experience has shown, can be established by design, construction and comprehensive testing of three or four LMFR plants. This stage has been reached in some countries. For example, in Russia an experimental reactor BR-10, an experimental 15MW(e) NPP BOR-60, a prototype NPP BN-350 (presently in Kazakhstan) and a semi-commercial NPP BN-600 have been operating respectively for 40, 30, 25 and 20 years, providing invaluable information on FR technology. That is why the BN-600 plant has been running successfully for 20 years with an overall lifetime load factor of ~72%. This success was achieved because in the design and construction of the plant, manufacture of the equipment and operation of the plant past errors were not repeated and good design solutions were incorporated.

The comprehensive operational experience with LMFRs BN-350, Phenix, PFR, BN-600, Superphenix and Monju has shown that, if plant components have been designed and manufactured without errors and representative specimens or models have been tested prior to installation, reliable operation can be ensured during the whole operational life.

The very low corrosion of sodium, the near atmospheric operating pressure, the use of ductile structural materials, and the reliable heat removal by a coolant having no phase change, imply that there should be nothing to provoke loss of the sodium system integrity in a LMFR.

PROBLEMS EXPERIENCED DURING OPERATION OF THE PROTOTYPE FAST REACTOR, DOUNREAY, 1974–1994

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Abstract

The UK Prototype Fast Reactor, PFR, was designed in the 1960s and was operated at Dounreay in Scotland from 1974 to 1994. By the time it was shut down it had demonstrated the feasibility of the technology of a large sodium-cooled fast breeder reactor, and had been shown to operate safely and reliably. It had also provided an invaluable test facility for advancing the technology, particularly in developing advanced fuel and cladding materials that had achieved high burnup and neutron dose.

As is usual in prototype plants the operation of PFR revealed the weak points of the original design concept. Several difficulties were encountered in the course of its operating life, all of which were successfully overcome. The purpose of this paper is to describe some of these difficulties and the steps taken to master them. In this way the benefit of the experience gained and the lessons learnt can be made available to the designers and operators of reactors of similar type. The intention is that future generations will not follow false trails in the further development of this promising technology.

INTRODUCTION

Nine major incidents and unforeseen developments are described. They are

1. A series of steam-generator gas-space leaks,
2. A major under-sodium leak in a steam generator,
3. The incidence of cracking in the steel of various secondary sodium circuit components,
4. Blockage of the secondary sodium cold trap,
5. Seizure of the primary sodium cold trap pump,
6. The effect of sodium aerosol deposits of the operation of primary circuit components,
7. Malfunctioning and cracking in the air heat exchangers of the decay heat rejection loops,
8. Neutron-induced distortion of core components and its effect on plant operation, and
9. A major oil leak into the primary circuit.

Each of these is summarised below, with diagrams. The causes and the steps taken to rectify the problem are explained, and the general lessons learnt are set out in the context of the future development of LMFR technology. References are given in the cases for which more detailed information has been published.

1. PFR STEAM GENERATOR GAS SPACE LEAKS

PFR had three secondary circuits, each of which had an evaporator, a superheater and a reheater. Figure 1.1 shows the general arrangement, and Figure 1.2 shows an evaporator in more detail. A total of 37 gas-space leaks was experienced in PFR steam generator units in the period 1974 to 1984 with 33 of these occurring in evaporators, 3 in superheaters and 1 in a reheater. All of the gas-space leaks originated at the welds between the tubes and the tubeplates. The effect of these leaks on PFR availability was considerable, so that the highest annual load factor prior to 1984 was only 12%.

PFR went critical for the first time in March 1974 and commissioning of the steam generators followed. Up to 1976 there were failures in gas-space leaks in one evaporator, two superheaters and one reheater. These early failures are believed to have been due to manufacturing faults.

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

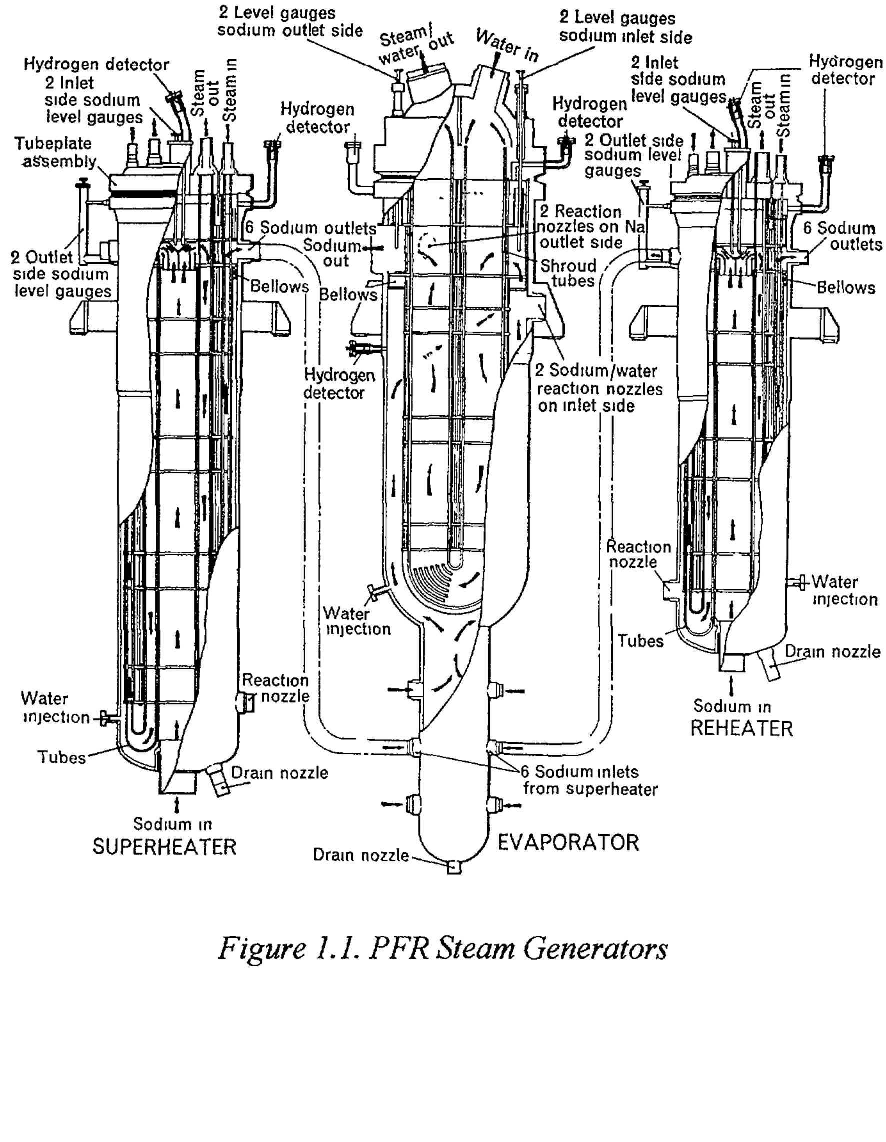

Figure 1.1. PFR Steam Generators

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.

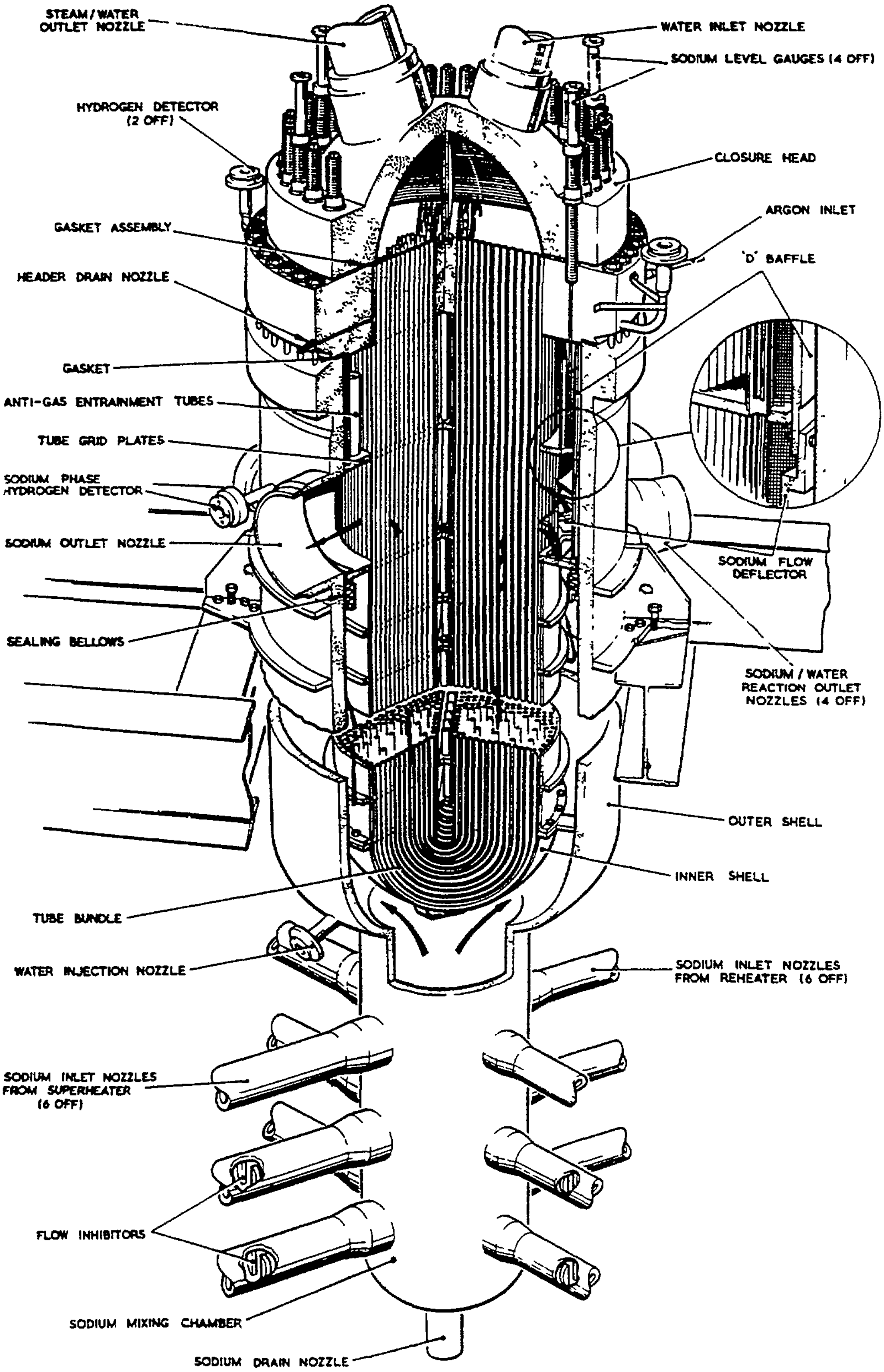

Figure 1.2. A PFR Evaporator

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, shown in Figure 1.3. The design benefited from the early experience of caustic stress corrosion following the leaks in the austenitic units. The new tube bundles were fabricated in 9Cr 1Mo ferritic steel, and six were available to replace the original units by 1984. The replacement work was completed by 1987.

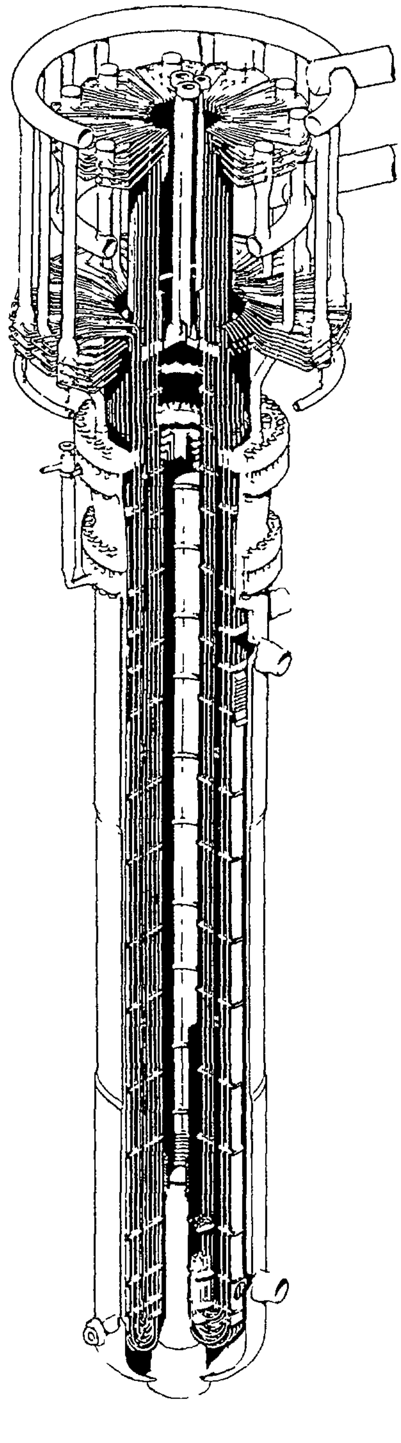

Figure 1.3. A PFR Replacement Reheater Tube Bundle

The 33 leaks experienced in the ferritic steel evaporator units (Figure 1.2) 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.

The problem was finally solved by the fitting of sleeves which spanned the original welds, as shown in Figure 1.4. In all 3000 sleeves were fitted over a 14-month period. The work was completed early in 1984. While sleeving was underway the station operated on a single circuit. Following sleeving no further problems were experienced with the evaporators.

Conclusions

Gas space leaks in PFR steam generators provided valuable information on the behaviour and detection of such leaks. They proved to be readily detectable by means of the hydrogen generated. Careful washing of the tubeplates with hot sodium limited the numbers of leaks and avoided major further plant damage, but did not cure the problem. Eventually a radical solution, involving sleeving all the tub-to-tubeplate weds in the three evaporators, had to be adopted.

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

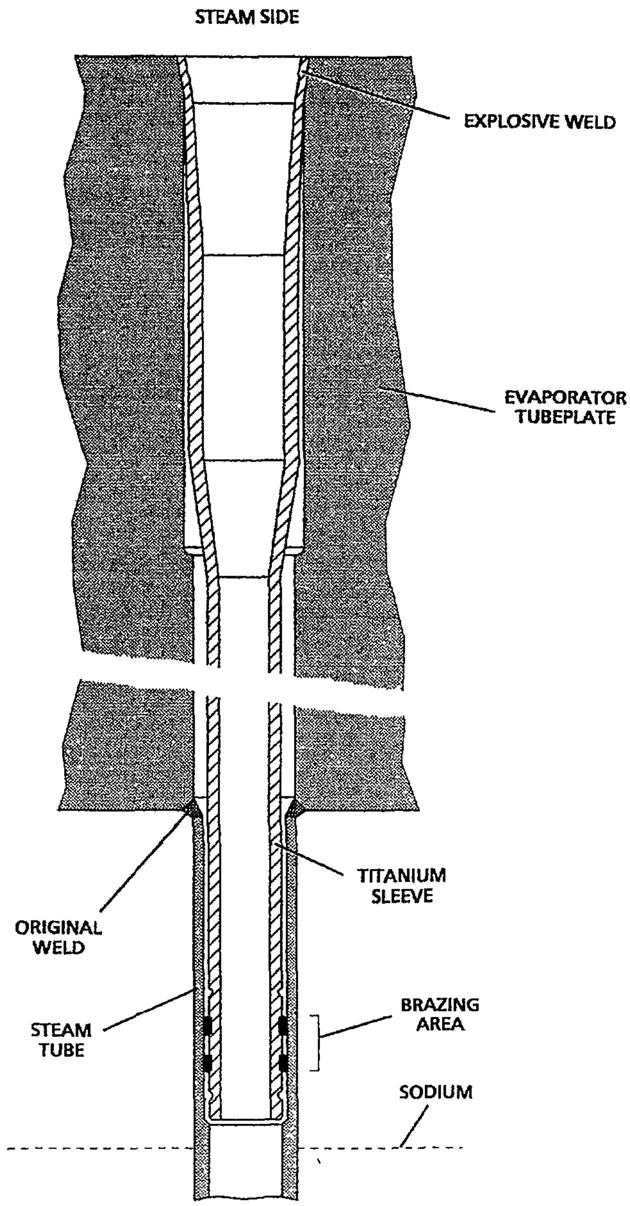

Figure 1.4. A PFR Evaporator Weld Repair Sleeve

REFERENCES TO SECTION 1

E R Adam and C V G Gregory "A Brief History of the Operation of the Prototype Fast Reactor at Dounreay": *The Nuclear Engineer*, 1994, **35**, 112 - 117

2. THE UNDER-SODIUM LEAK IN PFR SUPERHEATER 2

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

On 27 February 1987 PFR was operating 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 seconds, as designed.

It was confirmed shortly after the incident that a large under-sodium leak had occurred in superheater 2. Figure 2.1 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 tube 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.

The entire tube bundle was then dismantled and forty steam tubes were found to have ruptured, with longitudinal gapes of such a size as to be effectively equivalent to double-ended guillotine breaks. The locations of the failed tubes are shown in Figure 2.2. Sixteen of the failures were in the row adjacent to the central sodium duct and faced towards the duct. Four of the failed tubes had wear flats facing the central baffle and on one of these (tube 16) a circumferential crack at right angles to the main fracture was apparent. This was opposite a small wastage pit in the baffle, and it is concluded that this was the initiating leak.

Evidence of fretting on a further 13 tubes adjacent to the baffle was found, all close to the seams between the six plates forming the baffle. It was concluded that sodium leakage through the seams, shown in Figure 2.3, caused flow-induced vibration of the tubes resulting in contact with the duct and giving rise to wear and the eventual failure of tube 6.

Subsequent analysis of the event gave rise to the following conclusions:

1. A total of 40 tubes failed, 39 due to overheating in a period of 8 seconds following the plant trip. The remaining tube, which initiated the event, failed due to fretting damage caused by tube vibration.

2. The primary small leak grew after passing steam for some tens of seconds. Finally it grew rapidly to give a leak rate of 0.5 to 1.0 kg/s for a period of a few seconds.

3. This induced a plant trip by rupturing a steam-side bursting disc. Isolating valve closed. causing the steam flow to stop, but the steam pressure fell relatively slowly over a period of about 10 seconds.

4. Other tubes already weakened by fretting failed quickly, within a second or so, causing an increase in the heat generated by the sodium-water reaction.

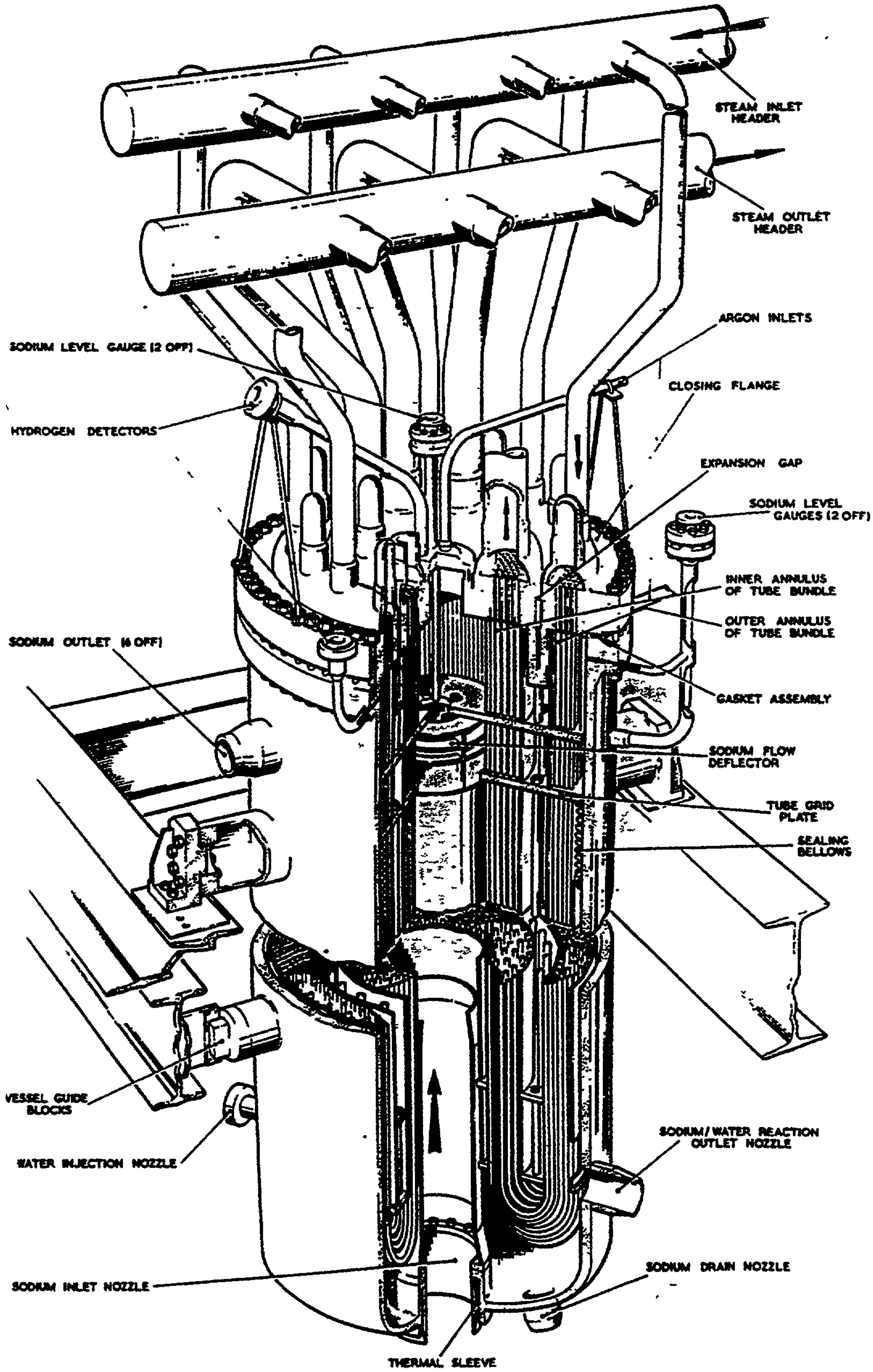

Figure 2.1 An Original PFR Superheater

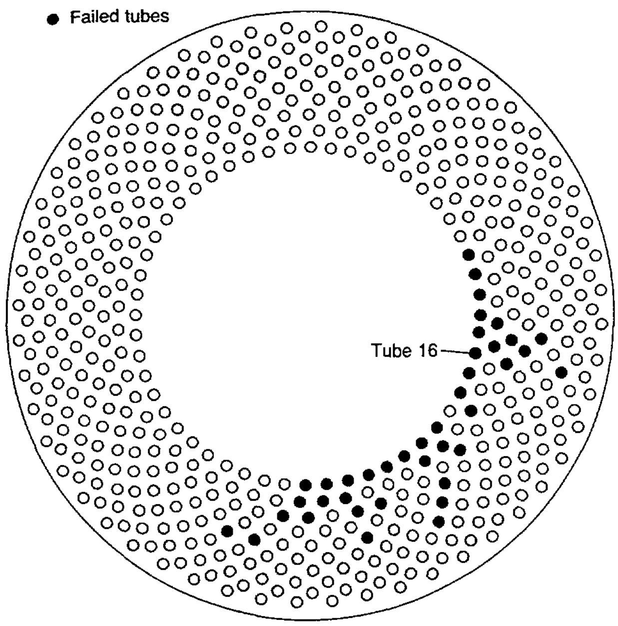

Figure 2.2 The Failed Tubes in PFR Superheater 2

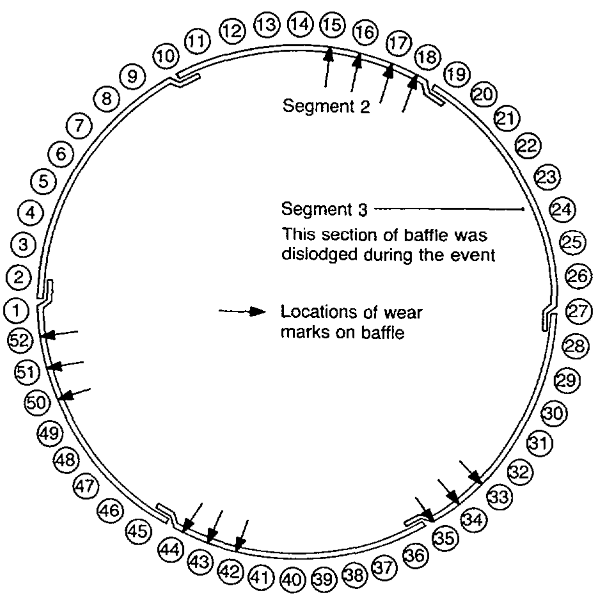

Figure 2.3 The Location of Fretting Marks on the PR Superheater 2 Central Baffle

5. The loss of internal cooling when the steam flow stopped and external heating by the sodium-water reaction made tube wall temperatures rise.

6. Reaction zone temperature increased above the boiling point of sodium. This caused high temperature tube failures at 1 325°C to 1345°C, even though the pressure in the tubes had fallen from 130 bar to 70 - 40 bar by this time.

7. As temperatures increased further more tubes would have failed but eventually the steam pressure dropped low enough to prevent further failures and tube swelling.

8. In spite of the large number of tube failures the pressure transient in the secondary sodium circuit was relatively mild. The maximum pressure in the intermediate heat exchanger did not exceed about 10 bars, well below its design pressure.

Conclusion

The under-sodium leak in superheater 2 demonstrated that it is possible for a large number of tubes to fail due to overheating in a period of a few seconds, but that such an event is unlikely to cause significant overpressurisation damage in the secondary circuit or the intermediate heat exchanger. As well as leading to modification of the PFR sodium-water protection system and replacement of the tube bundle, 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 seconds.

REFERENCES TO SECTION 2

A M Judd, R Currie, G A B Linekar and J D C Henderson: "The Under-Sodium Leak in the PFR Superheater 2, February 1987": *Nuclear Energy*, 1992, **31**, 221 - 230

3. CRACKS IN PFR STEAM GENERATOR VESSELS

From 1983 until its shutdown in 1994 PFR experienced cracking in type 321 stainless steel components in its secondary circuits, some cracks leading to sodium leaks. As a result a substantial repair and inspection programme was required in the final seven years of PFR operation. Although the two earliest leaks were in pipework (in 1983 and 1986) the majority were in steam generator vessels. The pipework leaks were only retrospectively identified as being caused by the same mechanism.

Figure 3.1 shows a typical steam generator vessel. The vessels were manufactured from cold rolled annealed type 321 (18Cr 10Ni 1Ti) austenitic steel plate. Three cylindrical courses, together with a flanged upper end and a domed lower end, were welded using type 347 (19Cr 9Ni 1Nb) weld metal. Manual metallic arc weld root runs with submerged arc weld fill were used. During manufacture the welds were inspected and in some cases inclusions and other defects were found. These were ground out and made good with additional rectification welds. None of the welds was stress-relieved. The vessels were considerably overdesigned to withstand a continuous pressure of 34.5 bar, although the operating pressure was only 2 bar.

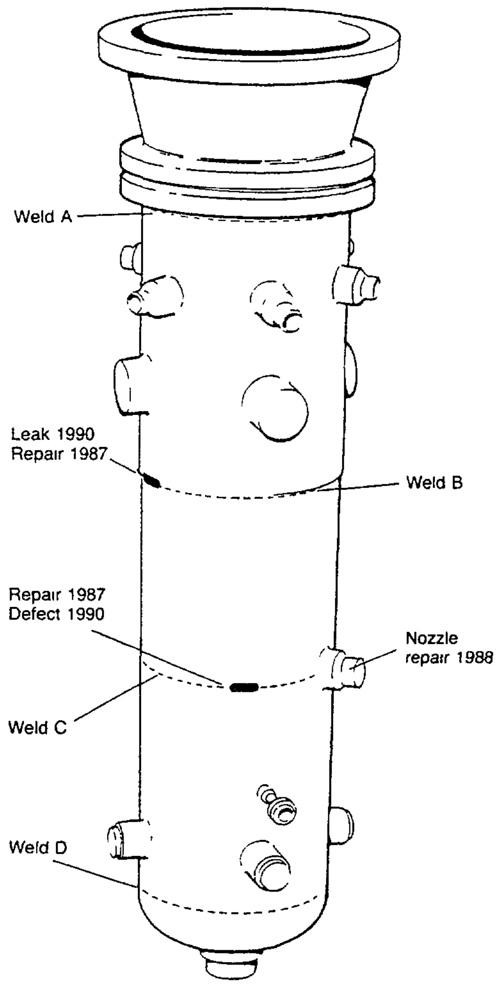

Figure 3.1. PFR Reheater 1 Vessel

The first crack occurred on a reheater pipework tee junction in 1983, and the second on a superheater pipework reducer in 1986. Both caused minor sodium leaks. The cause of the cracks was not identified at the time. Both leaks occurred at welds in which the large crystal grain size indicated severe overheating during fabrication.

In 1987 the first steam generator vessel leak occurred in seam B of the reheater 2 vessel. Following this cracks were found on vessels during every inspection except 1989. All the cracks occurred in the B, C and D welds with none in A. Seam A is in the gas space, above the sodium level, and therefore at much lower temperatures than the other seams. (Following the fitting of the new tube bundles in 1987 weld A also operated under sodium.) At the final inspection in 1993 cracking was found for the first time in the welds of set-on features such as the vessel supports and nozzles.

Detailed optical and scanning electron microscopic examination of samples cut from reheaters 1 and 2 and superheater 2 were carried out in 1987. In all cases the cracks were associated with deep weld rectifications in the circumferential weld seams. The cracks either passed along the fusion line of the rectification or were on or close to the fusion line of the original rectified weld. Cracks passed through the parent steel along or closely parallel to the fusion line and through the weld material. Cracking was intergranular.

The cracking mechanism was identified as delayed-reheat or stress-relief cracking. Titanium stabilised type 321 steel is subject to this form of cracking, caused by dissolution of titanium carbide close to a weld fusion line as a result of the high temperatures attained during welding. Subsequent re-precipitation of the carbides on dislocations produced by weld shrinkage during high temperature operation locks the dislocations and hardens the matrix. Relaxation strains occurring during operation causes intergranular failure.

In total some 27 cracks were detected on such welds during the period 1987 to 1993. Inspection was by ultrasonics and dye penetrant supported by acid etching of the surface. In general there was no sign of major escalation of the weld cracking problem during the period of monitoring. It is possible that cracking would have tailed off as the population of vulnerable sites diminished.

Repairs were made by removing the cracked region and welding on a stub nozzle with a blanked end. The technique is illustrated in Figure 3.2. It was chosen because it allowed all the work to be done from outside the vessel (so that it was not necessary to open the vessel to the atmosphere, which would have required scrupulous removal of all the sodium residues), and because it facilitated thorough ultrasonic inspection of the repair. Smaller defects were either ground out or backfilled and fitted with strain gauges for monitoring during operation. The PFR vessels were considerably overdesigned and although the excessive vessel thickness may have contributed to the cracking it was helpful when it was necessary to grind out defects.

Conclusion

The evidence indicated that cracking in PFR steam generator vessels was initiated by a delayed reheat mechanism driven by residual stresses in the non-stress-relieved welds. Weld rectification during manufacture gave rise to conditions which favoured cracking. It is

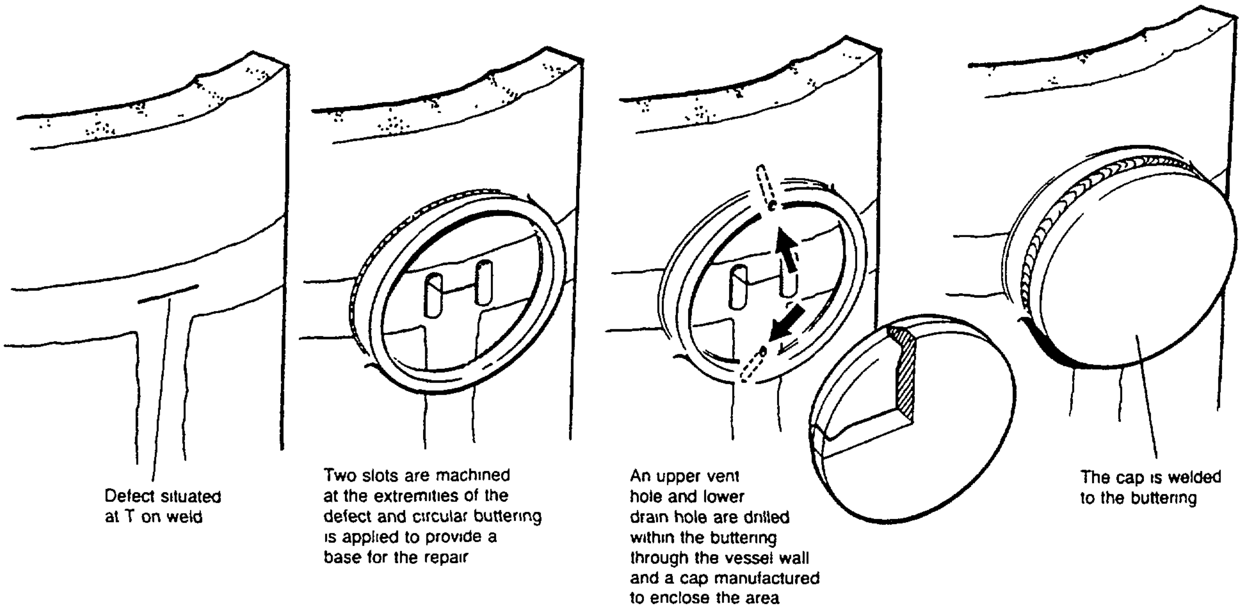

Figure 3.2. The Repair Technique for Cracks in the PFR Steam Generator Vessels

probable that replacement of the PFR steam generator vessels would have become necessary had operations been planned beyond 1994, as the repairs did not prove to be entirely satisfactory, with the repair welds beginning to develop cracks after a period of operation.

REFERENCES TO SECTION 3

D B Melhuish and A Sandison: "Engineering Improvements to PFR": *Nuclear Energy*, 1992, **31**, 193 - 205

4. BLOCKAGE OF THE PFR SECONDARY COLD TRAP

The PFR secondary cold trap vessel and basket are shown in Figure 4.1. Secondary sodium, partially cooled in a regenerative heat exchanger, was delivered to an annular chamber. From there it was injected through six 22 mm diameter holes into an annular space between the vessel and a mesh basket. The vessel was air-cooled and the intent was that the sodium should be cooled to about 20 °C below its current impurity saturation temperature before flowing through six annular "doughnuts" of steel wire mesh. The mesh presented a large surface area for the deposition of sodium oxide and hydride. Cold tap baskets were regularly removed and cleaned for re-use, fitted with new set of doughnut meshes, when a gradual reduction in the sodium flow as the mesh blocked up indicated that the maximum loading had been reached. Expected loadings were in excess of 100 kg of mixed sodium hydride and oxide.

The sixth basket was installed in July 1980 and had completely blocked by May 1981 with an estimated loading of only 52.3 kg of mixed oxide and hydride. Since this was not the first example of erratic cold trap behaviour it was decided to remove the basket for a special examination to ascertain the reasons for early blockage. Photographic records had been kept of earlier removals but until then no systematic detailed examination of the distribution of the deposits in the mesh had been made.

The deposits are pyrophoric and present a hazard that prevented close examination in air. On this occasion the trap was removed and kept under argon purge while an introscope examination was carried out. This gave an initial picture of the nature of the deposit before it was disturbed by being broken up.

The results of the examinations are shown in Figure 4.2. The deposits in the mesh were mixtures of sodium, sodium oxide and sodium hydride of varying composition, while the deposits on the sides and bottom of the vessel consisting virtually entirely of residual sodium. Detailed examination of the mesh doughnuts showed that the deposits were largely confined to the inner part of the annulus and the bottom, which was completely blocked.

Subsequent analysis suggested that the cold trap had not always been operated in the most efficient manner. Attempts to trap impurities too rapidly had led to the use of too large a differential between the actual sodium temperature and the impurity saturation temperature. The overcooled sodium precipitated its impurity burden preferentially at the bottom of the vessel and on the outer surfaces of the mesh doughnuts, so that the trap rapidly became blocked with an impurity loading well below the expected maximum. Modified operating techniques and led to improved loading and the final cold trap basket trapped in excess of 140 kg of mixed oxide and hydride.

Conclusion

Inefficient operation of the PFR secondary cold trap led to impurity loadings below the theoretical maximum. Improved trapping techniques aimed at avoiding cooling the sodium too far below its saturation temperature, and more careful operation solved the problem after 1981.

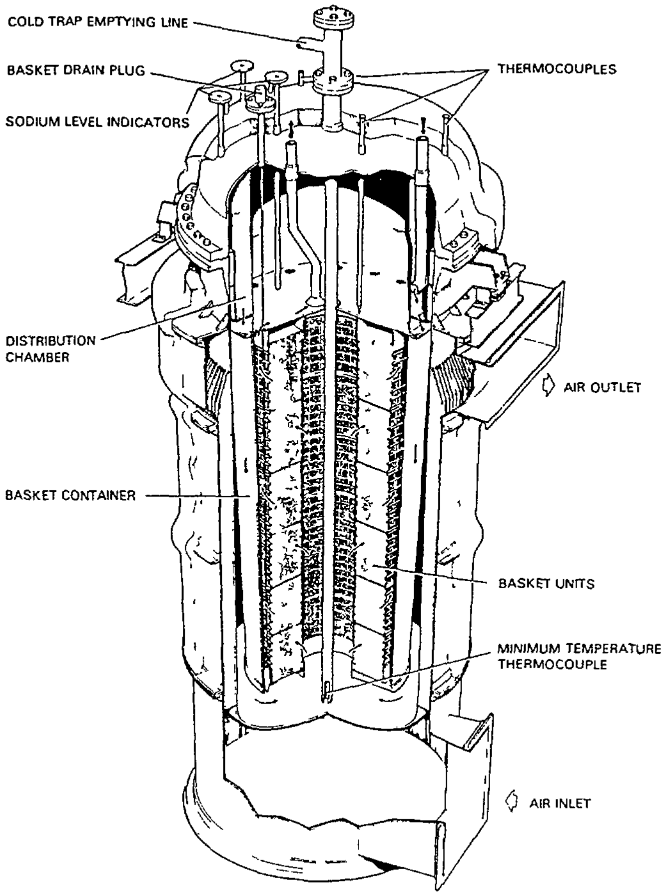

Figure 4.1. The PFR Secondary Sodium Cold Trap

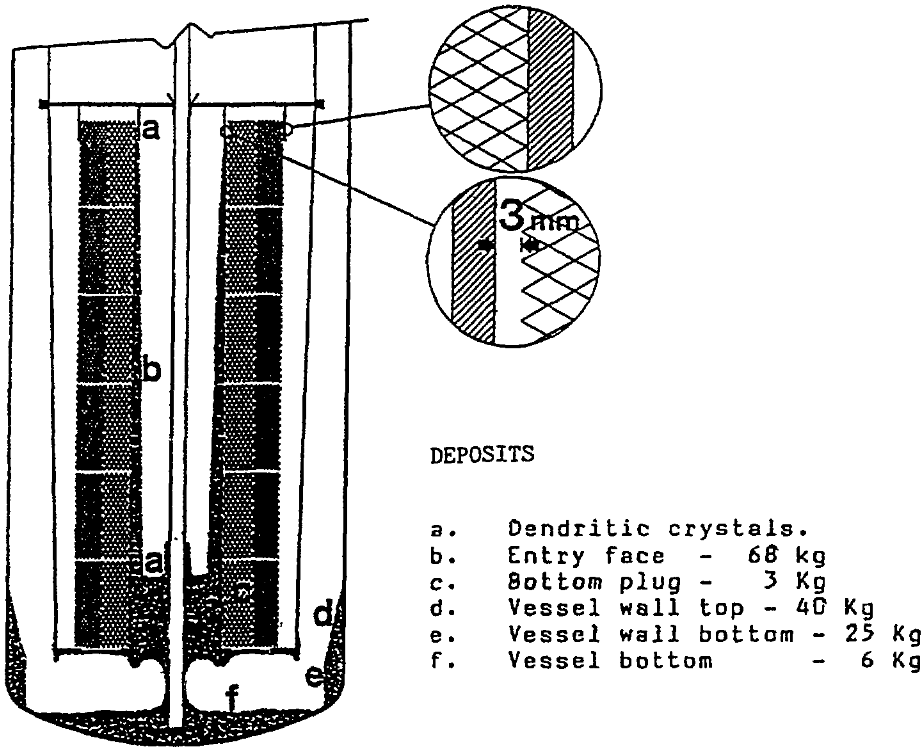

Figure 4.2. The Location of Deposits in the PFR Secondary Cold Trap

5. SEIZURE OF THE PFR PRIMARY COLD TRAP LOOP SODIUM PUMPS

The PFR primary cold trap loop (PCTL), shown in Figure 5.1, is an auxiliary circuit connected to the primary circuit. It purifies the active sodium by removing impurities. During PFR operation it also supplied sodium coolant to the core melt-out tray situated below the diagrid in the reactor vessel. During the current decommissioning period it remains in use for cold trapping, if necessary, and measuring impurity levels in the primary sodium. The PCTL is situated in a shielded air-cooled concrete vault adjacent to the reactor vessel, as shown in Figure 5.2.

Throughout the life of the reactor both the main and standby PCTL pumps have seized frequently due to sodium rising too high in the pump vessel. The pump vessel sodium levels are controlled by venting gas from the space above the sodium or injecting gas into it. In principle the sodium level was raised or lowered in 10 mm steps by operation of the gas valves in an automatic sequence. In practice, however, it was often necessary to intervene manually because the gas valves often passed or were blocked. In these circumstances operator error could easily lead to the sodium level into the annulus round the pump shafts where it would solidify causing the pump to seize.

Each time a pump seized attempts were made to remove the sodium mechanically or melt it by use of the trace heating, but on no occasion did this succeed. Pumps had to be removed, decontaminated and stripped down in order to free the pump shaft.

During maintenance it was noted that oxide or hydride could be seen floating on the surface of the sodium in the pump tank. It was considered that such material, if raised inadvertently into the vent or feed lines, would contribute to the blocking of the gas valves. In 1988 both main and standby pumps had seized during the year, so during refurbishment a modification was carried out on the pumps to deal with the accumulation of these solid impurities. A jet of sodium was diverted from the discharge sides of the pumps to disperse the deposits and prevent valve blockages, as shown in Figure 5.3. This modification was not entirely successful in preventing further blockages and the pumps seized again in 1989 and 1990.

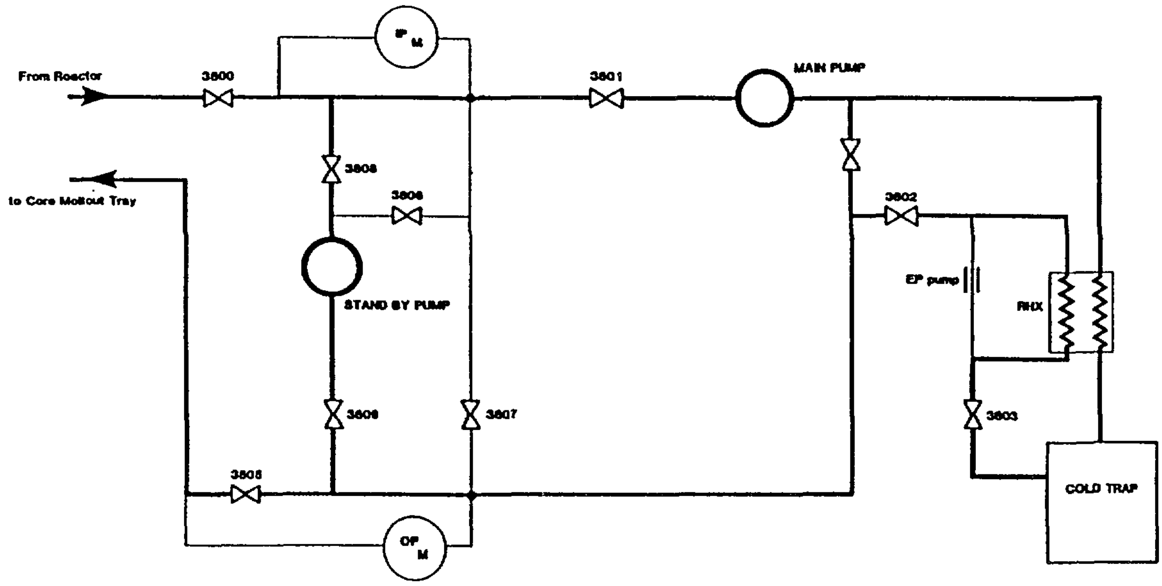

Figure 5.1. Schematic diagram of the PFR Primary Cold Trap Loop

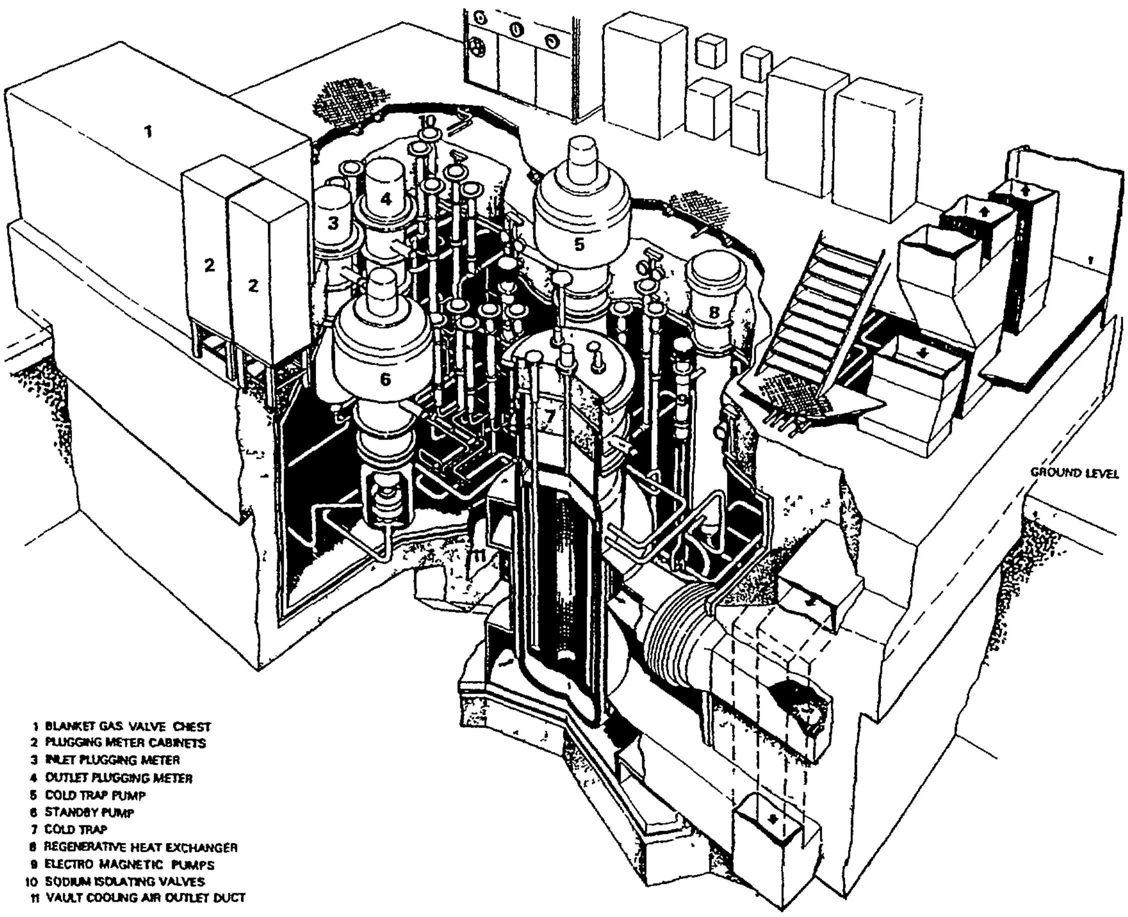

Figure 5.2. The PFR Primary Cold Trap Loop

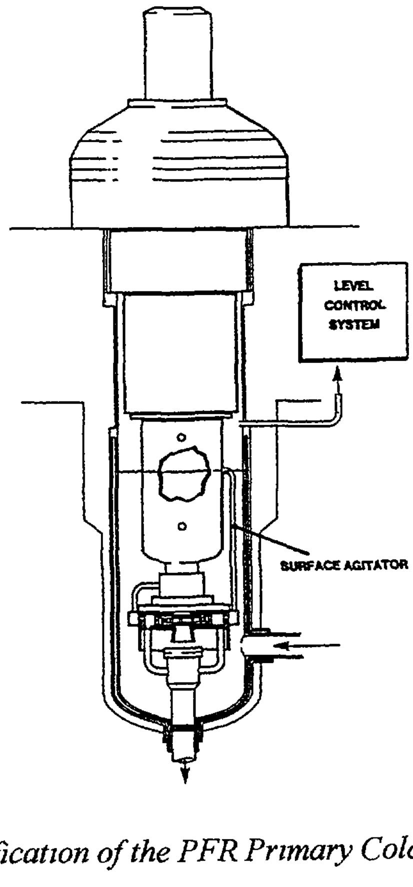

Figure 5.3. Modification of the PFR Primary Cold Trap Pumps

Conclusion

Sodium level control problems in the PCTL have frequently required stripdown and maintenance of the sodium pumps to clear sodium from the pump shaft annulus. The problem has never been fully resolved.

6. THE EFFECT OF SODIUM AEROSOLS ON THE OPERATION OF PFR

Sodium aerosols have been reported as a source of problems in all fast reactors. In PFR two particular problems arose during operation between 1974 and 1994.

A diagram of PFR absorber rods is shown in Figure 6.1. 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. On a trip all ten rods dropped. Magnet current, apparent rod weight, rod release time and time of flight were measured by installed instrumentation.

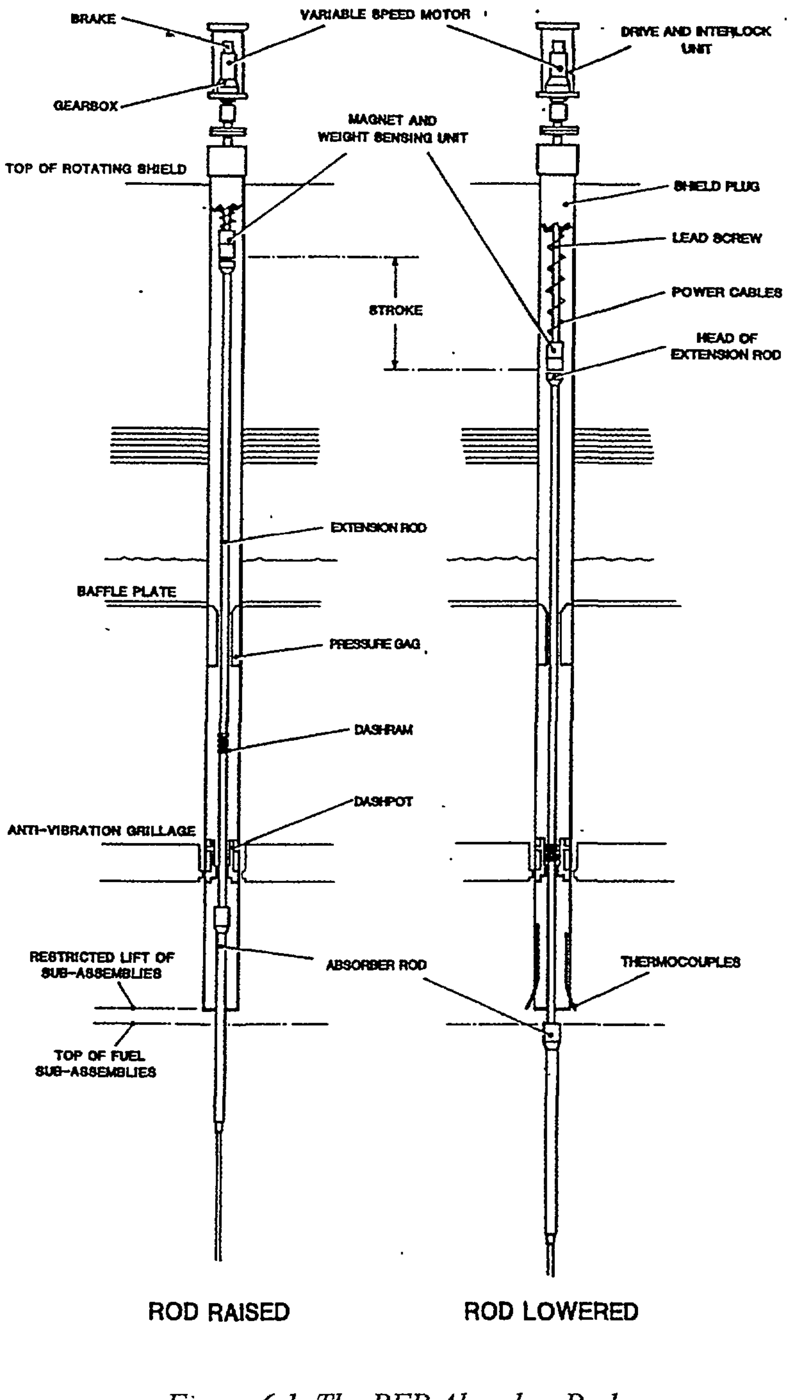

Figure 6.1. The PFR Absorber Rods

At the design stage it was recognised that following a trip there was potential for sodium aerosol to deposit on the parted magnet faces in the gas space. A continuous purge flow of argon gas was passed through the absorber liner tube to avoid the problem. Following trips the flow was enhanced to try to prevent aerosol building up. This however did not solve the problem fully and throughout its operational life PFR suffered from gradual reduction in the efficiency of the magnets due to the gradual buildup of sodium deposits on the faces of the magnets. This led to plant trips on a number of occasions due to absorbers dropping off their magnets during power operation.

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.

Because of the possibility of distortion due to swelling caused by neutron-induced voidage (NIV), which caused interaction between the rods and the guide tubes, absorber friction was measured on a regular, but initially infrequent, basis. Each rod in turn was raised and lowered while the reactor was operating (criticality being maintained by moving the remaining rods to compensate). The apparent weight of the rod, which varied along the stroke, was recorded. Up to 1985 measured friction was higher than expected, at a maximum of 40 kg compared with an expected level of about 25 kg due to the effect of NIV. The additional load was thought to be caused by the effects of aerosol but was not particularly worrying.

After 1985 when prolonged high power operation became more common friction levels were found to increase rapidly to about 80 kg. At such levels, coupled with the effect of magnet face contamination by aerosol, rods were liable to pull off and drop as they were being raised. Regular weekly exercising of the rods at intervals of seven days was initiated to monitor this effect.

Typical rod movements during exercising are shown in Figure 6.2. The exercising was found to have the effect of reducing friction. Peak friction occurred at about 900 mm (see Figure 6.2) but the magnitude depended on whether the rods were raised or lowered first. Lowering first reducing friction levels. Figure 6. 3 shows an actual trace of friction for a control rod, which in this case started by being raised. The actual weight of absorber plus extension tube is 230 kg. The dotted line marked FINISH is an immediate partial repeat of the test showing the decrease in friction resulting from the first test.

The friction is believed to have been caused by the buildup of sodium aerosol deposits on the inside of the liner tube and on the extension rod of the absorber in the cool region of the rotating shield, as shown in Figure 6.4. Lowering the rod moved the deposit on the guide tubes into the warmer region below the roof insulation where it melted off, reducing the friction.

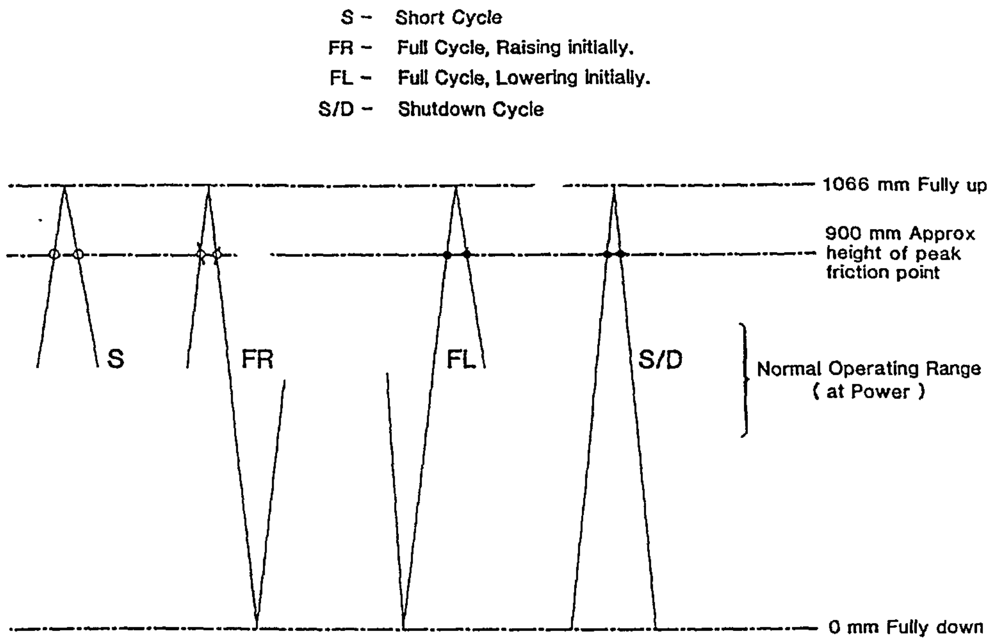

Figure 6.2. Movement of the PFR Absorber Rods during Exercising

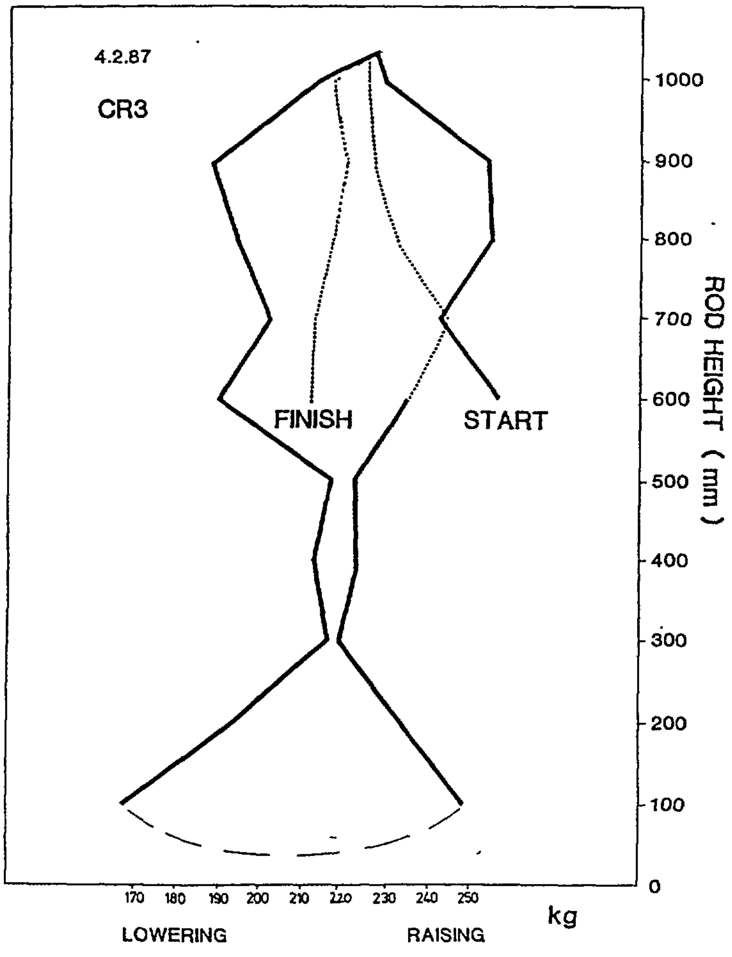

Figure 6.3. Apparent Weight of a PFR Absorber Rod during Exercising

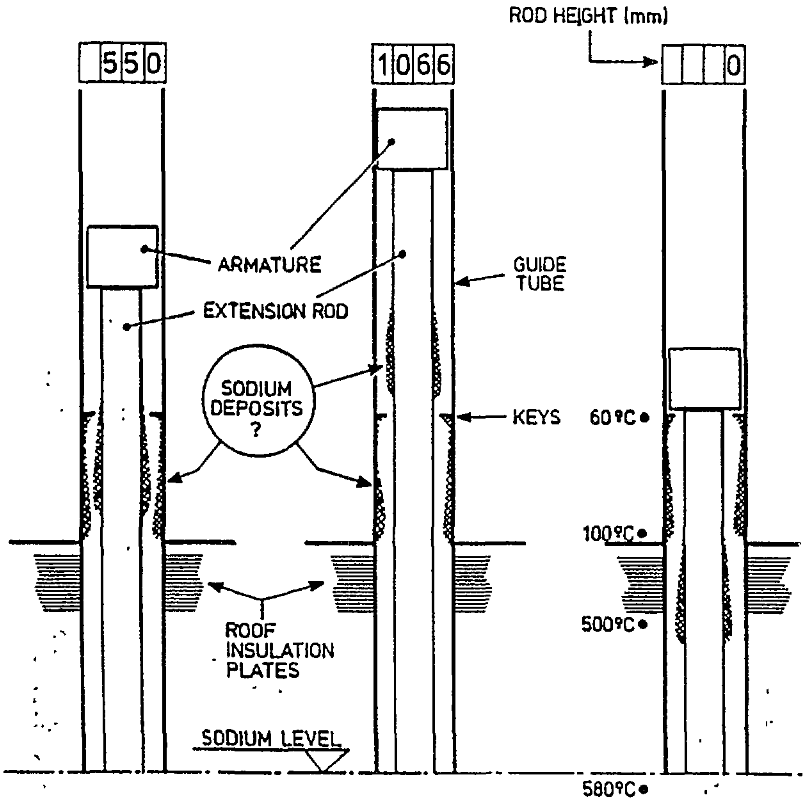

Figure 6.4. Location of Sodium Aerosol Deposits on a PFR Absorber Rod Mechanism

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 efpd of operation for friction levels to begin to rise noticeably.

Conclusion

In the case of PFR sodium aerosols caused no operational problems because movement of the absorber rods was carefully monitored and deposits were cleaned off well before they interfered with the mechanisms. The only effect was the operational burden of exercising the rods and cleaning the magnet faces. Aerosols had no observable effects on magnet parting times or rod drop times.

7. CRACKS IN THE PFR AIR HEAT EXCHANGERS

PFR had three thermal syphon decay-heat rejection loops, shown in Figure 7.1. 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 Figure 7.2. 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.

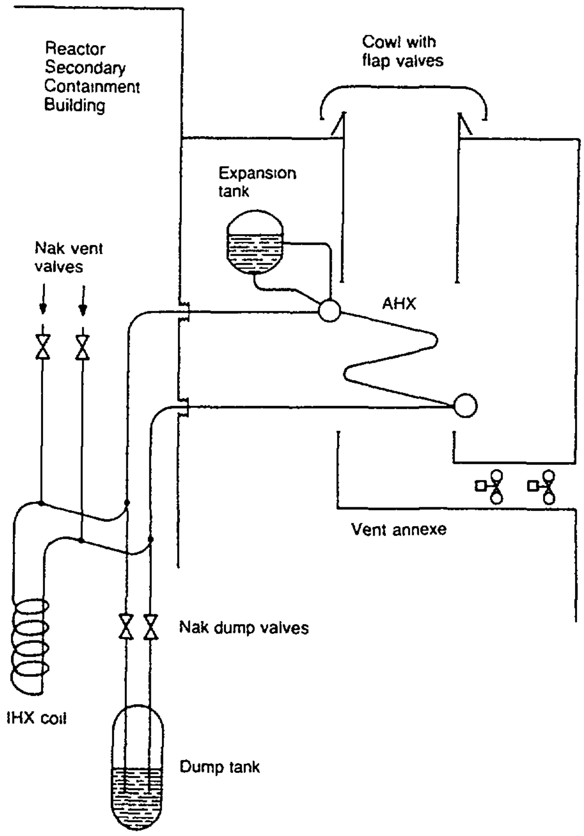

Figure 7.1. Schematic Diagram of a PFR Thermal Syphon Decay Heat Rejection Loop

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 following changes were made and are shown in Figure 7.2.

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

Two of the new RAHXs were fitted in 1986 and the third in 1987. Operation was trouble free until 1996, when the thermal syphons were finally emptied for decommissioning.

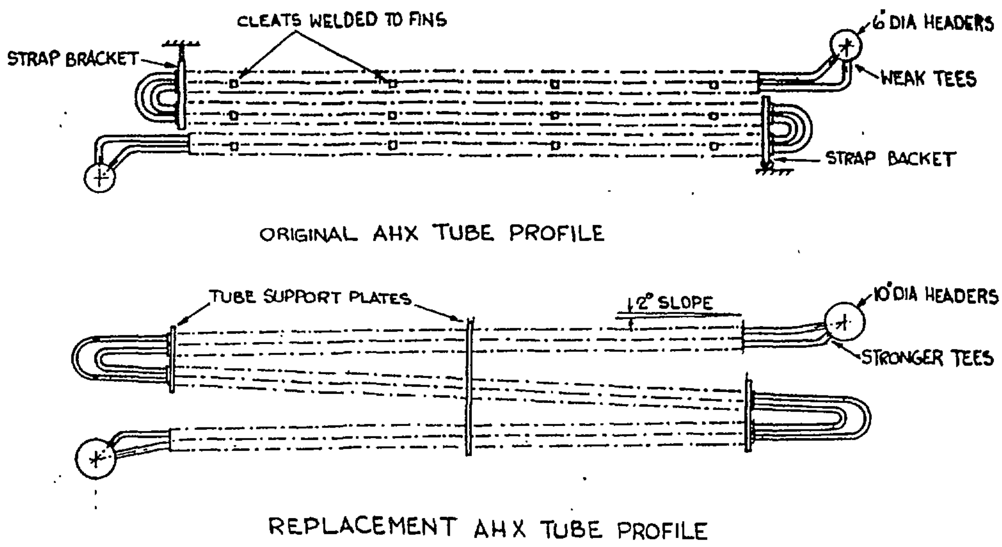

Figure 7.2. The Original and Replacement PFR THermal Syphon Air Heat Exchangers

Conclusion

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.

REFERENCES TO SECTION 7

D B Melhuish and A Sandison: "Engineering Improvements to PFR": *Nuclear Energy*, 1992, **31**, 193 - 205

8. THE EFFECT OF NEUTRON-INDUCED DISTORTION ON THE OPERATION OF PFR

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, as shown in Figure 8.1. 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 as shown in Figure 8.2, although not all subassemblies were affected. Two subassemblies in particular, "JRA" and "GYN" (see Figure 8.2), 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.

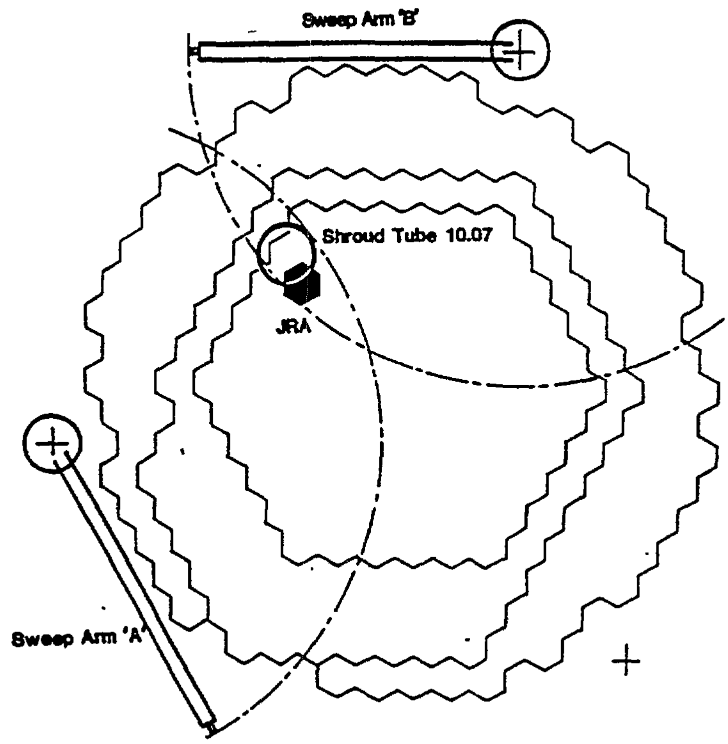

Figure 8.1. Location of an Elongated Subassembly in the PFR Core by the Sweep Arms

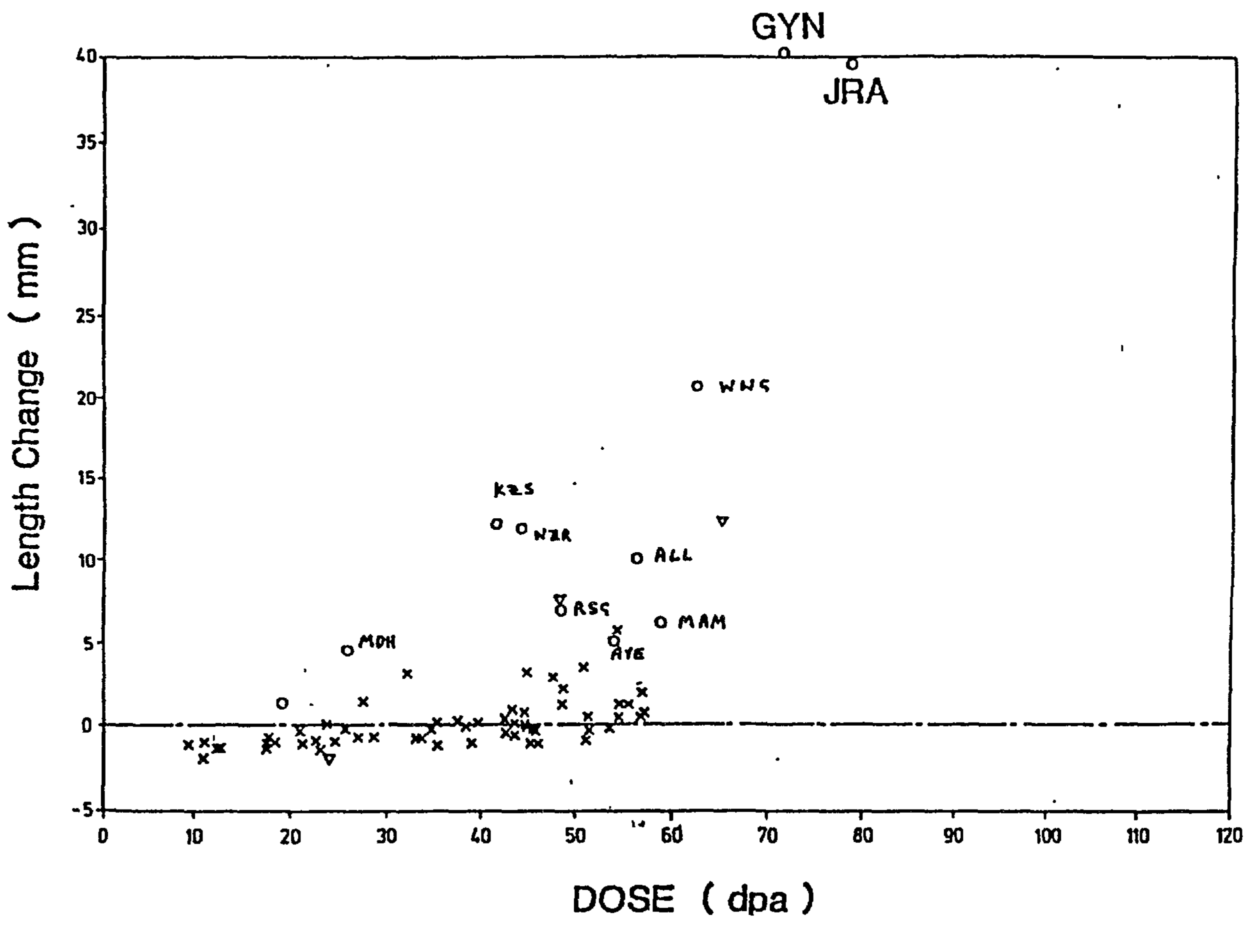

Figure 8.2. Increase in Length of EN58B Subassemblies in the PFR Core

On only one occasion were there 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, calculations 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.

Conclusion

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.

9. THE PFR PRIMARY CIRCUIT OIL SPILL

The PFR primary sodium circulation system is shown in Figures 9.1 and 9.2, and Figure 9.3 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 9.4 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.

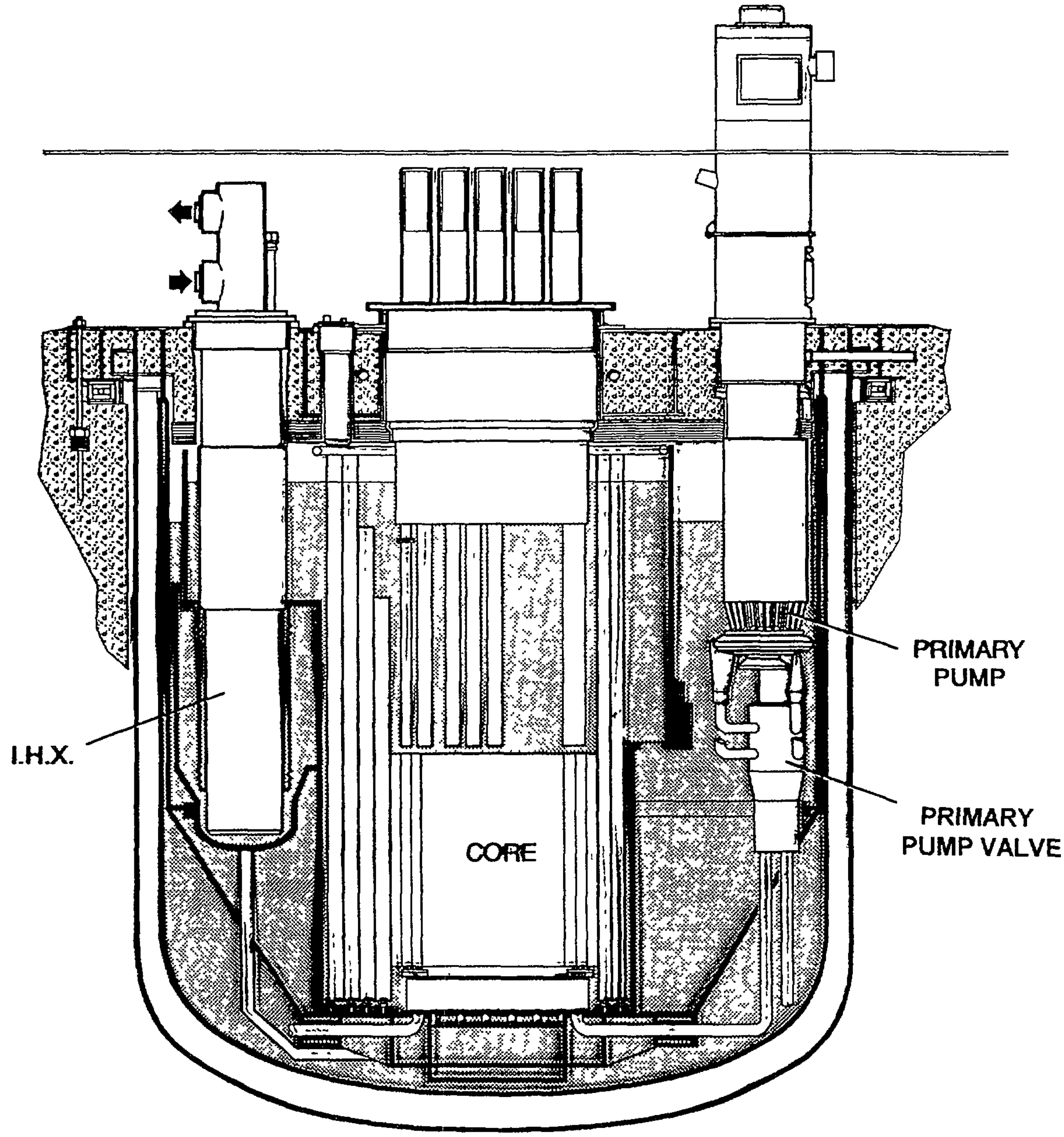

Figure 9.1. The PFR Primary Circuit

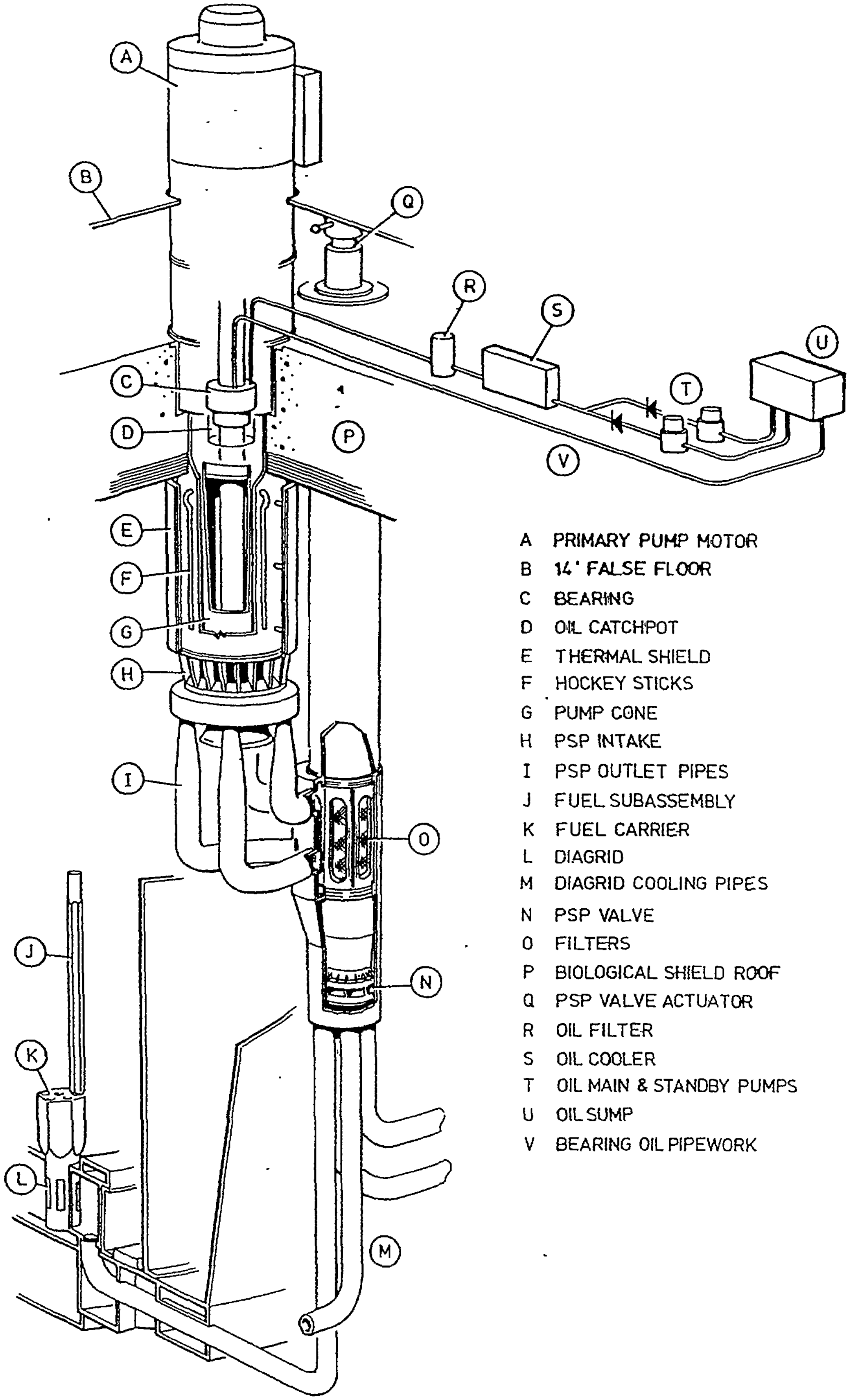

Figure 9.2. A PFR Primary Sodium Pump and its associated Valve and Filter Assembly

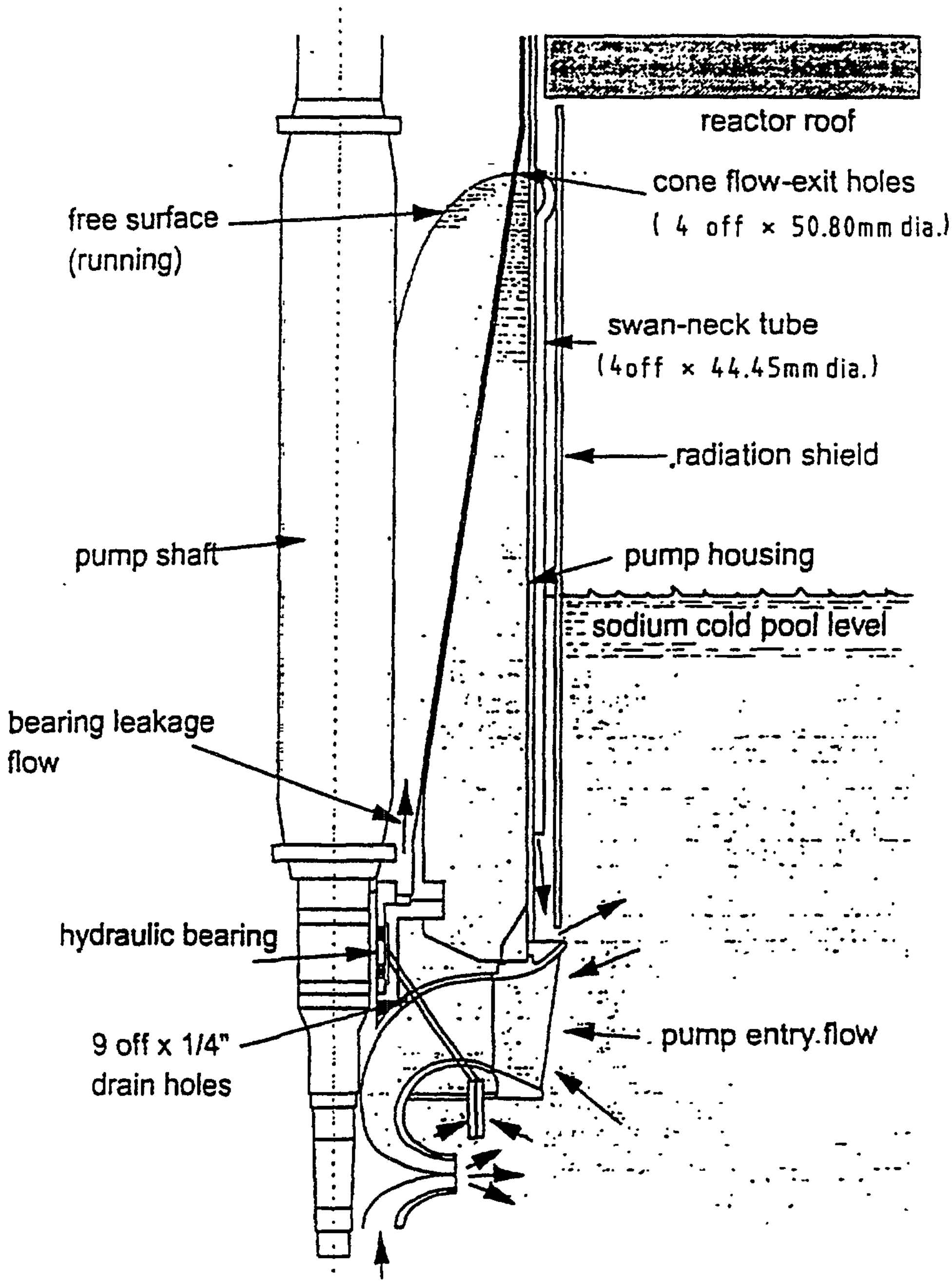

Figure 9.3. Detail of a PFR Primary Sodium Pump Housing

During this period of operation some 65 litres 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 ("O" in Figure 9.2) failed due to high differential pressure because it became blocked by oil-sodium reaction products. It is also thought that partial blockage of the pump casing overflow pipe (Figure 9.3) was to lead to the major problem in 1991.

No further problems were observed until 1990 when it was noted that PSP1 drive current was slowly dropping at constant pump speed and its discharge pressure was rising. The situation was under observation when suddenly the current rose, the discharge pressure

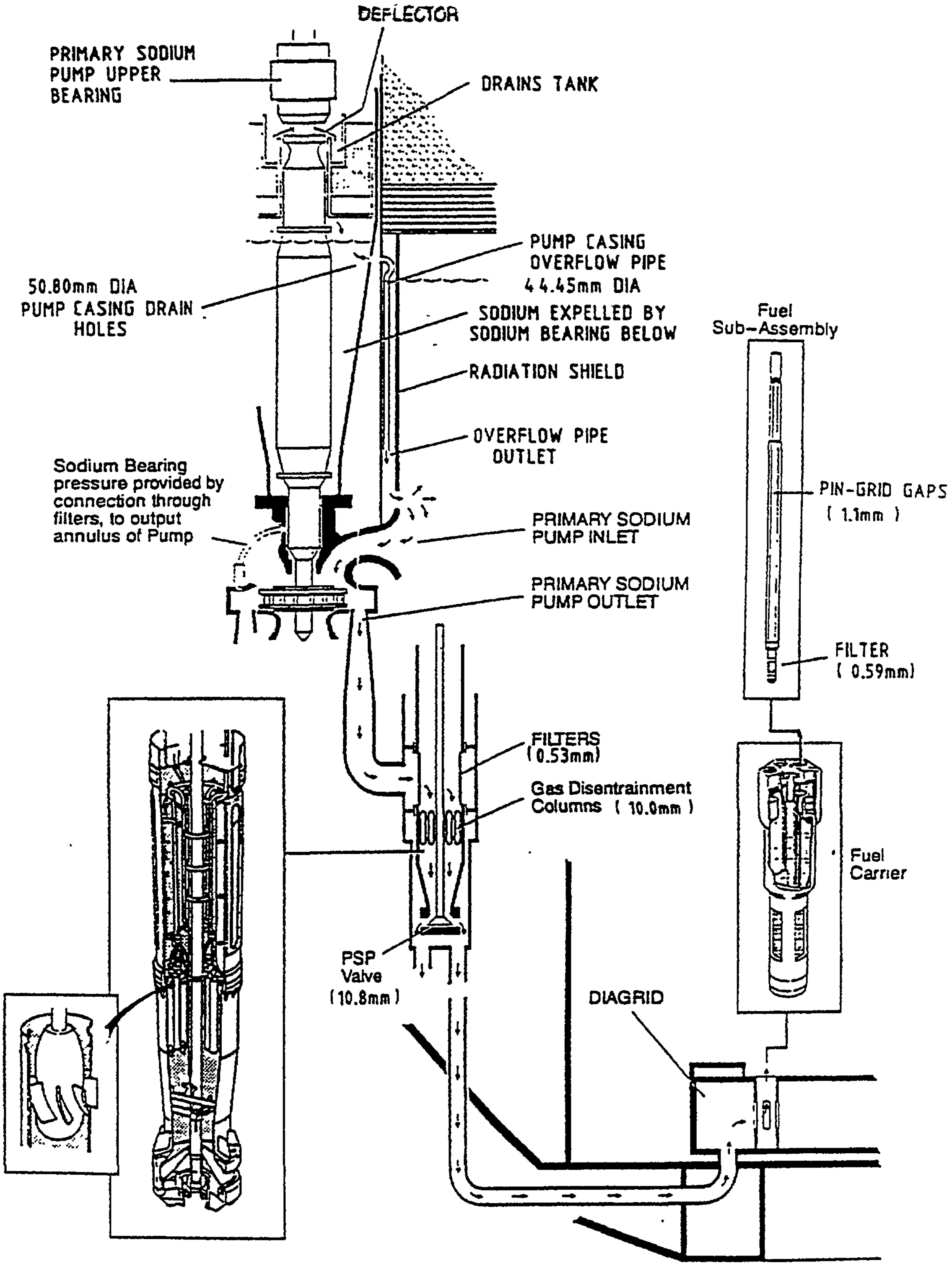

Figure 9.4. Filter Mesh Sizes in the PFR Primary Circuit

fell and the flow of coolant through the core increased. A similar sequence of events occurred a few weeks later, when flows and current returned to normal. Although no oil spill was recorded it is now believed that oil had entered the system and blocked the filter of PSP 1, causing it to fail in stages. When it had failed completely and offered no resistance the coolant flow returned to its normal value.

In 1991 a similar effect began to appear on PSP 3, and by the middle of the year the coolant flow was estimated to be 82 % of normal. Again no source of this apparent blockage is known but an oil spill is suspected.

On 24 June 1991 there was low flow in the argon gas blanket circulating system. During attempts to improve the flow by venting the gas blanket, high radiation levels in the PSP 2 well indicated that sodium had been raised into the top bearing drains tank. This was a result of the blocked overflow pipe causing a pressure differential between the gas blankets in the main vessel and the pump casing. The method used to vent the gas blanket resulted in a preferential flow of gas from the pump casing, reducing the pressure in it and forcing sodium up the pump shaft. On this occasion it is certain that oil was displaced from the drains tank into the reactor primary circuit.

Some subassembly core outlet temperatures in the sector of the reactor supplied by PSP 2 began to rise but stabilised after about 1.5 days. An accelerated increase in PSP 3 filter pressure differential was noted (by now of course PSP 3 filter was the only one intact). The plant was under close observation when on 29 June the oil bearing on PSP 2 failed completely causing a further oil spill, and the plant was tripped. It was observed from flow and differential pressure readings that PSP 3 filter failed at this time.

It is estimated that up to 17 litres of oil was released into the cone of PSP 2 during the June 1991 incidents. Release of oil debris from the pump cone into the main primary circuit was gradual, taking about 1.5 days as indicated by the increase of core subassembly outlet temperatures and the increase in the primary pump valve filter pressure drop.

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 metres, 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.

Conclusion

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 to gas bearings for its PSPs following the PFR oil ingress incident.

ACKNOWLEDGEMENT

The work described in this paper was originally financed by the UK Department of Trade and Industry. Preparation of the paper was financed by British Nuclear Fuels Ltd.

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SODIUM LEAKAGE EXPERIENCE AT THE PROTOTYPE FBR MONJU

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Abstract

Monju is Japan's prototype fast breeder reactor : 280 MWe (714MWt), fueled with mixed oxides of plutonium and uranium, cooled by liquid sodium. Construction was started in 1985 and initial criticality was attained in April 1994

On 8th December 1995, sodium leakage from a secondary circuit occurred in a piping room of the reactor auxiliary building. The secondary sodium leaked through a temperature sensor, due to the breakaway of the tip of the thermocouple well tube installed near the secondary circuit outlet of the intermediate heat exchanger (IHX). The reactor remained cooled and thus, from the viewpoint of radiological hazards, the safety of the reactor was secured. There was no release of radioactive material. There were no adverse effects for personnel and the surrounding environment. The thermocouple well tube failure resulted from high cycle fatigue due to flow induced vibration. It was found that this flow induced vibration was not caused by well-known Von Karman vortex shedding, but a symmetric vortex shedding. The design of the thermocouple well, which was subject to avoid this phenomenon, was reviewed. A new design guide against the flow-induced vibration was prepared by JNC (Japan Nuclear Cycle Development Institute). That is more comprehensive and definitive than the existing guide "ASME N-1300" (Flow-induced vibration of tube and tube banks). New thermocouple well designs were proposed consistent with this design guide.

To prevent a recurrence of the secondary sodium leakage incident, comprehensive design review activities were started for the purpose of checking the safety and reliability of the plant. As a result, several aspects to be improved were identified and improvements and countermeasures have been studied. The main improvements and countermeasures are as follows:

- To enable the operators understand and react to incidents quickly, new sodium leakage detectors (TV monitors, smoke sensors) and a new surveillance system will be installed.
- To reduce the amount of sodium leakage and damage by spilt sodium, the drain system will be remodeled to shorten the drain time.
- To extinguish a sodium fire in the secondary circuit, a nitrogen gas injection system will be installed.
- To limit the spread of aerosol, the secondary circuit area will be divided into four smaller zones.

These countermeasures will enhance the safety and reliability of the plant with regard to sodium leakage incidents.

1. INTRODUCTION

The construction of Monju is a major milestone in the Japanese national FBR development project which is based on the Atomic Energy Commission's long-term nuclear energy program. The Japan Nuclear Cycle Development Institute (JNC) is responsible for the management of the project. Construction of Monju began in October 1985 at a site near the Tsuruga city. The principal data on plant design and performance are shown in TABLE 1. Loading of the fuel assemblies into the core started in October 1993 and the reactor attained initial criticality in April 1994. Monju achieved

TABLE 1. PRINCIPLE DESIGN AND PERFORMANCE DATA OF MONJU

Reactor type	loop-type
Number of loops	3
Thermal output	714 Mwt
Electrical output	280 Mwe
Fuel material	PuO2-UO2
Core dimensions	
Equivalent diameter	1,790 mm
Height	930 mm
Plutonium enrichment (inner core / outer core)	
(Pu fissile %)	
Initial core	15 / 20
Equilibrium core	16 / 21
Fuel inventory	
Core (U+Pu metal)	5.9 t
Blanket (U metal)	17.5 t
Average burnup	80,000 MWD/T
Cladding material	SUS316
Cladding outer diameter/thickness	6.5 / 0.47 mm
Blanket thickness	
Upper / lower / radial	30 / 35 / 30 cm
Breeding ratio	1.2

Reactor vessel	
height / diameter	18 / 7 m
Primary coolant systems	
Primary coolant sodium mass	760 ton
Inlet / outlet reactor temperature	397 / 529 ℃
Primary coolant flow rate	5.1×10^{6}kg/h/loop
Primary coolant flow velocity	6m/s(inlet),4m/s(outlet)
Secondary coolant systems	
Secondary coolant sodium mass	760 ton
Inlet / outlet IHX temperature	325 / 505 ℃
Secondary coolant flow rate	3.7×10^{6}kg/h/loop
Secondary coolant flow velocity	5 m/s
Water - steam systems	
Feed water flow rate	113.7×10^{4}kg/h
Steam temperature (turbine inlet)	483 ℃
Steam pressure (turbine inlet)	12.7 MPa
Type of steam generator	Helical coil
Refueling system	Single rotating plug with fixed arm FHM
Refueling interval	6 months

Japan's first generation of electricity by an FBR in August 1995 and the electric power was raised gradually for a program of power buildup tests to be carried out ; the rated power test planned for June 1996. It was in the course of this program that the tip of a thermocouple well tube in the secondary circuit (loop C) broke away causing a sodium leak on 8th December 1995.

This paper summarizes the sodium leak incident, the cause of thermocouple well tube failure, and the improvement and countermeasure programs against sodium leakage incidents.

2. SUMMARY OF THE ACCIDENT AND POST-ACCIDENT RESPONSE

After a plant trip test, Monju restarted operation on 6th December 1995. On 8th December, power was being raised for the next plant trip tests, part of 40% electric power tests. The thermal power had reached 43% when an alarm sounded at 19:47 due to an off-scale sodium temperature at the outlet of IHX in the secondary circuit loop C. A fire alarm (smoke detector) sounded at the same time. A sodium leak alarm in the secondary circuit followed. The plant conditions of Monju at that time are shown in Fig.1. The presence of smoke was confirmed when the door of the piping room was opened. The plant operators decided to begin normal shutdown operations because they judged it was a small sodium leak had occurred. Reactor power-down operations began at 20:00.

The state inside the piping room (C) was checked again and an increase in white fume was observed. Accordingly, the reactor was manually tripped at 21:20. After the trip, the reactor was cooled down by the auxiliary cooling system (ACS) and was maintained in the low-temperature shutdown state. To minimize leakage, sodium in the secondary circuit (loop C) was drained at 22:55 and drain operations were complete at 00:15, the following day.

As it was in the secondary circuit, there was no release of radioactive material. There were no adverse effects for personnel and the surrounding environment.

An inspection of the affected piping room on 9th December confirmed the presence of solidified materials associated with the sodium leak around and near the thermocouple well of the outlet of the secondary side of IHX. The state of the piping room after the sodium leak is shown in Fig.2. Approximately 1m^3 of a sodium oxide on the 6 mm thick steel floor liner formed a semicircular mound, nearly 3 m in diameter and 30 cm high. Sodium aerosol was lightly diffused over and accumulated on the floor and walls of the room. The ventilation duct directly under the

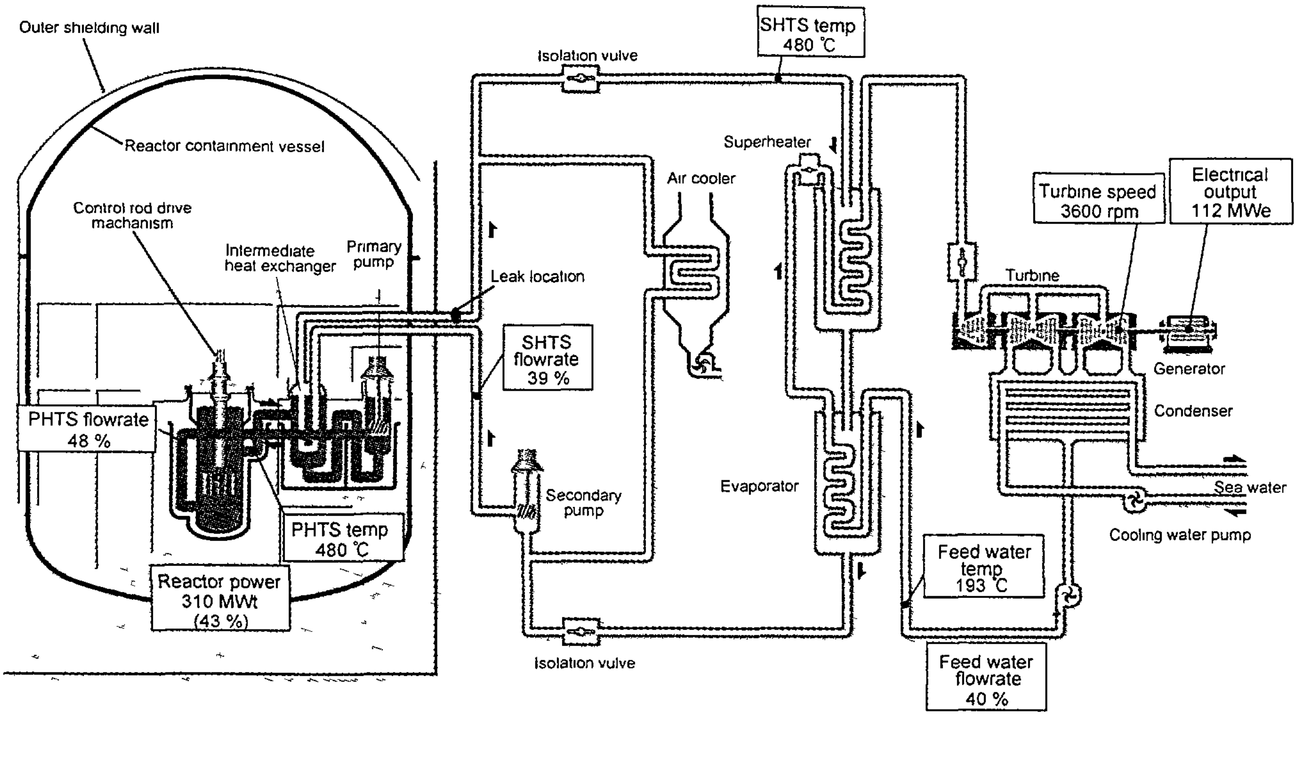

Fig 1. MONJU plant condition (just before the sodium leak).

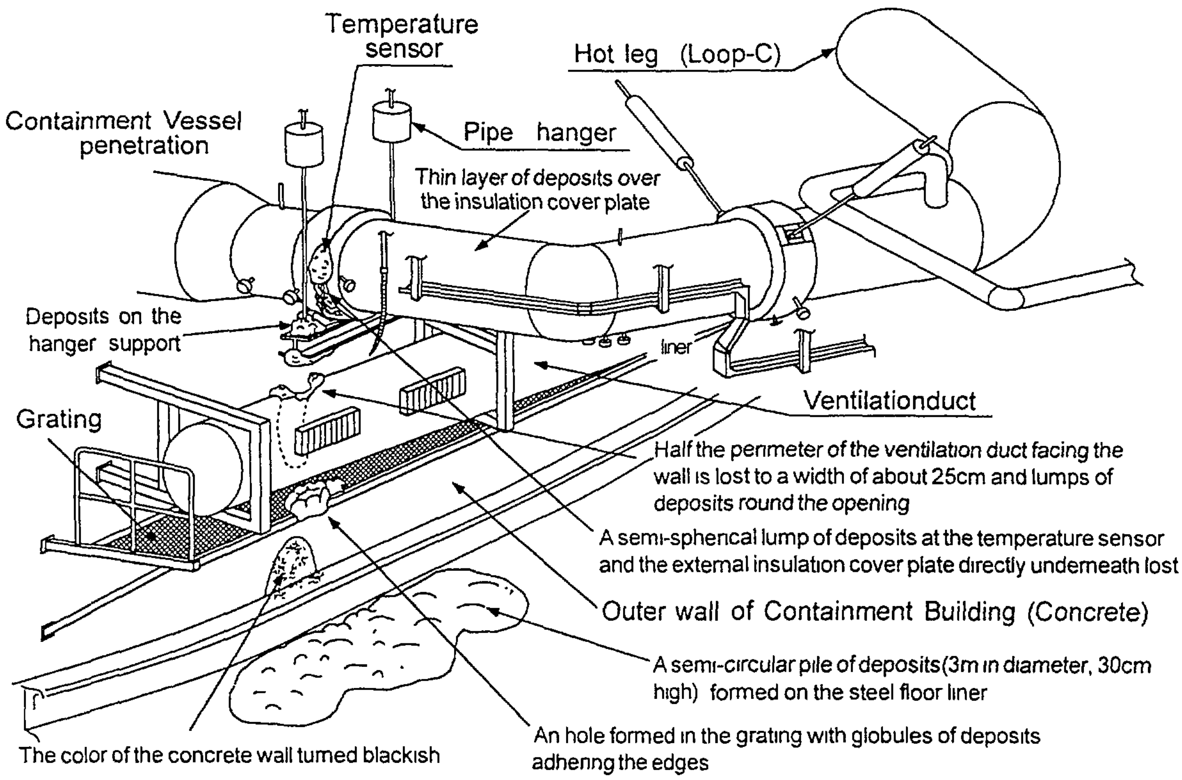

Fig. 2. Sketch of the affected area.

thermocouple well developed a hole extending over half the perimeter facing the wall with lumps of deposits around the opening. On the steel walkway grating under the thermocouple well, an opening was formed with globules of deposits stuck around the edges. There were no further anomalies observed in the piping of the secondary circuit. Sodium compounds covered the entire floor of the loop C steam generator (SG) room adjacent to the piping room and the passages on the first and second floors.

Temperature sensor and the well tube which leaked was examined to investigate the cause. On 7th and 8th January, radiographs were taken of areas close to the temperature sensor in order to estimate the extent of adherent sodium compounds around the temperature sensor and to assess its structural condition. The thermocouple (3 mm diameter) was found to be bent at 45 degrees toward the downstream flow direction. The protective tube of the temperature sensor was found to be filled with sodium compounds. On 9th February, the temperature sensor was cut out for detailed investigation.

On 28th March, the sleeve tip, a cylinder 1 cm in diameter and 15 cm in length, was located in the sodium-inlet part of the superheater, and JNC recovered the broken sleeve tip on April 24.

3. INVESTIGATION OF THE CAUSE

A taskforce was formed to direct the investigation of the incident. This taskforce was organized by Science and Technology Agency (STA) and the members consisted of STA's Nuclear Safety Technology Panel experts. JNC performed its own work under the direction of the Taskforce.

The outline of the investigation works was as follows ;

A. The affected temperature sensor

The thermocouple wires are enclosed by a sheath which is itself housed within the well tube. The tip of well tube is inserted horizontally into the center of the pipe. This tip (some 15 cm in length) is thinner in diameter than the base of the sensor. (Fig.3)

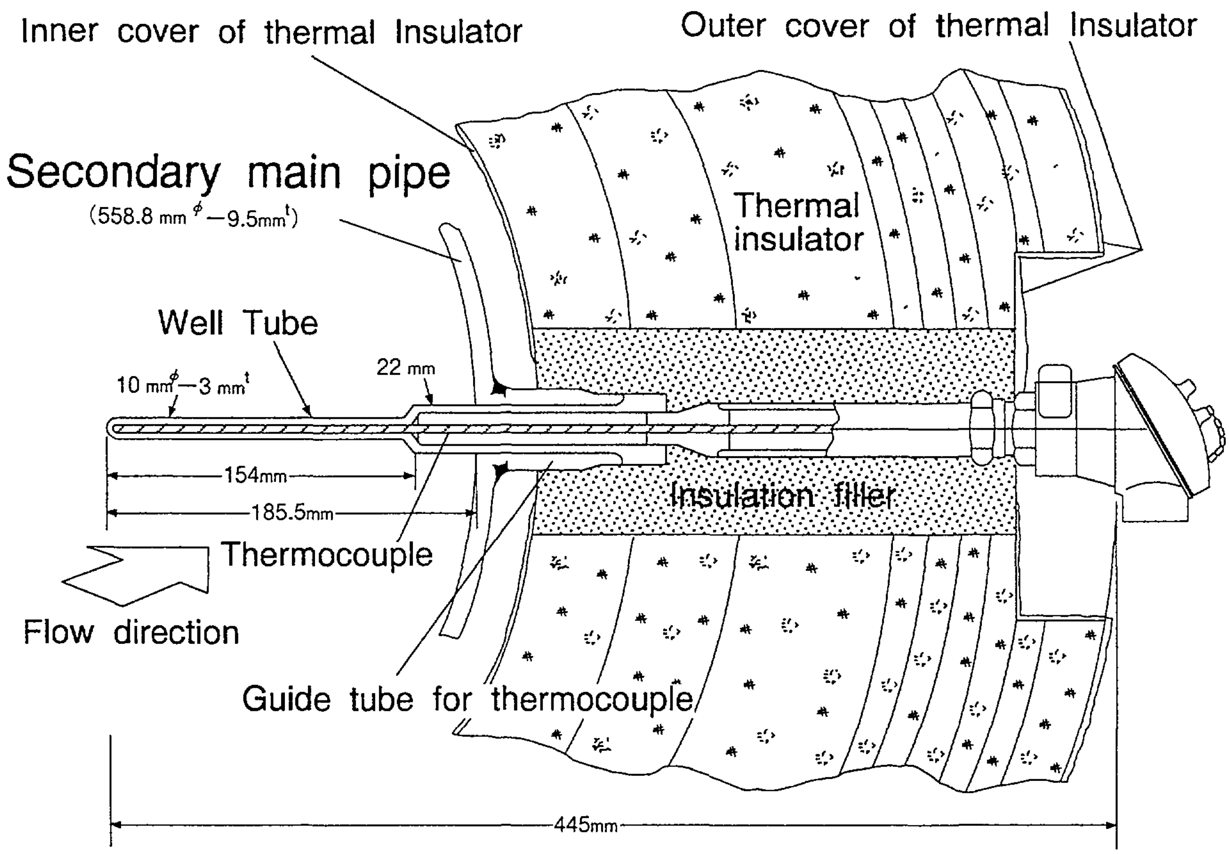

Fig. 3. The thermocouple well of the secondary circuit.

X-ray photographs were taken to estimate the extent of adherent sodium compounds around the temperature sensor and to assess structural condition. These revealed that the tip of the well tube was missing and the sheath containing the thermocouple wires was bent downstream. Other anomalies were not observed by the photographs. At that time, it was clear that the sodium leaked into the temperature sensor through the broken well tube. (Fig.4)

Temperature sensor, together with a small section of the adjacent pipe wall, was cut out for the detailed investigation and transported to the Japan Atomic Energy Research Institute (JAERI).

B. The cause of the well tube breakage

Detailed microscopic and metallographical examinations of the well tube and the fracture surface, examination of welded parts of the well, detailed examination of the sheathed thermocouple were carried out at JAERI and the National Research Institute of Metals (NRIM). The detailed measurement of the damaged part (fracture surface) was undertaken with microscopes and laser microscopes. As a result of these inspections, it was found that the fracture surface showed the typical features of high cycle fatigue with crack initiation, very slow propagation, and final ductile rupture, as shown in Fig.5. In parallel, flow-induced vibration analysis and mockup tests were conducted to identify the direct cause of the failure. These investigations confirmed that the breakage of the thermocouple well was caused by high cycle fatigue of the well tube tip due to flow-induced vibration. This vibration was not caused by well-known Von Karman vortex shedding, but a symmetric vortex shedding (Fig.6). According to fatigue crack analysis on the basis of the flow rate history, it was estimated that the cracks initiated at the early stage of the 100% flow operation, propagated in the subsequent operation, finally leading to the well tube failure in the last 40% flow operation. It was confirmed that the estimated breakage process agreed with the investigation results of the fractured surface and the full scale in-water experiment .

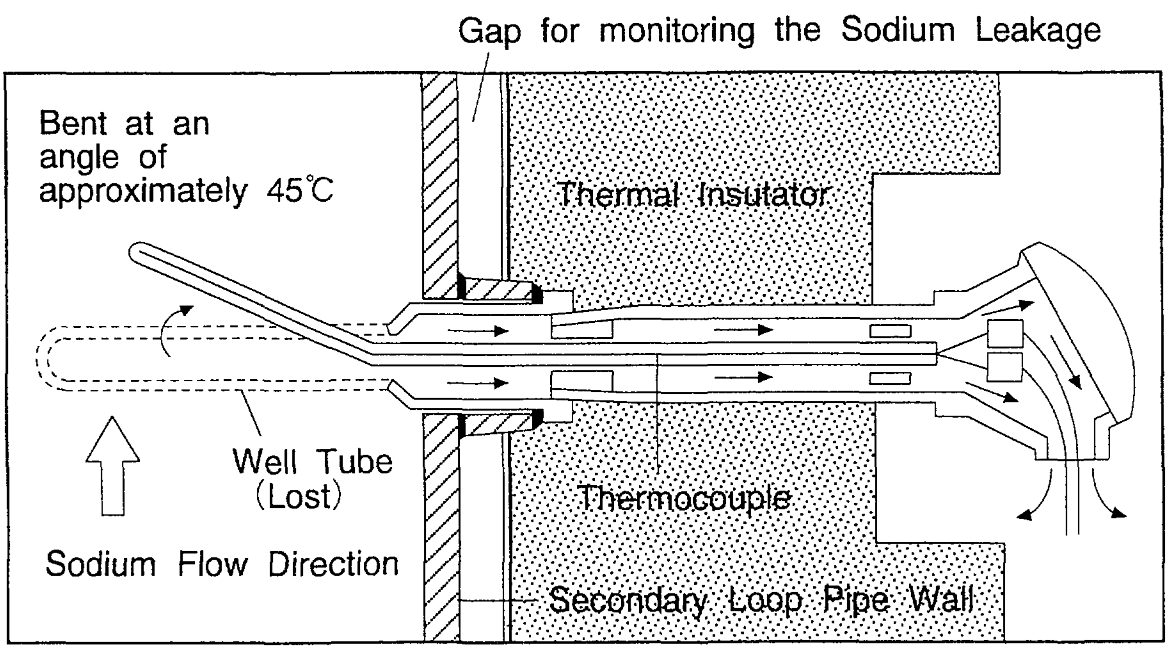

Fig. 4. The sodium leak flow path.

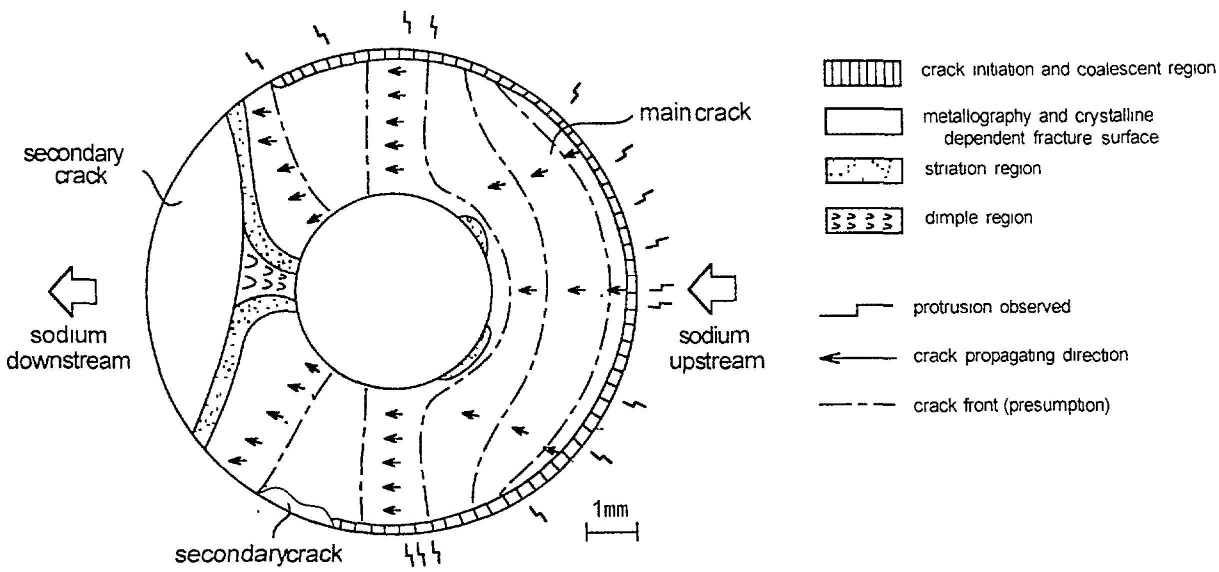

Fig. 5. Microscopic inspection of the fracture surface of the thermocouple wall.

4. MEASURES AND SOLUTIONS

A. Prevention of sodium leakage

It was decided to replace all the thermocouple wells susceptible to vibration in the secondary circuit with modified designs. A new design guide against the flow-induced vibration was prepared by JNC. It is more comprehensive and definitive than the existing design guide "ASME N-1300".

Using this design guide, three new thermocouple well designs were developed, according to their functional requirements and locations. All the designs use a tapered well, instead of steep diameter change. A leak suppression mechanism is also incorporated into the new design. The three thermocouple designs, Type A, B, and C are shown in Fig. 7, along with the existing design. The "Type A" thermocouple well is directly welded to the pipe wall, while the "Type B" and "Type C" thermocouple wells are mounted on the existing nozzles. The three design types have different penetration designs and different insertion length into the flowing stream.

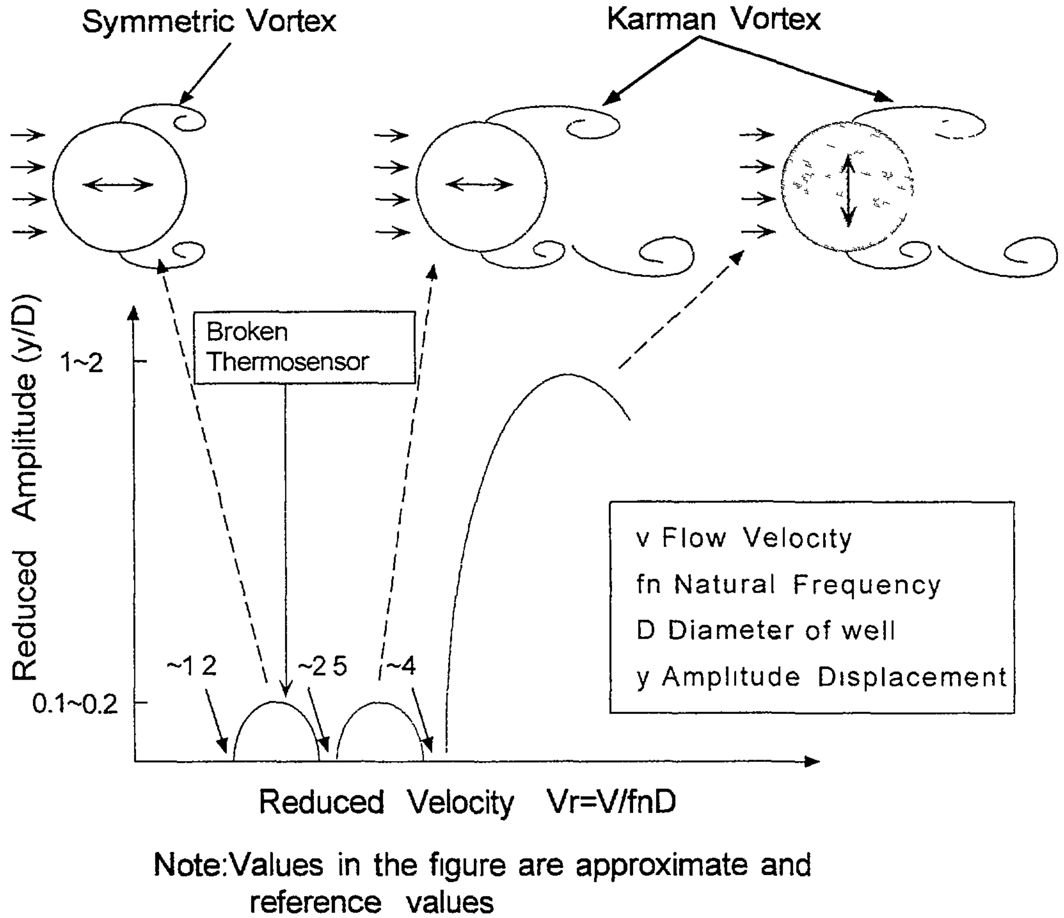

Fig. 6. Explanation of vortex shedding.

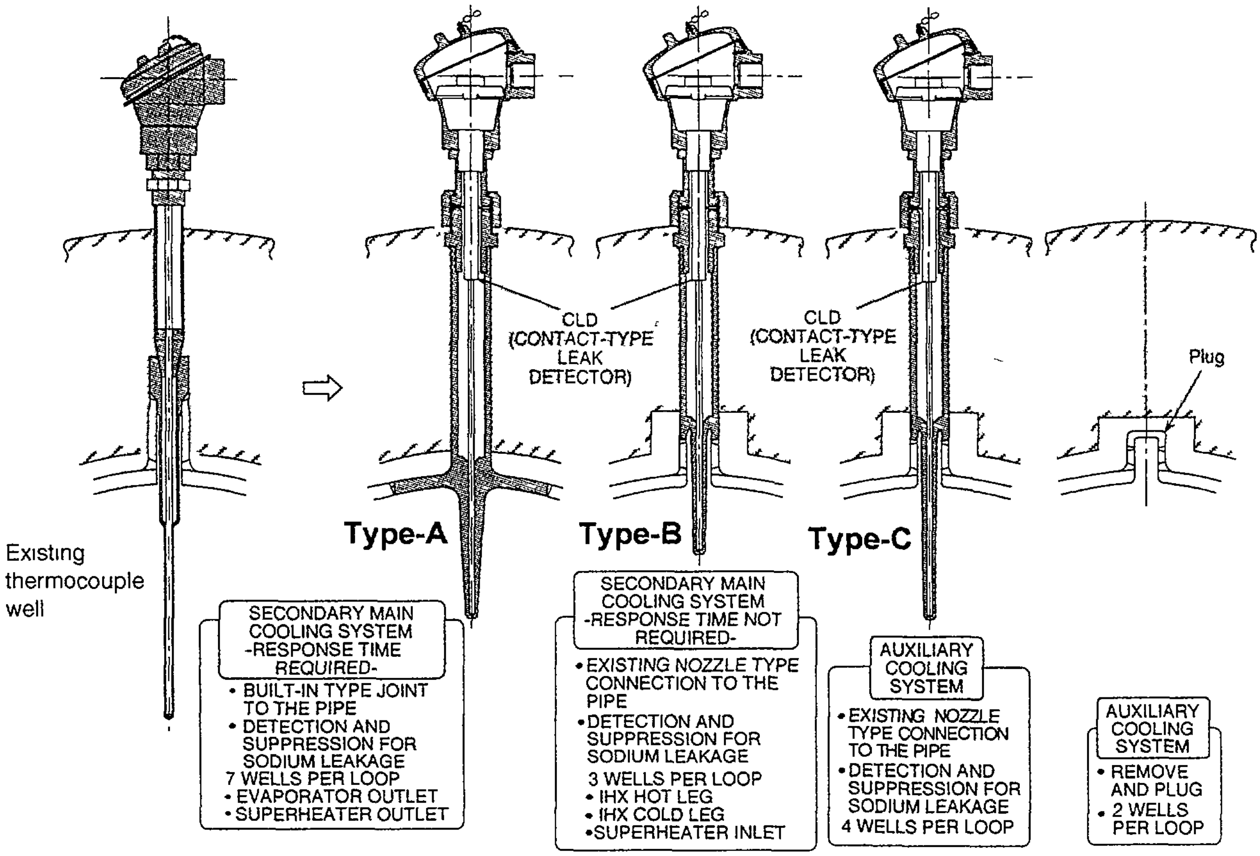

Fig. 7 Modified design concepts for the thermocouple wells on the secondary cooling system.

B. Mitigation of sodium leakage influence

A comprehensive review was performed concerning sodium leakage. The review concentrated on the following three points:

- Detection of sodium leakage at an early stage
- Reduction of the quantity of sodium leakage
- Limiting the diffusion of sodium aerosol and the combustion of spilt sodium

These were chosen in order to investigate if systems and components with regard to the sodium leakage could be improved or not.

The design of the systems and components was also reexamined considering the experience and knowledge accumulated at other fast reactors, research and development results and new findings. As a result of the review, some additional aspects were appeared. Some existing systems and components related to sodium leakage were found to require modification, and it was recommend that new systems and components should be introduced. Typical systems and components are ;

- Sodium leakage detection system (to be modified)
- Ventilation system (to be modified)
- Drain system (to be modified)
- Partition of the secondary circuit room (to be introduced)
- Nitrogen injection system (to be introduced)
- Thermal insulation structures (to be introduced)

Each of these is described in greater detail below :

(1) Sodium leakage detection system

It was possible to detect the sodium leakage because Monju already had sodium leak detectors and fire detectors in the secondary circuit. However it was found that these were insufficient for the operators to grasp accurately the conditions in the room at an early stage. To help the operators confirm sodium leakage more quickly and easily, the detectors will be increased in number and diversified method and an integral leakage monitoring system will be installed.

The fire sensors are able to detect sodium leakage by two different methods. One is smoke (aerosol) detection and the other is by temperature increase in the room where a sodium fire occurs. Monju already used the former method and this type of sensor will be increased to detect the sodium leakage more quickly and more certainly. The latter method type fire sensors will be introduced for greater diversity in the detection of the sodium fires. Visual information was found to be useful for operators to know the condition of the room, and so, TV cameras will be installed in the secondary circuit.

The integral leakage monitoring system will be composed of three parts : detection, data processing and monitoring display. Detection will consist of the signals from fire sensors, sodium leak detectors, plant process sensors (sodium flow rate, sodium level in the overflow tank, etc.) and visual information from the TV cameras.

A monitoring display panel will be installed in the main control room and information from the various sensors and TV cameras will be displayed. The new leakage monitoring system should enable the plant operators to take the necessary actions - such as a plant trip and loop drain - earlier, since they will be able better to understand the condition at the leak site without leaving in the main control room. A schematic of this system is shown in Fig.8.

(2) Ventilation system

To control the spread of aerosol and the combustion of sodium, the ventilation system will be shut down more quickly by signals from the newly installed integrated leakage monitoring system. At present, in the case of small and medium-scale leaks, the shut down of the ventilation system is carried out manually by the operators. Only in the case of a large sodium leak, the ventilation is shut down automatically by an interlock signal from an abnormally low level in the steam generator. At present, if a small or medium-scale leak occurs, the sodium aerosol spreads from the leak site room to all areas that are connected by the ventilation ducts until the ventilation system is shut down.

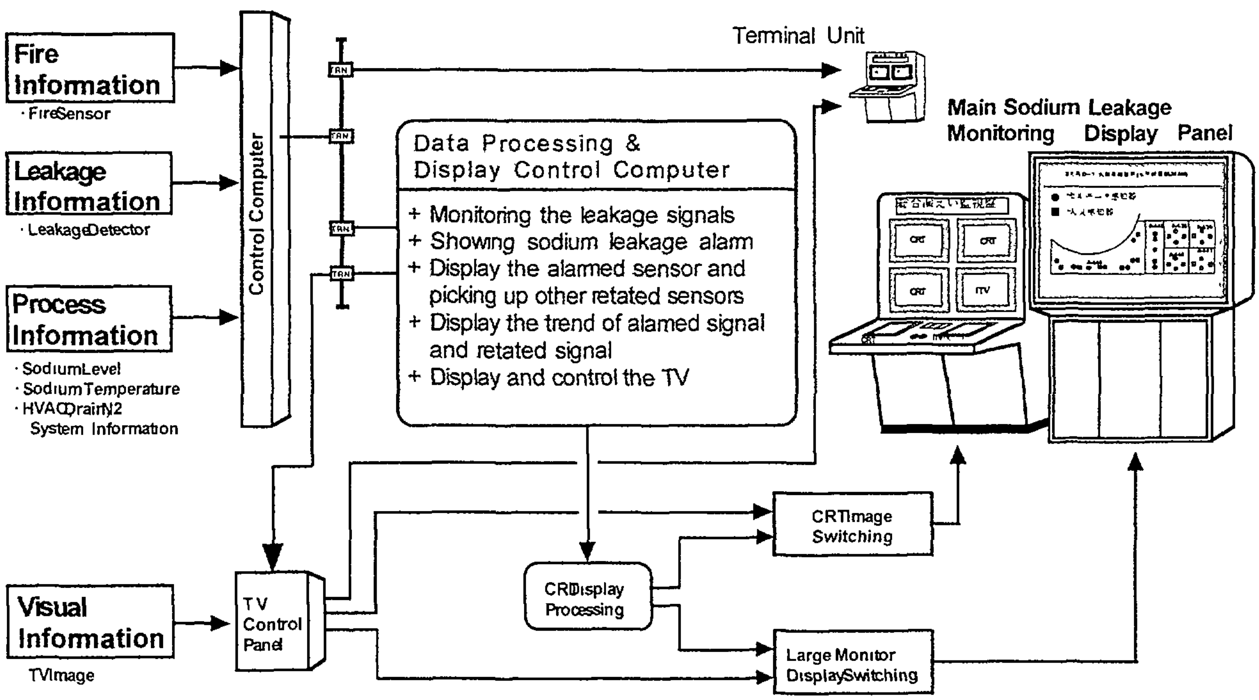

Fig. 8. Organization of integral leakage monitoring system.

By connecting the leakage monitoring system signals to the ventilation system via an interlock sequence, the shutdown of ventilation system can be achieved within two minutes after detection even for small or medium scale leakage.

(3) Drain system

To reduce the quantity of sodium leakage, the drain system was found to require two improvements. One is the addition of a new drain line pipe at the inlet of the secondary pump. The other is the replacement of all drain lines by pipes of larger diameter. In the existing system, each drain line has two drain valves in series to prevent accidental draining by single valve action error. In future so as to assure the draining, each drain line will have double drain valves in parallel. After these improvements, the drain time will be shortened from approximately 50 minutes to 20 minutes. It is estimated that it takes 40 minutes from the occurrence of sodium leakage to finish the drain for the secondary circuit. The modified drain system is shown in Fig.9.

(4) Partition of the secondary circuit room

To limit the diffusion of aerosol and the combustion of sodium more quickly, the area associated with each secondary circuit will be divided into four smaller zones. At first, more than four smaller zones were considered because partition is effective in controlling the diffusion of aerosol. However, it was found to be difficult to provide a dust-tight structure for some of the large openings between the proposed zones which allow air transfer. It was also difficult to prevent pressure in smaller zones from increasing and it is almost impossible to replace the existing ventilation ducts. Hence, four zones were adopted as the optimal solution. The openings for the cables or pipe penetrations through the wall between each zone will be improved to be almost airtight and additional isolation dampers will be fitted in the ventilation ducts between each zone. Pressure release lines will be installed to prevent an increase in internal pressure resulting from the smaller zones. The partition of the secondary circuit area is shown in Fig.10.

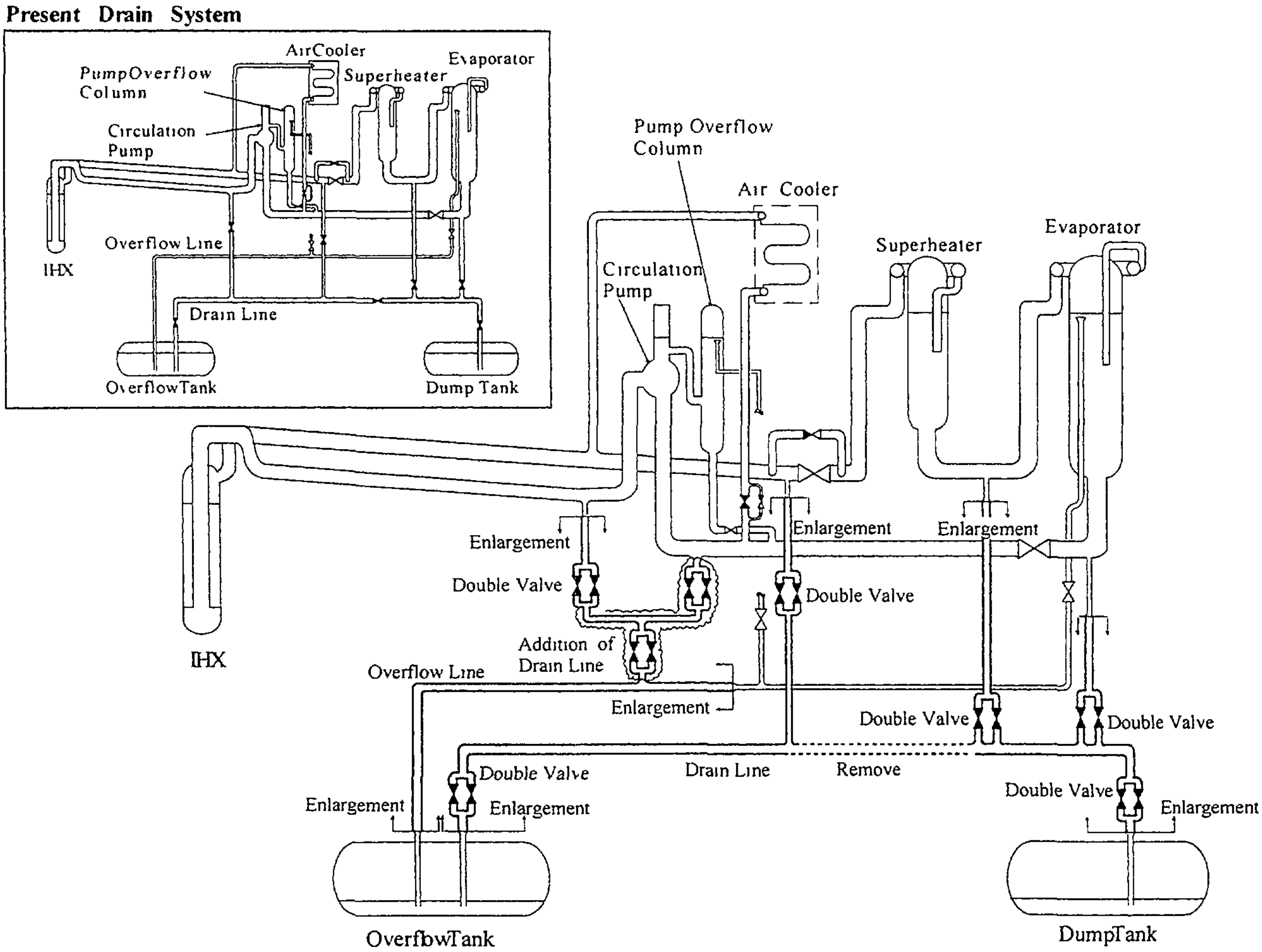

Fig. 9. Modified drain system.

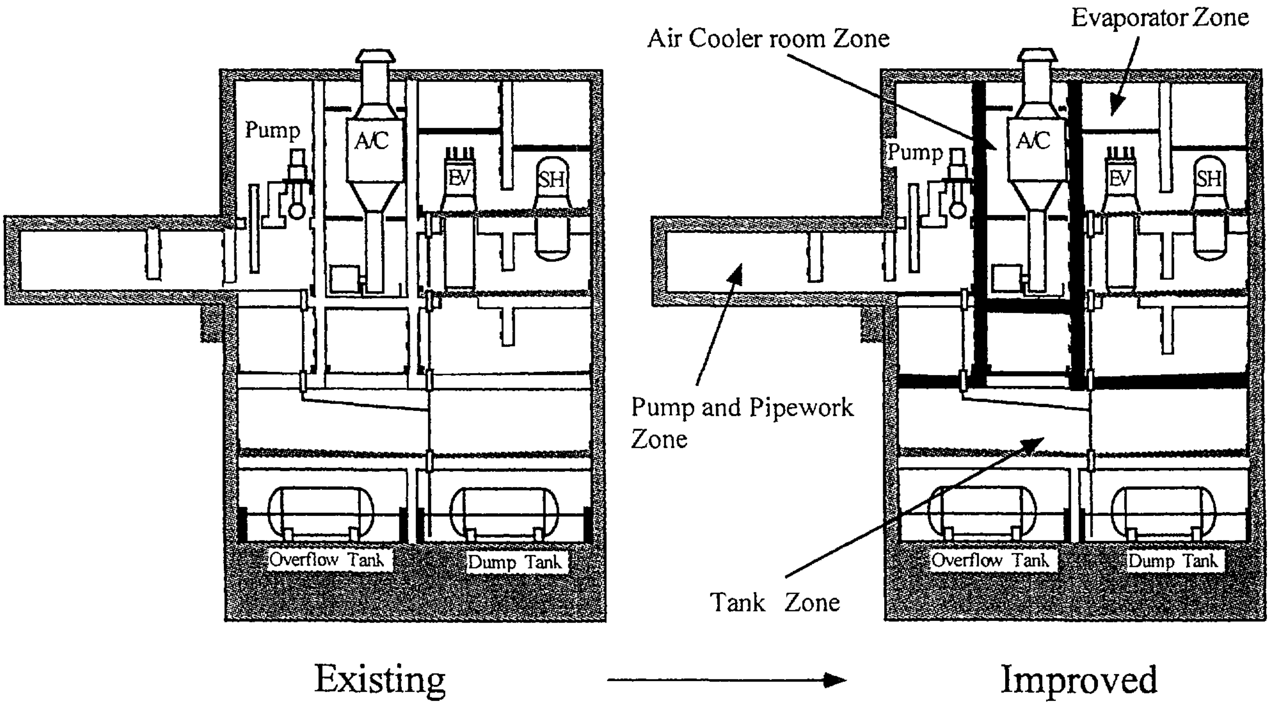

Fig. 10. Partition of the secondary circuit area.

(5) Nitrogen injection system

To extinguish the sodium fire more quickly and to prevent it from re-igniting in contact with oxygen, a nitrogen gas injection system will be installed to cut off the supply of oxygen. The nitrogen injection system is shown in Fig.11. The tanks are common to the entire system and the piping is connected to the each zone. After the personnel evacuation of the secondary circuit is confirmed, the operator will release nitrogen gas into the affected zone. According to an assessment of the relationship between nitrogen flow rate and extinguishing time, if more than 10,000 Nm^3/h of nitrogen gas is injected, a sodium fire can be extinguished within 15 minutes for any sodium leak rate.

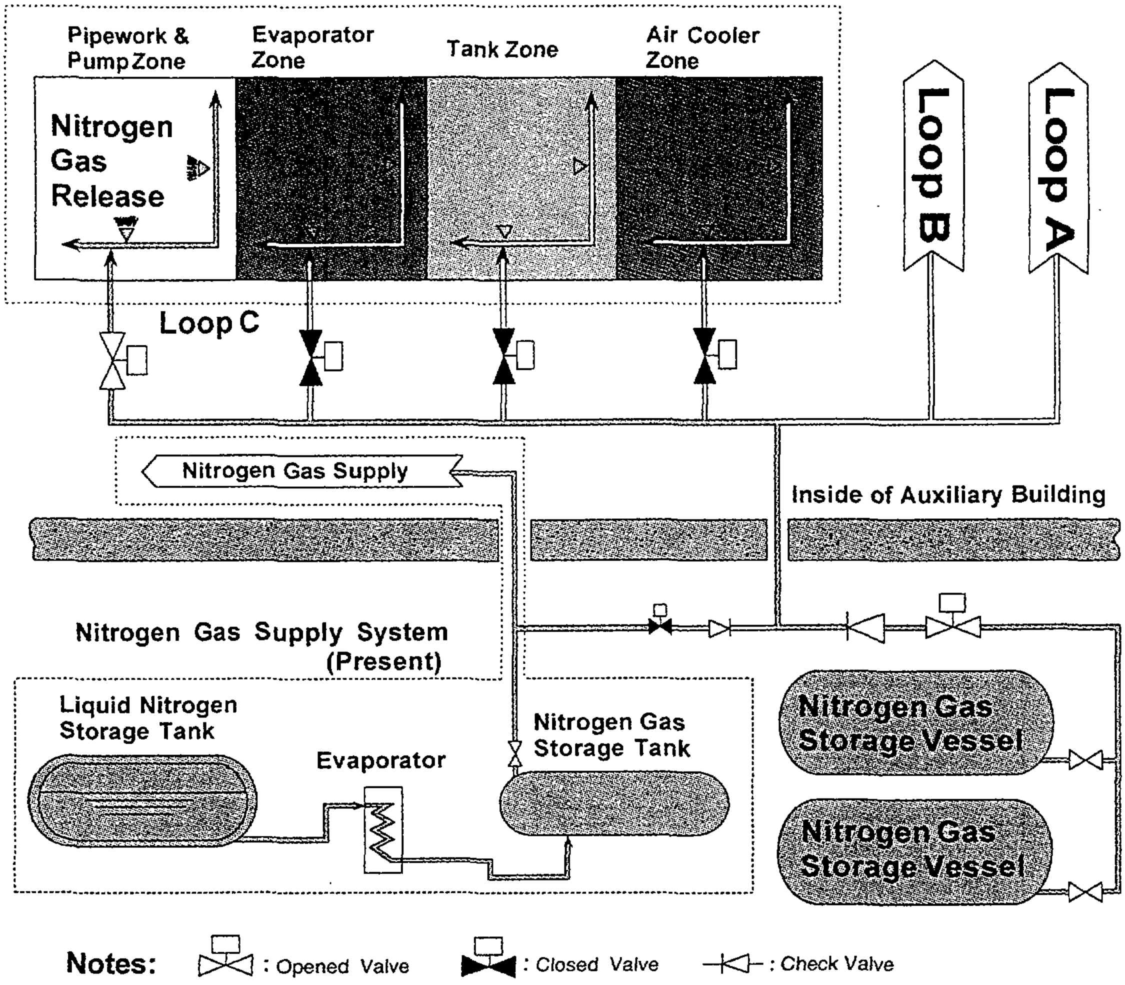

Fig. 11. Nitrogen injection system.

(6) Thermal insulation structure

To reduce the water release from the structural concrete, thermal insulation will be fitted to the walls and ceilings. This is important because if a large amount of water vapor is released from the concrete during a sodium leaks, there is an increased probability of generation of hydrogen and corrosive hydroxide in the chemical composition of the sodium debris. By using insulation, the temperature rise of the wall can be brought down to under approx. 100℃. The structure of thermal insulation is shown in Fig.12.

5. ASSEESSMENT OF COUNTERMEASURES AGAINST SODIUM LEAK INCIDENT

The effectiveness of the plant improvements described above have been assessed by the latest sodium combustion analysis code, called ASSCOPS. Typical results of analysis are as follows:

(1) Since the duration of the leakage is shortened approximately 40 minutes by the modified drain system, the total amount of leakage will be reduced to approximately 50%. The period of exposure to high temperature decreases (See Fig.13).

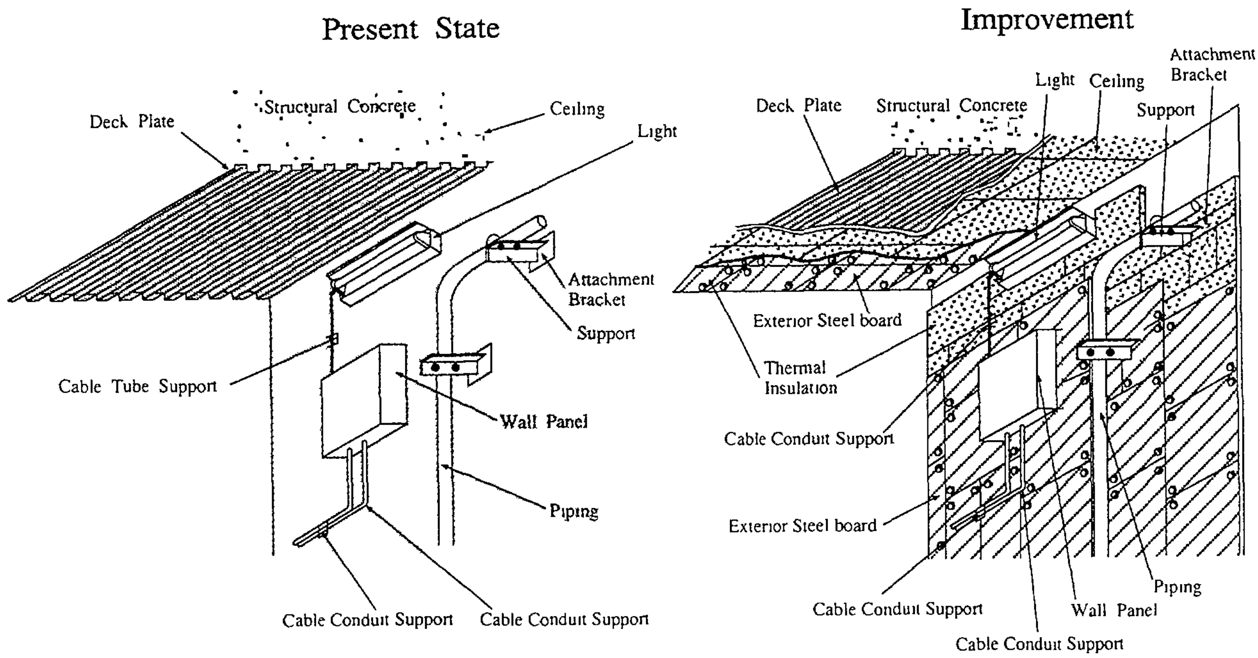

Fig. 12. Thermal insulation for walls and ceiling.

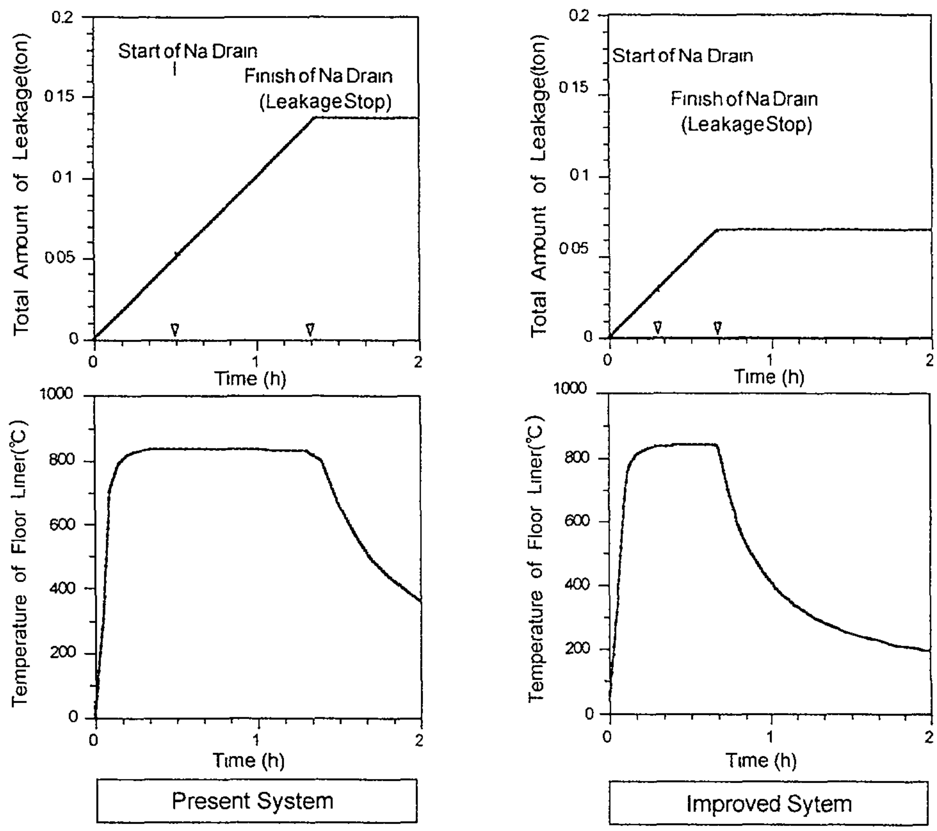

Fig. 13. Effectiveness of improvements (pipework room, leakage rate: 0.1t/h) (without nitrogen release).

(2) Since the temperature rise of the concrete is reduced from over 200 centigrade to approximately 100 centigrade by the thermal insulation on the walls and ceilings. At this temperature, the water release is considerably reduced. (See Fig.14).

(3) Since the ventilation system is shutdown automatically and the air inventory in the room is reduced by the division into four smaller zones, oxygen concentration in the combustion room and neighboring room is reduced. (See Fig.15).

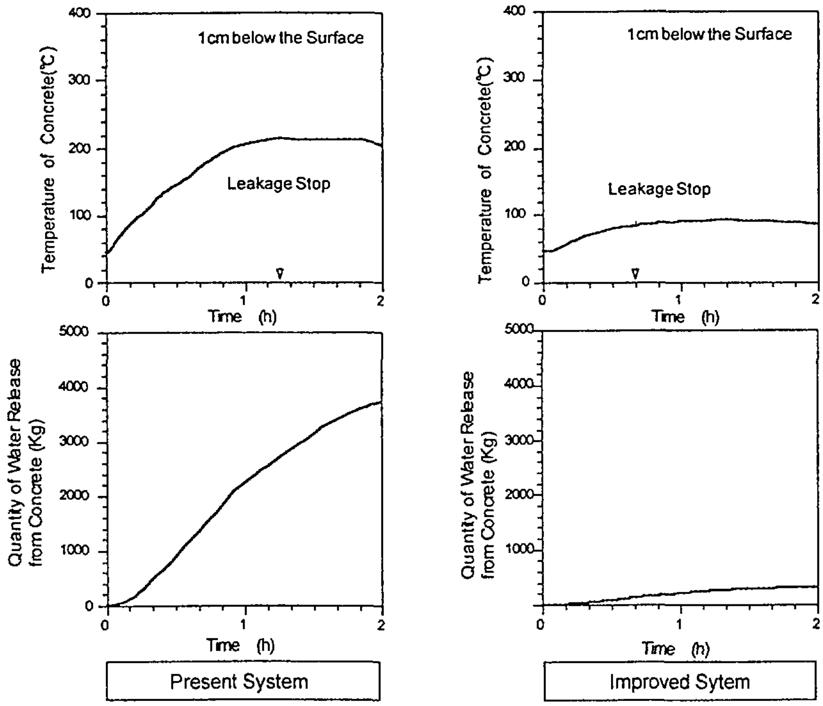

Fig. 14. Effectiveness of improvements (pipework room, leakage rate: 10t/h) (without nitrogen release).

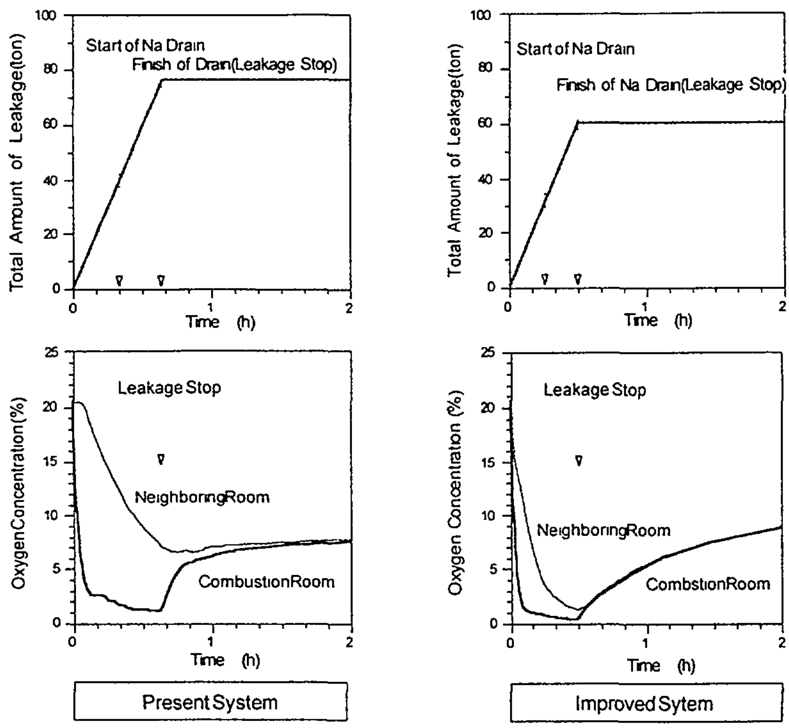

Fig. 15. Effectiveness of improvements (pipework room, leakage rate: 119 t/h) (without nitrogen release).

6. CONCLUSION

On December 8 1995, the sodium leakage from the secondary circuit occurred at Monju. The secondary sodium leaked through a temperature sensor, due to the breakaway of the tip of the thermocouple well. The reactor remained cooled and the safety of the reactor was secured. There were no adverse effects for operating personnel or the surrounding environment. The cause of the thermocouple well tube failure resulted from high cycle fatigue due to flow induced vibrations. It was found that this flow induced vibrations were not caused by well-known Von Karman vortex shedding, but a symmetric vortex shedding. The original design of the thermocouple well was reviewed. A new design guide against the flow-induced vibration was prepared by JNC. According to this design guide, the modified thermocouple wells were proposed.

The comprehensive safety review was completed in March 1998. It has been demonstrated that the following improvements are effective against the sodium leakage incidents and enhance the safety and reliability of the plant.

(1) As prevention of sodium leakage,
- A new design guide against the flow-induced vibration was prepared
- secondary circuit thermocouple wells will be replaced.

(2) As detection of sodium leakage at an early stage,
- Sodium fire sensors will be increased in number and diversified method,
- Integrated sodium leakage monitoring system will be introduced.

(3) As reduction of the quantity of sodium leakage,
- The drain system will be remolded.

(4) As limiting the diffusion of sodium aerosol and the combustion of spilt sodium,
- Ventilation system will be shut down automatically,
- Secondary circuit area will be divided into four smaller zones,
- Nitrogen gas injection system will be introduced,
- Thermal insulation for walls and ceilings will be introduced.

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SPX SIGNIFICANT EVENTS AND WHETHER IT WOULD HAVE HAPPENED ON EFR

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Abstract

In the 13 years since commissioning of the Creys-Malville nuclear power plant, exactly 100 unusual events were recorded on the French and later on the INES scales. The resultant ratio is slightly lower than the French PWR average. This is a noteworthy accomplishment, considering that the plant is a prototype, went through significant design changes, was repeatedly put to test in operating transients and, in addition, holds roughly twice the number of components as a PWR of comparable power. It may be inferred that fast reactors are not more difficult to operate than PWRs, which is also the opinion of most people having taken shifts in both types of reactors. Although Superphenix was labelled a white elephant by public opinion makers, this little known characteristic should remain part of its legacy.

In this period 7 events are registered at the level 1 of the French and INES scales, owing either to misconception, material or operational failure. At the level 2 of the scales, 2 events are registered, which is admittedly quite high. The first one was the sodium leakage from and pervasive cracking of the revolving "drum" of the fuel handling line, in retrospect the result of the choice of steel grade not fully compatible with sodium, which questions the designer's decision making process. The second level 2 event started as a massive air ingress in the primary circuit atmosphere, bringing on a pollution of the sodium up to 15,5 ppm of oxygen (although its significance in terms of corrosion was shown to be minimal). Although this event originated from a maintenance mix-up, it revealed a lack in understanding of sodium chemistry and the inadequacy of the instrumentation.

The operational feed-back of the Superphenix reactor was thoroughly combed for clues to potential anomalies by a working group comprising representatives of the operator, utility, designer and R&D bodies. All the gathered information (together with experience gained from other FBRs, most notably PFR and Phenix) was then analysed in periodic project reviews validating the European Fast Reactor project, mainly in the areas specific to FBR technology (referring to Superphenix' second level 2 event, it led to the addition of a gas analyser). That feed-back process, with complementary contributions and mutual checks of designer teams of various backgrounds, allows an optimistic view of EFR's seaworthiness.

Introduction

Between the start of fuel loading in July 1985 and the start of definitive shutdown operations in July 1998, the Superphénix reactor which powers the Creys-Malville NPP has been effected by 101 anomalies and incidents. These events are analysed based on different characteristic criteria. In particular, we noted:

- the period during which the event occurred, notably in order to highlight the role of the start-up period and to distinguish between operation and shutdown periods,

• the method of detecting the original event, in particular to highlight latent anomalies,

• the safety function affected by the event, together with deterioration or otherwise of this function,

• the original causes of the event to notably differentiate between human and material causes, but also to separate the events due to design or construction errors from those due to reactor operation errors,

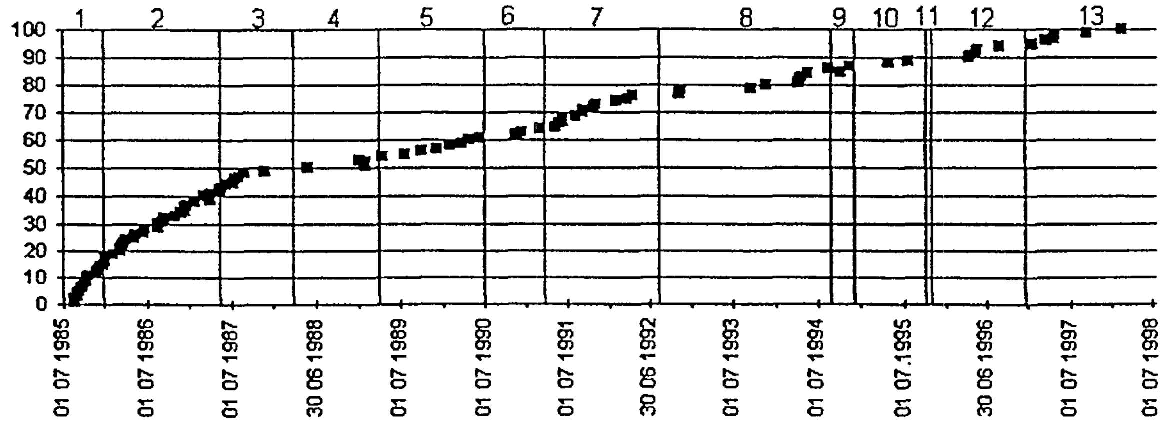
• the importance of the event as graded on the international nuclear event scale (INES).

The main results of this examination are described below and a few graphs given. The events are then listed following a thematic classification.

Lastly those events which are specific to the LMFR technology are again considered from the angle of the concepts retained for the European Fast Reactor (EFR).

1. HISTORICAL TREND

The figure below gives the breakdown of significant events over time.



The main periods of reactor operation are as follows:

1• second half of 1985: reactor loading, first criticality and zero power tests. This period runs from 19 July 1985 (start of reactor loading with fuel assemblies) to 14 January 1986 (first connection to the grid),

2• first half of 1987, limited to 25 May 1987 (reactor shutdown following on from a fuel storage drum incident) : tests on build up to nominal power,

3• May 1987 to March 1988 : reactor shutdown as a result of the drum incident (no.40)

4• March 1988 to April 1989 : reactor shutdown during the administrative procedures prior to restart,

5• April 1989 to July 1990 : operation,

6• 25 June 1900 to March 1991 : reactor shutdown following primary sodium pollution (incident 61), 7• March 1991 to July 1992 : works intended to reinforce the reactor defence against sprayed sodium fires, and administrative procedures prior to restart,

8• July 1992 to August 1994 : public inquiry into restarting by government decision

9• August 1994 to December 1994 : operation

10• December 1994 to September 1995 : reactor shutdown following on from an argon leak from an intermediate exchanger bell (event not in declaration criteria and described in annex 1)

11• September 1995 : administrative procedures prior to restart

12• September 1995 to 24 December 1996 : reactor operation

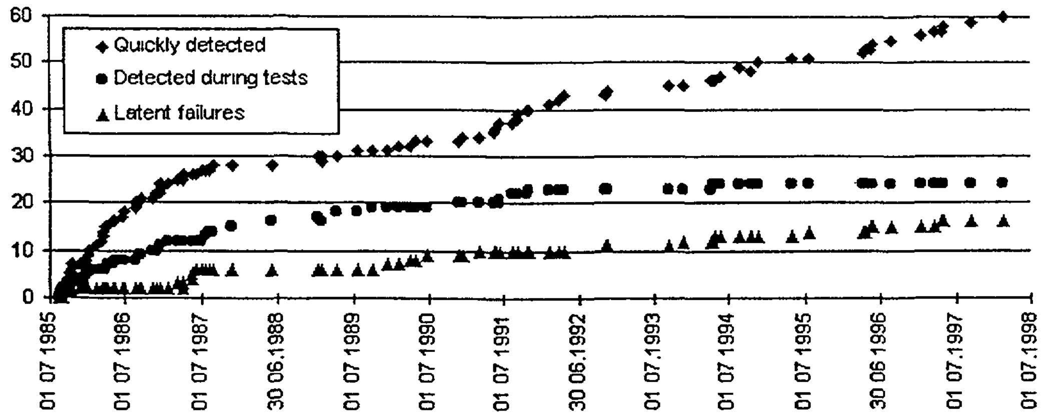
13• since 24 December 1996 : reactor scheduled shutdown followed by legal cancellation of its operation licence followed by definitive decision to shut down the reactor by the government.

The frequency of the events was 7.8 per year over the entire period considered. In this respect three periods can be identified :

- loading, first criticality and zero power testing of the reactor, reactor power build up testing and start of the drum shutdown until August 1987, during which there were 48 events, i.e. an annual average of 24 per year. Although this figure is high, it can be explained by the discovery of a certain number of design and construction anomalies on start-up testing, and by installation take-over,
- long reactor shutdown periods as a result of the drum leak (no.40), pollution of the primary sodium (no. 61), argon leak on an intermediate exchanger and cancellation of the decree authorising operation : 36 events can be listed on these shutdown periods which total 7.75 years, i.e. 4.6 events per year.
- operation of the reactor during the periods noted 5 - 9 - 12 on the figure above : 16 events over a total of 36 months, i.e. on average 5.3 events per year.

It is noted that 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 units after commercial start-up. It is remarkably low after the initial period of start-up and take-over of the installation, whether in the period of shutdown or operation.

2. DETECTION METHOD



60% of the anomalies were detected quickly. These were equipment or operating anomalies the consequences of which were felt in the short term, notably by emergency shutdowns, start-up of safeguard systems[1], control rod lowering. These anomalies continued to appear at a regular frequency until the end of the period considered.

24% of the anomalies were detected on routine inspections : essentially detected during periodic tests (of which it is one of the aims). This detection method was only effective during the first years of operation.

16% of anomalies, of various degrees of seriousness, were discovered by accident or due to their long-term consequences. They are therefore worth more detailed examination:

- no.5 : encountered on account of an inadvertent signal due to incorrect setting of a threshold. The anomaly discovered is the absence of counting systems for detecting fuel element cladding failures on one of the two reactor protection channels;

- no.12 : shutdown without inertia of the two reactor coolant pumps as a result of a loss of one of the two external power supply lines : this incident revealed a common mode defect on the 4 reactor coolant pumps resulting from a design fault in their control system after a first modification was carried out;

- no.40 : sodium leak from the fuel assembly storage drum ; this fault was compounded by not taking into account the alarms in the control room and involves the unnoticed development of a vast network of cracks in a liquid sodium tank;

- no.42 : error in calculating and setting the radioactivity thresholds leading to dome isolation ;

- no.43 : break of sodium-air fan blades discovered by a visitor, which might have however been noted during a routine test;

- no.44 : inoperability of the automatic standby diesel on a safety-related switchboard due to opening of a manually operated circuit-breaker and non-indication of the signal;

- no.57 : multiple failures of the feedwater sodium loop drainage valve positioning systems (generic fault);

- no.59 : absence of a system of auto-blocking support on the residual heat evacuation sodium circuit : the contractor although aware of the anomaly had not declared it;

- no.61 : air intake into the sodium circuit as a result of an error in the maintenance procedures : the incident also highlighted the absence of means for checking the purity of the circuit cover argon;

- no.64 : uncontrolled discharge of slightly irradiated metal waste : the normal inspections did not detect their presence at the site exit;

- no.77 : inoperability of automatic standby by one of the two diesels on a train due to a maintenance error on the electrical batteries;

[1] decay heat removal, reactor confinement, diesel generators

- no 80 : inoperability of the sodium leak detection alarms in the components submerged in the reactor coolant system due to erroneous determination of the signalling thresholds;

- no 83 : inoperability of the sodium leak detection on a feedwater loop storage tank due to inaccurate transmission of data between workers;

- no.89 : definitive loss of a sealed radioactive source (1 μCi) used to monitor the accesses, thrown into a workshop bin by a repairer who was unaware of its nature;

- no.92 : melt of the liquid metal seal on a rotating plug with the reactor critical due to an error on the operation sheet;

- no.97 : inoperability of the sodium leak detection on several sections of a feedwater sodium loop auxiliary circuit due to an error in an operating procedure.

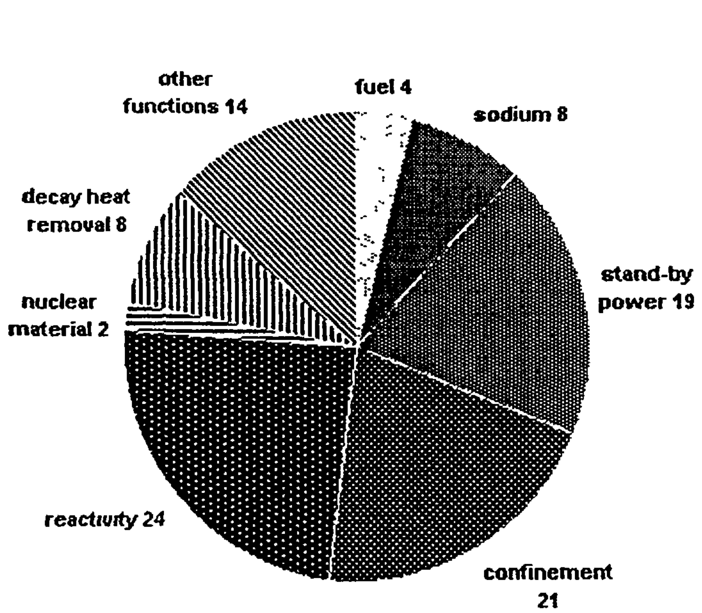
In all, seven of these events involve a design error, three an erection or equipment adjustment defect and eleven indicated faults in the plant operation organisation.

3. SAFETY FUNCTIONS INVOLVED IN THE EVENTS

3.1 Functions generally involved

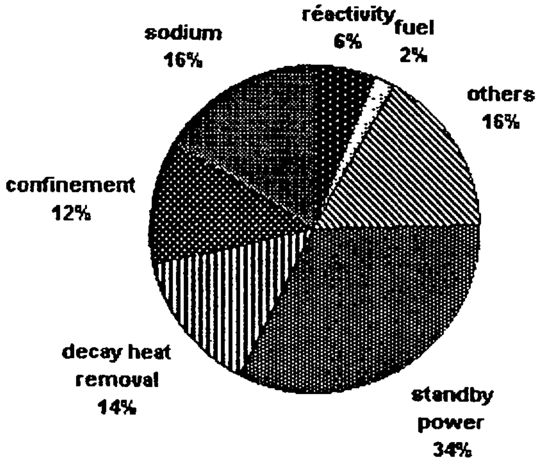
The events are here classified according whether they concern the fuel, radioactivity control, residual heat evacuation, emergency electricity supplies, confinement, control of nuclear material or other functions. In this paper we will first of all examine only the functions involved even if the event does not indicate any deterioration of this function.

We observe that almost two thirds of the anomalies involve the reactivity control function (command of control rod positions), the containment (notably dome isolation and reactor building ventilation) and standby supplies (diesel generator sets, batteries and switchboards that they supply). The anomalies concerning sodium only involve eight events



3.2 Impaired functions

If we exclude a half (51) of events which have not led to a deterioration (real or potential) of a reactor safety function, it becomes obvious that the largest number of really significant anomalies is, like on other installations, associated with the loss of standby electricity supply (which did not have important consequences for Superphenix since there are 4 identical sets), but also that the second cause of safety function deterioration is linked to sodium. This illustrates that the two, sodium-related, incidents which were classified level 2 on the INES scale (nos. 40 and 61) were not isolated events.



Note : in this paper we have covered deterioration of a safety function and not the consequences for safety : in general, for a deterioration of a safety function to impact the safety of the installation, this function must also be required. If we adopted this criterion, the relative importance of the anomalies associated with the sodium and its cover gas would further increase since these must be permanently confined owing to :

- their contribution to the transport of fission and activation, volatile or gaseous, products;

- the risks they engender (sodium fires, anoxia).

3.3 Absence of a safety function degradation

In more than half the events declared, no reactor safety function was degraded. This category includes:

- inadvertent emergency shutdowns, dome isolation, ventilation configuration of the reactor in the reinforced confinement position,
- lowering of control rods,
- three cases of non-compliance with procedures for checking the protection functions,
- three events involving fuel handling,
- two events involving uncontrolled exit of radioactive matter involving a very small amount of activity,
- two cases of human errors making the equipment assuring a safety function inoperable without operator action,
- one case of inadvertent start up of a residual heat evacuation circuit.

This large number of events non-directly significant for reactor safety could lead to the conclusion that the criteria which cover the declaration of significant events are excessively strict. However, these events reveal problems of the plant operation organisation.

3.4 Construction defects without a direct impact on safety

Event no.63 would not be included in the list of those associated with a loss of a safety function if its initiator (collapse of a part of the turbine hall roof under snow weight) had not caused a long loss of one of the external electrical sources.

The argon leak from an intermediate exchanger bell, an event which was outside declaration criteria, caused 8 months' shutdown.

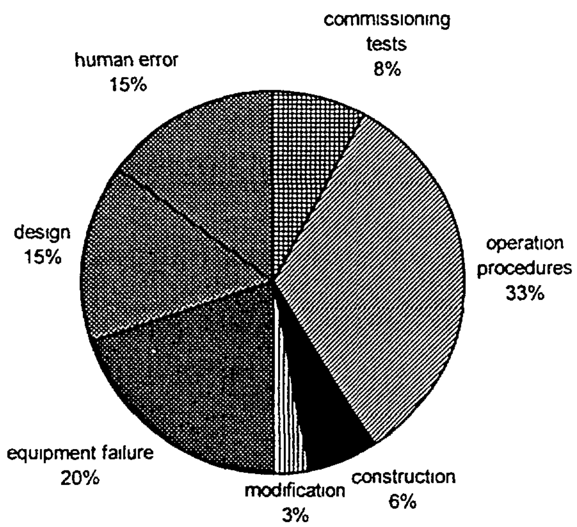
These two events are directly linked to malpractices, undetected through the quality assurance system. The management structure chosen for building the plant (mainly in the view of distributing the contracts between participant countries) rendered the quality control probably more difficult. It should be noted yet that, as a whole, building deficiencies contribute only marginally to the number of significant events.

4. CAUSES

The causes have been listed according to their origin as follows :

- installation design,
- installation construction,
- execution of start-up tests,
- modifications executed since start-up,
- methods associated with normal operation of the installation,
- equipment failure,
- human errors during execution of a task.

The following graph gives the global percentages, while the table indicates, for each given period, the annual frequency of events.



Period (operation shaded)	1	2	3	4	5	6	7	8	9	10	11	12	13
Equipment	**6**	**5.3**	0	2.2	1.3	1.3	1.5	1.8	3	0	0	0	0
Design	**8**	3.8	1.2	1.1	2.5	0	0.8	0.9	0	0	0	0	0
Construction	**4**	0.8	0	0	2.5	1.3	0	0	0	0	0	0	0
Commissioning	2	**3.8**	2.4	0	0	0	0	0	0	0	0	0	0
Modification	**4**	0	0	0	0	0	0	0.9	0	0	0	0	0
Operation	**8**	2.3	3.6	2.2	1.3	1.3	**5.3**	3.4	3	2.7	0	3.2	1.7
Human error	**4**	3	0	0	1.3	0	0.8	0.9	3	0	0	0.8	**3.4**

This classification is partly subjective since in most cases there is not a single cause of the incident. Nevertheless, out of all the events occurring at Creys-Malville, the following trends can be identified:

- Events attributed to the design, the construction of the installation, and its commissioning are preponderant at the start of the plant life, but only represent 29% of the total. The phase of installation take-over has therefore correctly fulfilled its role.

- Equipment failures and human errors together caused one third of the events. Although precautions (as a matter of quality assurance) can be taken in this respect (selection of equipment, preventive maintenance, training, preparation for works, etc.), these two types of events are nevertheless difficult to avert totally.

- The definition of modifications on the equipment brought about 3 events:

  • no.4 : deformation of a new fuel assembly (before loading) : the modification executed as a result of incident no.1 (addition of a pointer to the biological protection hood for new fuel sub-assemblies) have caused this incident,

  • no.12 : no-inertia coast-down of reactor coolant pumps : a modification to avoid the alternator supplying their power operating in motor mode had been previously implemented on the four drive sets by using information supplied by a non-emergency supplied electrical source,

  • no.80 : inoperability of the sodium leak protection alarms in the gas spaces : the thresholds validating the alarms had been set erroneously by the team responsible for designing the modification and the error was detected thanks to the critical observations of an operator.

  The increasingly strict approach to the management of modifications has, since 1994, avoided such events recurring in spite of the many modifications made.

- Lastly, 33% of events can be attributed to the methods or the usual organisation of the reactor operating staff, including maintenance. Out of these, 7 occurred during reactor shutdown for purification of the sodium. However, there is no apparent correlation with the reactor status.

These events, which are not, generally, specific to the Fast Reactors technology, could be reduced in frequency thank to the progress made in matters of man-machine interface and equipment testability.

5. SCALE OF SERIOUSNESS

All the significant events occurring on Creys-Malville have been classified using the criteria on the INES scale of nuclear events. It must be observed that for events occurring prior to application of the French scale, the subsequent classification cannot take account of the discussions which have taken place in certain limit cases. However, this classification benefits from all the experimental classification work which had been carried out by IPSN and EDF when defining the French scale.

The following 2 incidents characteristic of the fast reactors are classified level 2:

- sodium storage drum leak (no.40)

- reactor coolant system pollution (no.61).

These two incidents fully reflect the prototype nature of the installation.

The following six events are classified level 1:

- no.12 : emergency shutdown with loss of inertia on two reactor coolant pumps
- no.18 : heating of a fuel assembly,
- no.56 : fall of a temporary lifting crane,
- no.63 : collapse of the train A turbine hall roof,
- no.88 : simultaneous opening of several containment barriers,
- no.92 : fusion of the liquefiable metal seal on the large rotating plug, reactor critical.

Three of these events are associated with erection of the installations, the other three with operation.

The analysis of these 8 events according to the criteria examined previously are given in the following table.

It will be observed that none of these events did endanger reactor safety. On the other hand one (no.40) definitely compromised any hope to cover the plant operating costs, for the repair could not restore hot irradiated fuel storage capacity and compelled the operator thereafter to wait for the decay power to subside before unloading the core, meaning long outages.

Four events brought about long idling periods (nos 40, 61, 63 and the leak of the intermediate heat exchanger bell) and contributed to alienate decisive sectors of opinion, including in the nuclear field.

The description of these eight events is attached in the annex 1.

NO	PERIOD	DETECTION	SAFETY FUNCTION	CAUSE
40	2	late	sodium confinement	design
61	5	late	cover gas integrity	design
12	1	late	core cooling	modification
18	2	commissioning test	core cooling	operation
56	5	equipment test	none	construction
63	6	instantaneous	none	construction
88	10	instantaneous	reactor confinement	operation
92	12	late	reactor confinement	operation

The two incidents (nos.40 and 61) which originated in a design failure have been taken into account in the EFR project as follows:

- By the choice of material used for vessels (which implies that a thorough explanation has to be given to incident 41, including its reproduction in laboratory conditions), by anchoring the safety vessel in the vessel pit, itself constructed in sodium-refractory concrete,
- By permanent inspections of the chemical composition of the gas cover.

6. GROUPING OF THE SIGNIFICANT EVENTS

The significant events occurred at Creys-Malville can be grouped in coherent families, each incident being referred to only once according to its most important criterion.

a - Justified implementing of a safeguard system (residual heat evacuation circuits, dome isolation, diesel generator sets)

no.6 : destruction of a feeder circuit breaker to the 6.6 kV switchboard

no.54 : voltage loss a 6.6 kV switchboard

no.63 : collapse of the turbine hall roof, train A

no.81 : loss of external electricity supplies to train B

no.93 : tripping of stage 1 dome isolation

In the first four cases this involved start-up of one or two generator sets.

b - Sodium leak

no.40 : sodium storage drum leak

no.60 : sodium leak on a feedwater loop auxiliary circuit tee junction

no.65 : sodium leak on the plugging indicator of a residual heat evacuation circuit

This type of incident is specific to fast reactors and justifies them being placed in a separate category. It will be observed that the risk involved in sodium fires motivated the regulating authority to demand, specifically after the Almeria accident, that a sprayed sodium fire resulting from a complete and sudden rupture of a main secondary pipe be taken account of.

c - Physical phenomena leading to reactor shutdown (emergency or intentional shutdown) :

no.18 : fuel assembly overheating

no.24 : emergency shutdown due to temperature fluctuations

no.55 : emergency shutdown due to reactor coolant pump stop

no.61 : reactor coolant pollution

no.79 : failure of a reactor coolant pump coupling

d - Failure of the reactor protection system (bringing into question its operation when solicited)

no.2 : non-lowering of three complementary shutdown system rods

no.5 : absence of train B cladding break detection channels

no.62 and no.69: no-voltage relay anomalies

The two first events are associated with the reactor start-up tests. However, the two last ones, although they initiated an emergency shut-down thank to the "fail-safe" design of the reactor protection system, present the problem of its reliability.

e - Failure of another safety function (evacuation of residual heat, containment)

no.9, no.26 and no.46: diesel generator sets heating

no.12 : non-inertia shutdown of reactor coolant pump

no.13 : non-coupling of a pair of diesel generator sets

no.15 and no.42: poor adjustment of the dome activity threshold

no.36, no.47 and no.50: loss of battery capacity

no.37 : no-voltage on 6.6 kV standby switchboard

no.43 : break of sodium-air exchanger fan blades

no.44 : voltage loss on emergency-supplied 6.6 kV switchboard no.49 : non-start-up of a diesel generator

no.53 : non-closure of a steam generator water valve

no.57 : anomalies on feedwater loop drainage valve

no.59 : absence of a self-blocking support on a residual heat evacuation circuit

no.67 : inoperability of a sodium-air exchanger

no.68 : inoperability of a diesel generator

no.71 : loss of train B standby raw water circuit

no.73 : inoperability of a diesel generator

no.77 : inoperability of two diesel generators on one train

no.80 : inoperability of sodium leak detection alarms in gas spaces immersed in the primary sodium

no.82 : through crack on the balancing line of secondary loop rupture discs

no.83 : inoperability of sodium leak detection on the storage tanks

no.84 : inoperability of a residual heat evacuation system

no.87 : inoperability of a secondary loop argon sweeping circuit

no.88 : simultaneous opening of several containment barriers

no.92 : liquefiable metal seal fusion on large rotating plug

no.94 : loss of intermediate containment integrity by gas activity control circuit

no.95 : inoperability of diesel set

no.96 : inoperability of chilled water production system

no.97: inoperability of sodium leak detection on 8 sections of a secondary loop auxiliary circuit

no.98 : loss of slab cooling

no.100 : shutdown of reactor coolant pump

These events are significant of potential or effective deterioration of a safety function (evacuation of power in most cases).

f - Events concerning handling (of assemblies or components) :

no.1 : drop of a new fuel assembly into its pit

no.4 : deformation of a new fuel assembly no.25 : break of an observation window on the small cask

no.56 : fall of a temporary lifting crane

The separation of the installation into two parts, reactor and handling, together with the specificities of this type of events, allows this category, which is relatively diverse as to the events it covers, to be defined.

g - Emergency shutdown orders or inadvertent rod lowering (without anomaly on the reactor protection system) :

no.7 : shutdown by the power/reactor coolant flow protection

no.8 and no.11 : emergency shutdown tripped by the train B cladding break detection channel

no.10 : inadvertent manual emergency shutdown

no.14 : emergency shutdown tripped by the seismic channels

no.16 and no.17 : emergency shutdown tripped during works on the reactor protection system

no.19 : lowering of a control rod

no.20 : lowering of 3 rods on the diverse shutdown system by permanent supply cut off

no.21 : lowering of 3 rods of the diverse shutdown system during a fast shutdown

no.22 : emergency shutdown by the neutronic measurement channels after a fast shutdown

no.23 : lowering of 2 control rods

no.27 : emergency shutdown by non-inhibition of the neutronic power low level measurement channels

no.28 : emergency shutdown by the neutronic measurement channels on movement of a physical sensor in the core

no.29, no. 30 and no.32 : lowering of one or two diverse shutdown system rods

no .33 : emergency shutdown by the neutronic measurement channel after a fast shutdown

no.34 : lowering of no.2 main control system control rods

no.39 : emergency shutdown order by the power/reactor coolant flow protection

no.41 : emergency shutdown due to works on the computer monitoring the heating of train B reactor

no.51 and no.52: emergency shutdown due to a fault on a power/reactor coolant flow protection

no.85: emergency shutdown without initiator

Generally, these events translate faults on materials or organisation design (man-machine interface) but do not have physical consequences for installation safety.

h - Inadvertent start-up of a safeguard system :

no.31, no.35, no.38 and 45: dome 2nd stage isolation

no.48 : dome 2nd stage isolation

no.58 : dome 1st stage isolation

no.66 : inadvertent start-up of a residual heat evacuation circuit

no.70 and no.72 : dome 1st stage isolation

no.74 : dome 2nd stage isolation

no.75 : dome 2nd stage isolation

no.76 : dome 1st stage isolation

no.78 : dome 2nd stage isolation

no.99 : excessive pressure drop inside the reactor building

Same comment as for the previous category.

i - Non-compliance with procedures (tests guaranteeing correct operation of systems associated with reactor safety)

no.3 : incorrect estimate of the critical mass

no.86 : non-execution of control rod translation force measurements

no.90 : operation of 2 control rods without recording the translation force

no.91 : non-compliance with the delays for executing periodic tests of fire detection

j - fault in the management of nuclear matter

no.64 : uncontrolled discharge of slightly irradiated steel

no.89 : definitive loss of a sealed 1 μCi radioactive source

7 SIGNIFICANCE OF EVENTS FOR THE FAST REACTOR TECHNOLOGY

Out of 101 events considered (including the intermediate exchanger bell argon leak) :

- **66 can be considered non-specific to fast reactors, i.e** :

. in view of their causes (but not necessarily their effects), they could have occurred on PWR reactors,

. they can be prevented by improving the quality approach in terms of the design or procedures and/or by generalising the recent approaches relative to the design of control and instrumentation systems and the man/machine interface (testability, ergonomics).

In particular, they consist of faults on the electrical supply, diesel generators and batteries, faults on electrical connections, routing errors, in-service works or calibration errors.

One could be tempted to mitigate this favourable observation (2/3 of events are not specific to fast reactors) by the fact that the number of components is twice more and that certain circuits are more complex (primary and secondary gas circuits) as compared to PWRs.

The overall results do not however confirm this reservation which would imply that, for an equivalent design (control and instrumentation, man/machine interface), the probability of failure should be identical for individual components and thus roughly double for all the systems and functions.

This is probably due to the fact that, apart from the control of reactivity and the protection of the steam generators, the accidental transients on fast reactors are much slower than those on the PWRs and that there are therefore fewer systems and actuators requiring both very fast and very reliable start-up.

- **35 can be considered specific to the fast reactors**. Among these a distinction must be made between :

18 SPX specific events which correspond to :

. steps not adopted for EFR (biological protection bell, dome, intermediate exchangers bell, ensuring leaktightness between hot and cold headers),

. procedures improved and validated on SPX (e.g. lowering of clusters before changeover to auxiliary motor) : these procedures must obviously be retained or adapted for future fast reactors.

- **17 "SPX & EFR generic events"** (see annex 2) which correspond to :

. non-quality of the equipment or fabrication:

plug left in an assembly leg,

fan blade break,

primary pump coupling break,

. system (control and instrumentation) or component design improved then validated on SPX :

reactor coolant pump control and instrumentation,

cask observation window design,

control and instrumentation on the fast decompression isolation sequence,

design of plugging indicators preheating,

. points on which studies are required :

measurement of the primary flow and tripping of associated protections (2 events),

reactor measurement sensitivity to cable impact due to very currents (2 emergency shutdowns by the train B cladding break detection channel),

fluctuation of the temperature at the sub-assembly outlets (first fertile row),

. generic technological aspects :

design of a main vessel (storage drum),

conditions of hot sodium/cold sodium mixture,

checking of the primary sodium impurity level,

design of dead legs subject to sodium aerosols

design of the cover gas circuit.

8 THE PROJECT REVIEWS OF THE MAIN SYSTEMS OF THE FBR TECHNOLOGY BASED ON OPERATING FEEDBACK

Since 1995, a systematic approach to check the design of the EFR systems has been carried out in the framework of the Project Reviews of the main systems and components based on operating feedback.

These PR have concerned :

- Control rod drive mechanisms
- Main sodium pumps (primary and secondary)
- Intermediate heat exchangers
- Primary sodium auxiliary systems
- Primary argon auxiliary systems
- Sodium circuits
- Steam generators

Each PR has been organised practically in the same way, as follows :

- Introduction, including design specifications and safety requirements of the systems surveyed.
- A survey of the characteristics of the process fluids (if appropriate).
- A comparison between EFR design and the existing (or having existed) designs : mainly PFR, PX, SPX.
- Recommendations, proposals for EFR (and future) design.
- Conclusions.

In order to prepare each PR, certain specific studies have been carried out :

- Comparison of EFR design to existing designs.

- Operating feedback of the same or similar systems from plants in operation (or having been operated).

- Main characteristics and the main feedback of the process fluids.

Each PR has given rise to a synthesis. Hereafter are indicated only those of the conclusions which derived from the feed-back of Superphenix.

PR on control rod drive mechanisms

DSD mechanisms

These mechanisms are mainly derived from the Superphenix DSD mechanisms, so it has to accommodate the results of in-service inspection and maintenance works on the DSD mechanisms and the DSD absorber rods at Superphenix.

PR on Intermediate Heat Exchangers

The conclusions are mainly that EFR intermediate heat exchangers design globally integrates features from Superphenix intermediate heat exchangers, the design of which already benefits from experience gained at Phenix.

PR on primary sodium auxiliary systems

Concerning the "principle design sheet", there are no particular remarks from the operating feedback : integrated electro-magnetic pumps at Superphenix are reliable; an external circuit makes maintenance operations easy.

Adding to the system an integrated plugging meter, improved and optimised as regard to the Superphenix one (200 l/h instead of 50 l/h), might be considered.

This way the two main functions of oxygen content measurement and cold trapping will be independent.

PR on primary argon auxiliary systems

Operating feedback suggests having the working pressure range for the circuit as low as possible. This is easily attempted with a so-called constant pressure circuit (such as Phenix and Superphenix). There is also a concern to limit the sodium leak hazards from the reactor upper closure by geyser effect ; it needs a low argon cover gas pressure. Moreover, it must be possible to lower the relative pressure of the cover gas circuit near zero, especially during refuelling and maintenance and primary components handling, in order to avoid (or to substantially limit) argon leaks to the outside.

Hence it has been suggested that the EFR argon reactor cover gas circuit design return to a (almost) constant pressure circuit, bringing benefits from operating feedback.

PR on sodium circuits (secondary, auxiliary, decay heat removal)

Phenix and Superphenix secondary circuit designs are simple; filling and draining are easy. Moreover, at Phenix, most of nozzles of small piping have been eliminated from secondary pipe-work during renovation works. It follows that the need to have a high point vessel, and a degassing circuit (as actually in design) has to be considered.

In the current EFR design, the slope of the secondary piping is towards the intermediate heat exchangers. This arrangement, which does not allow complete draining of secondary circuits at intermediate heat exchangers, must be avoided (slope to intermediate heat exchangers for the cold leg and from the intermediate heat exchangers for the hot leg as at Phenix and at Superphenix).

Based on French operating experience and usual operating procedures, it has been suggested that the main sodium pumps of EFR Consistent Design be equipped with pony motors in order to assure sodium forced convection in secondary circuits in any configuration.

As the main pollutant of the secondary circuit is hydrogen, the secondary auxiliary circuits have to be optimised in order to measure hydrogen contents and in order to trap hydrides; it would be useful to have data about tritium diffusion from the primary circuit.

PR on steam generators

From the operating feedback from British, French and Russian steam generators which have equipped sodium-cooled fast reactors, it can be confirmed that it is possible to manufacture and to operate large and reliable 'integral once through' steam generators without major difficulties, and respecting high Quality Assurance rules.

The helical steam generator design can be considered favourably based on the Superphenix operating feedback,

The selection of a helical steam generator as a fallback option for EFR is therefore fully justified .

9. CONCLUSION

The frequency of the significant events which occurred in the Creys-Malville facility (with the Superphenix reactor), equal to 7.8 per year for the entire period considered including the start-up period – i.e. 6.8 per year when only considering the period subsequent to first coupling to the grid (1986), is similar to the one recorded for the PWR reactors which is about 8 per unit and per year.

Around half the significant events had no consequence, whether effective or potential, for reactor safety.

On the other hand, it is observed that in 16% of the events, the installation was in a degraded condition without the operator noticing this. It may be noted that seven of these events occurred after June 1990, date of the air ingress detected late in the primary circuit, i.e. around 1 per year or 18% of the events occurring during this period.

48% of the events can be attributed to an imperfect man-machine interface, to the methods, usual organisation of reactor operation or human failures. These factors represent all the causes of events occurring during the last three years, notably the period of continuous operation of the reactor in 1996.

It is observed that the significant events which involved the electricity supply are in a majority. This observation underlines the importance which has to be attached to the integrity of the electricity supplies.

Lastly, the events associated with the presence of sodium are the second largest category (16%) among those which affect the safety functions and the largest in the length of resulting plant idling and in seriousness (two events classified level 2 on the INES scale). This means that contrary to the opinion commonly held during reactor design, the fast reactor technology was not totally mastered.

If one adopts the point of view of future fast reactors, particularly the EFR project studies, the analysis shows that, out of 101 events :

66 may be considered as "non-LMFR specific" i.e.

from their causes they might as well happen on PWRs;

they can be prevented by a strengthened approach of quality assurance in design or procedures and/or by an extensive use of recent concepts relative to instrumentation and control systems and man-machine interface (testability, ergonomics).

18 events correspond to specific Superphenix features not retained for EFR, or to procedures improved since the event occurred.

17 "generic" events correspond either to designs (of systems, equipment) which may have be improved and validated on Superphenix, or to special technical notions, a few of which still require some development.

A systematic review of the operation of Superphenix allowed to validate several concepts which were then retained for the EFR project in the most characteristic fields of liquid metal circuit technology: control rods, pumps, intermediate exchangers, sodium and argon auxiliaries, sodium circuits and steam generators.

The experience of Superphenix thus furthered the technical quality of the next fast reactor generation.

Annex I

DESCRIPTION OF EVENTS CLASSIFIED LEVELS 1 AND 2 ON THE INES SCALE FOLLOWED BY' DESCRIPTION OF THE ARGON LEAK FROM AN INTERMEDIATE EXCHANGER BELL

EMERGENCY SHUTDOWN TRIPPED BY LOSS OF 225 KV AUXILIARY POWER

No.12 Date 22 November 1985 at 23 00

1 - NATURE OF THE INCIDENT

Emergency shutdown tripped by loss of 225 kV auxiliary power

2 - SEQUENCE OF EVENTS

Reactor was critical at a temperature of 195°C The reactor coolant pumps were operating on the main motor on 110 rpm

Train A was supplied by the 400 kV network, the 225 kV auxiliary line was available

Train B was supplied by the 225 kV auxiliary line, the supply by the 400 kV network being locked out for maintenance works and elimination of the neutral switch

At 23 00 58, a defect on the 225 kV auxiliary line caused the two circuit breakers on line LGR A and B to open

At 23 01 03 the min voltage threshold on the 6 6 kV LGA C and D and LHA C and D switchboards on train B was reached tripping emergency shutdown AU2 and start-up of the two train B diesel generators

Train A remained supplied normally by the 400 kV system

At 23 01 05, reactor coolant pumps RCP C and D went into reverse rotation as a result of the loss of inertia This loss of inertia is due to opening of the self-exciting circuit breaker 03 JA of the variable speed alternators supplying the motors of these two pumps

Secondary pumps BCS C and D were driven by the auxiliary motors

At 23 10, procedures I 14 C and D applicable in the event of loss of non-emergency supplied switchboards LGA C and D came into force

Closure of the obturators on the two reactor coolant pumps C and D took place at 23 28

At 01 09, dispatching indicated that the 225 kV line would be inoperable at least for the night It was therefore decided to re-supply train B as quickly as possible through the 400 kV line (locked out)

The end of lockout of the 400 kV train B took place at 02 00

The instrumentation and control re-qualification tests for the train B power evacuation system began at 4 00

Re-energising the 400 kV train B transformer took place at 07 00

Train B switchboards were re-supplied from the 400 kV network at 08 00 and the train B diesels were stopped

3 - STEPS TAKEN

Application of incident instruction I 14 loss of non-emergency supplied switchboards

Re-supplying in the shortest possible time of the train B switchboards by the 400 kV system

Drainage of the sodium circuits since preheating function not emergency supplied (MAS 0)

Cancellation of the planned tests (reactor critical) pending the explanation and the elimination of anomalies having caused pump reverse operation

On 23 November 1985, a helicopter search was initiated to detect the line fault This fault was due to the accumulation of frost on the cables (one phase in contact with a tree)

4 - NO-INERTIA STOPPING OF REACTOR PUMP

4.1 - STATUS OF THE SYSTEM AT THE ORIGIN

Since excitation of the alternators of the pump drive sets, compound type, can be interrupted by circuit breakers RCP* 03 JA and straps had been installed so that these circuit breakers could open as soon as the 6 6 kV supply of RCP* 01 MO tripped out, the system for progressive coast-down of the primary coolant pumps was not operational (common mode defect)

4.2 - FIRST MODIFICATION

Relaying modifications were carried out to overcome the anomalies observed during the tests non-opening of RCP* 03 JA on loss of voltage, motorisation of the alternator on take-over by the auxiliary motor These modifications can be summarised as follows

- installation of a section to take into account "auxiliary motor tripped in", instead of a measurement of the motor charge current, which is not very reliable due to the small difference between the charge and no-load currents,

- to avoid operation of the alternator in motor mode, opening of RCP* 03 JA is controlled by the following conditions

- RCP* 01 MO main motor stopped,

- and, either alternator speed less than 45 rpm, or auxiliary motor RCP* 02 MO started

The "main motor not started" and "alternator speed less than 45 rpm" conditions also open RCP* 01 JA (which leads to opening of RCP* 03 JA)

4.3 - INCIDENT

On the incident, RCP* 03 JA opened since the following conditions were met

- speed less than 45 rpm (threshold was generated from the non-emergency supplied LKB* switchboard, the threshold being reached by a voltage drop on train B),

- RCP* 01 JA open due to voltage drop on LKB*,

- RCP* 01 MO stopped (no 6 6 kV at switchboard LGAC/ D)

4.4 - NEW MODIFICATIONS

The insertion of additional relays could reduce reliability Moreover, on a definitive voltage loss on switchboard LGA, maintaining the excitation contacts closed cannot damage the set In the other configurations, the risks of alternator operation in motor mode are eliminated by opening RCP* 01 JA through contacts 02 and 08 XR It was therefore considered preferable to eliminate relay 16 and 17 XR and the relay contact 20 XR on the start-up channel since the protection of the alternator in terms of the motorisation risks is provided by RCP* 01 JA Maintaining RCP* 03 JA in a closed position when the pump is at the end of slow down or in reverse rotation does not present any particular problems On a loss of voltage, the inertia slow down of the pumps is thus maintained as long as the auxiliary motor is not started

ABNORMAL HEATING OF FUEL SUB-ASSEMBLY COEC 3200

No.18 Date Tuesday 7 January 1986

1 - NATURE OF THE INCIDENT

Since the start of power build up, heating of fuel assembly COEC 3200 situated in the core at position 41/25 was detected clearly above that of the other assemblies and monitored without ever reaching the TRTC alarm threshold

2 - ANALYSIS

A first examination of the sheets of the inspection carried out at the plant before entering the new sub-assembly in the handling drum indicated that the loss of air pressure of this sub-assembly is in the high range of the criterion loss of pressure equal to 16 mbar for an acceptability of 12 ± 5 mbar

An enquiry carried out in parallel with the manufacturer COGEMA showed on 7 January in the evening that a natural rubber protective plug could have been forgotten in the leg of the assembly

A set of consistent correlations confirmed this hypothesis and showed that this assembly, forming part of the batch of four prototypes whose legs had been re-machined, was the only one involved

3 - STEPS TAKEN

The reactor was stopped as soon as the results of the COGEMA enquiry were known and the faulty assembly was discharged to the storage drum

An enquiry into the organisation of the manufacturer's quality was carried out in parallel by the main constructor NOVATOME-NIRA and EDF's manufacturing inspection service (SCF)

Chemical analysis of the plug consisting of 98% natural rubber highlighted the presence of a low content of impurities which, when diluted in the 3500 tonnes of reactor coolant sodium, led to a proportion lower than the limit permitted for nuclear quality sodium

CEA carried out a series of static sodium tests to study the behaviour of the plug at high temperature

Sub-assembly COEC 3200 was examined before being stored in cask IL 49 until dispatched to the LSAI laboratory (irradiated assembly monitoring laboratory at Marcoule) in June 1989

The criteria for checking new fuel sub-assemblies before loading by verifying the loss of air pressure were fine-tuned An endoscopic examination inside the leg has been added

4 - CONCLUSIONS

The investigation by SCF demonstrated that the presence of a plug is only possible on assemblies re-machined to modify the self-orientating legs CEA concludes from the tests that, at 400°C with sodium, pyrolysis of the plug is complete and results in the formation of an "amorphous" coke and gas products Endoscopic examination carried out on 16 and 17 June 1986 confirmed the hypothesis of a forgotten plug The analysis carried out on the residue samples confirmed this oversight

SODIUM LEAK IN THE SPACE BETWEEN STORAGE DRUM VESSELS

I.S. No.40

Date 3 April 1987

1 - NATURE OF THE INCIDENT

- 8 March 1987 leak detection alarm in the space between the storage drum vessels this alarm is not confirmed locally

- 9 March 1987 reactor pit bottom intermittent leak alarm investigations ongoing

- 31 March 1987 as a result of the investigations, cold nitrogen sampling in the inter-vessel space did not reveal any traces of sodium Nevertheless, the balances for the sodium levels of the drum and the storage tank show that there are 20 m^3 of sodium between the vessels

2 - IMMEDIATE ANALYSIS

The leak was confirmed and the first investigations to locate the leak were started (leak at level 14 6 m ± 1 m)

3 - IMMEDIATE STEPS

Evacuation of the equipment (new fuel, rods, COEC 3200 irradiated dummy assemblies) was begun as soon as possible to enable the drum to be emptied

An additional barrier was installed relative to the sodium which could leak from the retention tank and the mechanical strength of the drum safety tank in the new conditions created by the leak were confirmed

A strategy was developed to define the reactor operating conditions during the period of drum inoperability

Partial temporary operation of the APEC (fuel evacuation workshop) for storage of the " steel " assemblies withdrawn from the storage drum and placed in containers

Storage drum emptying in conditions allowing pre-location of the leak

This took place between 27 August 1987 and 9 September 1987 and enabled the leak to be located (infrared thermograph, xenon and helium detection in the inter-vessel space)

4 - CAUSES OF THE LEAK

The leak was caused by a horizontal crack around 60 cm long on the lower angle welding bead which secures a plate This rectangular metal plate welded on the inner face of the main storage drum vessel contributes to maintaining the drum sodium cooling circuit

After discharging the assemblies contained in the drum and emptying, in September 1987, the sodium contained in the drum main vessel, samples by cutting the metal from the plate which was at the origin of the leak were taken

The results obtained at the end of 1989, and the laboratory tests and examinations identified the most probable scenario involving the nature of the drum steel (ferritic 15 D3) and the simultaneous presence of three factors the existence of start sites (micro-cracking) in zones of high hardness, residual stresses close to the elastic limit of the material, and lastly, the contributions of hydrogen which allowed the brittling phenomenon to occur

However the cracking phenomenon could not be re-created in laboratory using specimen of the same steel grade in similar conditions with the presence of these three factors

5 - CONSEQUENCES OF THE INCIDENT

The progress of the incident showed the merits of having a retention vessel around the drum main vessel to collect and confine any sodium leak It also highlighted that this step could be improved by additional steps concerning leak control, monitoring of the integrity of the retention tank and the actions needed in the event of a leak from the second tank

Processing of the drum incident thus led to similar principles being applied to reactor vessels and reinforced the procedures to be applied in the case of leaks from the main vessel by supplementary measures These measures are contained in procedure U4 "Steps taken to limit the consequences of a hypothetical leak from the safety tank as a result of a leak from the reactor main vessel"

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 The reuse of the initial sodium drum after repair therefore proved to be impossible and it was necessary to define a 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

Lastly, reflection after the incident underlined the need to proceed to re-examine the design and manufacturing file for safety-related components in contact with sodium to confirm the absence of zones which could present the risks of leaks

6 - ADDITIONAL SAFETY ANALYSES CARRIED OUT AS A RESULT OF THE DRUM INCIDENT

The safety analysis carried out as a result of the drum incident covered the causes of the incident its sequence and its consequences Based on this analysis, additional technical measures were considered necessary before restarting the reactor These main measures are as follows

- setting up the procedures for interventions needed in the event of a reactor main vessel leak
- re-examination of the radiographic images taken during the manufacture of safety-related components (reactor vessels etc),
- in situ confirmation using the MIR (reactor inspection machine) robot of the good condition of the reactor vessel and execution of the first periodic inspections of this vessel

6.1 - PROCEDURE IN THE EVENT OF MAIN REACTOR VESSEL LEAK FOLLOWING A SAFETY VESSEL LEAK (PROCEDURE U4)

The reactor leak is an event considered extremely improbable and the reactor main vessel is surrounded by a second vessel, so-called safety vessel, to collect any leaks

However, although these vessels, which are constructed in stainless steel, are different from those of the initial fuel storage drum executed in carbon steel grade 15 D3, it was considered necessary to draw all the conclusions from the drum incident and the measures that had to be taken on the spot Indeed, when a leak on the first vessel occurs, it is essential to take a certain number of additional steps to measure and control the leak, preserve the integrity of the second vessel to the maximum and arrange around these two vessels the means for monitoring the tightness of the second vessel and the steps to limit the consequences of a leak from it

So-called procedure U4 was established with these aims in mind This enables very different accident scenarios associated with a main vessel leak to be dealt with and in this respect breaks down into actions spread over a period of time

Thus, it is foreseen that in the event of a reactor main vessel leak

- the reactor is immediately shut down and cooling begins,
- steps are taken to overcome the risks of a secondary vessel leak by pumping the sodium which flows into the inter-vessel space back to the main vessel this pumping circuit is called the leak recovery circuit and

it can be installed in around 10 or so days, this time being due to the primary sodium reactivity, the parts of this circuit are stored,

- sodium containment and cooling of the core in the event that the safety vessel itself were to leak is guaranteed To this end, the reactor pit is made leak-tight at its penetrations to allow argon blanketing by an injection sleeve installed permanently, the bottom of the reactor pit is lined with a layer of alumina (Cristalba) with suitable granulometry which, on the one hand, absorbs any sodium leak and acts as a blanket regarding inflammation risks, and on the other completes the leaktightness of the pit bottom The level of sodium in the main reactor is maintained above the heating part of the assemblies to ensure their cooling, if necessary by bringing in outside sodium from the SNA tanks or secondary loop (BCS) The residual heat is evacuated inside the vessel by the BPR cooling systems via the intermediate exchanger or by the RUR, and to the outside by the RUS circuits, the water of which is replaced by an organic liquid which does not react with the sodium in the event of a leak The RRI water circuit for cooling the reactor pit concrete is drained

These actions are spread over a time and chronologically can be separated into

- reactor shutdown which takes place immediately,

- first phase reactor cooling to 180°C which takes place with the BPR sodium-air exchangers available on the secondary loops in less than one week,

- the installation of the leak recovery circuit which requires a period of ten days associated with the sodium 24 decay,

- measures for completing leaktightness of the reactor pit in the lower part the neutron detectors are withdrawn and the accesses are plugged The bottom of the reactor pit, after opening the access door, is lined with alumina (Cristalba) This access first requires cooling of the sodium and decay of sodium 24, and can only take place after 10 or so days Installing the Cristalba takes several months The reactor pit access door is then welded up and the reactor pit is inerted (argon) Injection of the argon which normally takes place in the lower part of the reactor pit can also be injected in the upper part

- during the entire U4 procedure, steps for cleaning the heating part of the core are taken so as to evacuate the residual heat addition of sodium from tank SNA or a second loop are possible

Ultimately, core assemblies are discharged The discharged sub-assemblies are washed and installed in the APEC pool The reactor sodium can then be drained

6.2 - RE-EXAMINATION OF THE FABRICATION FILES

The design and fabrication files were re-examined in the light of the incident encountered on the storage drum This long task covered in priority the main reactor vessel and its safety vessel

The new analysis of the radiographic images of the two vessels taken during manufacture highlighted

- signs already noted during the first examination and wrongly interpreted as satisfying acceptance criteria,

- new indications which had not been noted during the initial inspection

The eventual noxiousness of these indications were then studied This consisted in adopting penalising hypotheses for the dimensions of indications and their position As a first step in the calculation, the propagation of the supposed fault was estimated by taking into account mechanical loads which may arise on it during plant operation This calculation estimates in particular the consequences of a certain number of shutdowns or events leading to mechanical stresses on the structures studied Thus, we can determine the maximum and pessimistic extension in length and depth of the defect A second calculation is then made to assess the effect on this defect of an increasing mechanical load which, depending on the case, might be an earthquake or other accident situation All these calculations take into account the penalising hypotheses and have shown that the anomalies encountered are not noxious

6.3 - INSPECTION OF THE MAIN REACTOR VESSEL WELDS

The inspection, which was made on 27 June 1988 to 23 August 1988, was carried out using the MIR robot (inspection module for fast reactors)

This module consists of a carriage which bears on each one of the two vessels through its four wheels

A television camera reads the marks etched on the safety vessel and guides the robot

The welds are inspected using focalised ultrasonic transducers, which employ the echographic technique, mounted on the robot A coupling fluid is installed between the transmitter and the weld to be checked using a small vessel containing this fluid This vessel is in contact with the wall of the zone to be inspected through a seal A visual inspection of the weld is also carried out by a CCTV camera

The inspection programme took into account the indications noted during the examination of the radiographic images at the end of the main vessel fabrication

The inspection carried out did not show any unacceptable faults on the welds corresponding to the size of the faults taken into account in the " noxiousness " studies as a result of rechecking the radiographic images, nor the " noxious " evolution of the triple point weld condition The triple point designates the horizontal circular weld corresponding to the position where the weight of the core and the reactor internals are mechanically supported

As with the water reactors, new inspections will be carried out at regular intervals using the MIR robot, the inspection system of which proved to be accurate and effective during this first inspection operation

FALL OF THE POLAR CRANE DERRICK

No.56 **Date** : 2 October 1989 at 17.00

1 - NATURE OF THE INCIDENT

The incident concerns an item of plant temporarily fitted on the reactor building polar crane carriage with a view to dismantling the access gangway to the crane arch. The gangway is a structure fixed on one of the polar crane beams which it was decided to replace by a simple hooped access ladder in 1988.

The incident occurred during the statutory testing of this derrick after erection. The reactor was shut down and the dome was entirely closed. The test load on platform R 805 (no-risk zone) had just been raised 15 cm The upper orientation bearing on the derrick boom broke causing the latter and its winches to fall.

The consequences on the installations were very limited: striking the dome and the chilled water system DEG without loss of leaktightness, deformation of the cable tray without break or electricity fault.

2 - ANALYSIS

The incident is due to the failure of 8 fixing screws on the upper bearing of the boom mast at the gantry upright. The range of boom orientation adopted on design was 90° between the longitudinal and transversal axes of the crane In the sequences of gantry dismantling operations, the maximum bearing on the derrick could reach 10 m in the crane longitudinal axis (removal of items in R 805) but should not exceed 4 m in the transversal axis (elements on the crane) At the planning stage, the orientation bearing had therefore been placed in the most penalising position, i e. on the longitudinal axis. During execution, the bearing rotational axis was installed in the transversal axis. On testing the load in the most penalising conditions with the boom in the longitudinal position, the bearing fixing screws were subject to shear stresses which led to their successive failures It is to be noted that the statutory test in the fabrication workshop was carried out in the direction of greatest strength due to the lack of space

3 - STEPS TAKEN

Inspections showed.

- dome : absence of fault according to GDL (EDF group of laboratories) report,
- small west cupola · impacted zone undamaged (inspected by dye penetrant and radiographic examination) and longitudinal weld situated close to the impact undamaged (radiographic examination),
- small cupola gangway : structural steel work non-deformed,
- dome gangway . welding of the support on the dome undamaged (dye penetrant examination),
- chilled water piping · replacement of a 7 m section and inspection,
- cable trays repair, replacement of cable

The eventual " noxiousness " of the faults was analysed

- west cupola demonstration of the buckling performances under dimensioning loading (-0 2 bar) of the deformed part.

Analysis of the incident having demonstrated that the fundamental cause was a failure of the quality assurance measures set up, an inquiry was initiated (see mail SX90-0174 of 19 January 1990)
Henceforth works implying load transport in the reactor hall, risk studies were required and protective measures taken if needed to avert damages.

INCREASE OF THE PRIMARY SODIUM IMPURITIES LEADING TO AN OVERSHOOT ON THE RANGE AUTHORISED BY THE OPERATING SPECIFICATIONS

I.S. N°61 Date 20 June 1990

1 - GENERAL INCIDENT CHRONOLOGY

After unit shutdown to permute the diluents (Sept 7 1989 to April 13 1990) and shutdown for works as a result of detection of a leak on the feedwater purification circuit (April 28 1990 to May 31 1990), the reactor was moved into critical mode and reached its nominal power on June 11 1990

During all this period up to the temperature build-up (June 10 20% nominal power), sodium cleanliness was monitored by the operating teams on recordings delivered by the plugging indicators using the usual operation methods plugging temperature Tb[1], general slope of the curve Monitoring confirmed that there was a good level of cleanliness, in particular, the 110°C [2] level lasted around six hours

From the temperature build-up, the operator observed a plugging temperature rise and therefore an increase in the impurities

After analysing the recordings, confirmed by the constructor and the sodium chemistry experts, the plant concluded that the reactor was still operating in the range limits authorised by the operating specifications (120°C<Tb<150°C) and that the return to normal conditions could be envisaged in the delay of one month authorised by the specifications Indeed, the increase in the plugging temperature after shutdown for works with the reactor cover open is a phenomenon which has already been observed It corresponds to the phase of re-dissolving with the temperature of the impurities formed during the shutdown (the works having allowed air to ingress with associated oxygen and humidity)

It was then observed that the integrated purification filtration cartridges were saturated

In addition, plugging temperature measurements did not follow the temperature decrease expected by the experts, the operator therefore decided to stop the reactor on July 3

After reactor fast shutdown, it was maintained isothermal at a temperature of 250 °C, i e at least 40 °C above the oxygen saturation temperature in the primary sodium when the rate of impurities was maximum (15 ppm oxygen)

On July 9 1990, an analysis of the cover gas revealed a nitrogen content of 17% and brought into question the previous interpretation As a result, the plugging measurements led to the conclusion that the reactor had operated outside the operating range before being shutdown

The application of a programme of investigations to identify the air ingress into the argon of the cover gas resulted in the origin of the pollution being discovered on July 23 This was a circulation pump on an activity measurement channel whose diaphragms were defective 540 l/h of air entered the reactor argon circuit downstream of this circulation pump

2 - INCIDENT DIAGNOSIS, POLLUTION LEVEL, PURIFICATION

On completion of the analysis of the plugging indicator recordings over a long period and the analysis of the samples made at different points of the primary argon circuit, the conclusion was reached that around 120 kg of oxygen entered the reactor system

Moreover, it results from this last analysis that the reactor operated outside the operation specification range for three days before shutdown

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(1) The plugging temperature is obtained on the cooling gradient of the plugging indicator pellet This gradient is initiated when the flow in the pellet is stable at its higher value The plugging temperature matches the beginning of flow decrease

(2) The low level is initiated when the pellet temperature reached 110°C in a cooling cycle and ends when the flow in the pellet reaches 55% of the initial flow The low level time is therefore an indicator of the clogging speed on the pellet in a measurement cycle and therefore the sodium purity

2.1 - ELIMINATION OF THE REACTOR COVER NITROGEN

The nitrogen content of the primary argon circuit (17% on maximum pollution) fell as soon as the argon circuit blowers were stopped on July 13 to reach 6% when the air ingress was eliminated on July 23

This rate then stabilised at 4% during August The argon was maintained static

During September, a series of inflation - deflation operations on the reactor cover brought the rate to 0 4%

Thereafter, the evacuation of the residual nitrogen brought it down to <0 01% at the end of August 1991 This value is below the maximum value permitted by the operating specifications updated after the incident (0 3%)

2.2 - SODIUM PURIFICATION

To reach the level of cleanliness required in all the reactor normal operating conditions (primary sodium oxide rate <3 ppm operation with an oxygen rate between 3 and 5 ppm during one month at the end of a handling period), a purification programme was implemented

This programme was run with a view to constantly controlling the changing parameters and maintaining these in a more favourable range than on the shutdown of July 3 1990 Therefore, sudden re-dissolving or carrying along of the sodium oxides whose presence was presumed to be at the surface of the sodium had to be avoided

It also enabled us to acquire more thorough knowledge of pollution phenomena and behaviour of plugging indicators

The programme therefore proceeded in several campaigns, each one of them being run based on the results obtained during the previous campaign

1stcampaign July - October 1990 Purification at 250°C

This campaign began by changing the cartridges on the two clogged purification units during the shutdown of July 3 and ended by installing new cartridges It enabled 40 kg of oxygen to be trapped in this set of cartridges

2nd campaign October 1990 - April 1991

This consisted of the following

- initial condition reactor at 250°C, reactor pumps at 75 rpm
- reactor sodium temperature increase to 350°C
- reactor coolant pump speed increase to 250 rpm,
- temperature of the reactor increased to 400°C Disturbances on the plugging measurement led to progressively lowering this temperature until 300°C As soon as the 375°C stable level is reached these disturbances are substantially mitigated
- 300°C level,
- once again, the reactor sodium temperature increased to 400°C The values measured by the plugging indicator authorised a fall to 180°C without any risk of clogging,
- final condition reactor block at 180°C, reactor coolant pumps at 75 rpm

On completion of this stage, the oxygen rate in the reactor sodium was brought to a value close to the 1 ppm considered satisfactory

3rd campaign June-September 1991

This stage was preceded by washing of the separation column situated on the reactor argon circuit in order to dissolve any sodium oxides in deposits trapped at the reactor outlet of the argon circuit

The campaign was undertaken in the following stages

- Initial conditions, reactor block at 180°C, primary pumps at 75 rpm,
- Reactor sodium temperature increased to 395°C Interference on the plugging indicators appeared similar to those encountered during the second stage,
- Reactor pump speed increased to 433 rpm,
- Reactor temperature increased to 420°C No evolution of the plugging parameters confirmed that there is no oxygen pollution input due to this isothermal operation at high temperature in nominal reactor pump rotation conditions,
- Return to 395 °C, reactor coolant pumps operating at 110 rpm Sampling of reactor sodium for analysis The purification units indicated signs of clogging at the sodium inlet/outlet moderating heat exchanger This clogging, which translates into a fall in the thermal exchange coefficient, is attributed to the presence of impurities other than sodium oxide These impurities, which are in solution in the reactor sodium during isothermal operation at a temperature above 375°C are deposited on the cold walls of the exchanger tubes,
- Return to 180°C, reactor pumps at 110 rpm

During all the reactor sodium purification phase thermal hydraulic monitoring of the reactor took place in order to ensure that there is no clogging phenomenon by carrying along any surface creams or impurities in narrower sections of the sub-assemblies

2.3 - ALTERNATIVE PURIFICATION STRATEGY BY THE EXCEPTIONAL PURIFICATION CIRCUIT

As a protective measure, the auxiliary reactor sodium purification circuit of the storage drum was re-filled in order to maintain the exceptional reactor sodium purification available through the two circuit cold traps

Since integrated purification had proved to be sufficient to eliminate the reactor sodium pollution, exceptional purification was not undertaken

3 - EVALUATION OF THE DIRECT OR POTENTIAL CONSEQUENCES AND ASSOCIATED ACTIONS FOR REINSTATING THE INSTALLATION

In parallel with purification, reinstatement works, non-noxiousness studies and investigations were performed

3.1 - HYDRAULIC RELIEF VALVE PROTECTING AGAINST PRESSURISATION - DE-PRESSURISATION IN THE PRIMARY ARGON CIRCUIT (RAA0 01 ZH)

This saw its calibration liquid (NaK sodium-potassium mixture) oxidised by the air which entered the circuit and this prohibited maintaining the argon in circulation This relief valve has been replaced

3.2 - INSPECTION PROGRAMME - RE-QUALIFICATION OF THE REACTOR COVER AND THE PRIMARY ARGON CIRCUIT ON WHICH THE OXIDE SPRAY COULD HAVE DEPOSITED

These inspections concluded in the absence of deposits prejudicial to the long term operation of the equipment which were in contact with the sodium aerosols during the pollution period

3.3 - STUDY OF THE CONSÉQUENCES ASSOCIATED WITH CORROSION AND NITRIDING

A study demonstrated that the effects of the corrosion are minor and limited to the equivalent of 70 days' operation at nominal power in normal conditions of sodium cleanliness The effects associated with nitriding are non-significant

The conclusions of this study were validated by a formal notice issued by the CEA/EDF Materials Working Group

4 - FUNDAMENTAL CAUSES - INFORMATION GAINED

The fundamental causes of the incident were identified as

- incomplete analysis during primary argon circuit design of the risks of air ingress and more generally non-leaktightness, and their prevention ; this occurring notably at the activity measurements circulation pumps,
- absence of primary argon on-line monitoring devices in spite of such a system having been installed in the Phenix, PFR, KNK and SNR 300 plants,
- incomplete referencing of equipment and preventive maintenance programmes,
- incorrect interpretation of the plugging indicator recordings,
- inappropriate operating specifications both in terms of their clarity and their applicability

As a result, further actions with a view to drawing all the information from the incident resulting in measures implemented before plant start-up were undertaken before the unit was restarted

4.1 - INSTALLATION OF MONITORING EQUIPMENT (CHROMATOGRAPH) ON THE PRIMARY ARGON CIRCUIT

4.2 - DESIGN IMPROVEMENT OF THE SYSTEM FOR SAMPLING THE GAS ON THE ACTIVITY MEASUREMENT CHANNELS

A system for detecting diaphragm failure has been installed on the circulation pumps

4.3 - REVIEW OF THE PHENOMENA ASSOCIATED WITH POLLUTION

A series of studies and research was carried out to determine the nature, mechanical and thermodynamic behaviour in reactor sodium of the impurities, their noxiousness in terms of the risks of clogging and corrosion of steels, their influence on the plugging curves and the effectiveness of the trapping function with regard to each of the impurities

4.4 - REDRAFTING OF THE REACTOR CLEANLINESS MONITORING PROCEDURES

New criteria for monitoring the purity of the primary sodium have been defined since the parameters used until now to detect any changes (loop temperature Tb) were considered insufficient.

Henceforth, the operator will base its operations on the unplugging temperature measurement [1] associated with the sodium oxide (low unplugging temperature) and on the duration of the low level which is an effective and simple method of qualitively monitoring the sodium purity and its evolution over time

The unplugging temperature is close to the saturation temperature for the impurity present in the sodium Nevertheless, it confers a boundary which is higher than the saturation temperature based on thermodynamic data and the cinetics of the dissolution in the pellet.

4.5 - PROJECT REVIEW OF THE PRIMARY ARGON SYSTEM

The principle of this project review was to identify the causes of the anomalies and the dysfunctions which could affect the " intermediate containment barrier " and " the primary sodium inert gas cover " functions of the primary argon systems. They led to additional studies, corrective actions, counter-measures and modifications being defined and progressively implemented.

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[1] The unclogging temperature is obtained on the heating gradient of the clogging indicator pellet This gradient is reached when the flow in the pellet reaches 55% of the initial flow and terminates when the high temperature level has been reached

5 - OVERALL ACTIONS

In order to make an analysis which goes beyond the specific measures directly associated with the incident deeper reflection has been undertaken in the following aspects

5.1 - *REVISION OF THE GENERAL OPERATING RULES*

Chapters 3 and 9 of the GOR have been read in detail in order to check that each specification is clear and precise and that it results in an operating instruction which is consistent with it and that the physical means to check compliance are appropriate

5.2 - RE-EXAMINATION OF THE PREVENTIVE MAINTENANCE PROGRAMMES

The plant has undertaken work on two main lines to ensure exhaustiveness of the preventive maintenance operations undertaken on safety-related equipment

1st line : full listing through a campaign to identify all the equipment

2nd line by a team of equipment expert engineers, preparation of a zero point list of the preventive maintenance procedure and then a drafting schedule

5.3 - MAINTAINING COMPETENCES AND EXPERT CAPACITIES AVAILABLE TO THE PLANT

Actions have been undertaken in order to ensure permanent management of the competences which have to surround the plant based on a 3-stage approach

a) list of competences available in all the organisations involved (Novatome-Nira, CEA, EDF) ,

b) preparation of " management contracts " formalising the undertakings of each organisation ,

c) regular preparation of a statement of available competences

5.4 - EXPERIENCE FEEDBACK, ANALYSIS OF THE PAST, 2ND LEVEL ANALYSIS

An organisation has been set out with means for guaranteeing satisfactory processing of experience feedback from the plant, other French and Overseas fast neutron reactors and the PWR plants

Group for analysing potential operating problems

Its aim is, through the reading of tests, transfers, modifications and incident documents, to check that all the information has been drawn from the plant start-up period

Group R

The role of Group R is to organise experience feedback from French and foreign fast neutron reactors together with PWR plants

6 - CONCLUSION

The primary sodium pollution incident, which is particularly significant of the prototype nature of the fast neutron reactor at Creys-Malville, led to a long shutdown which justified its classification as a level 2 seriousness incident, although it did not bring into question the safety of the installations

Shutdown for sodium purification was used to make a thorough analysis of the direct and indirect causes of the events and to take corresponding measures re-examination of the plant maintenance and operating procedures, potential review of expertise available and needed around the operator

COLLAPSE OF THE TRAIN A TURBINE HALL ROOF UNDER THE WEIGHT OF SNOW

I.S.No.63 Date: 13 December 1990 at 11 00

1- NATURE OF THE INCIDENT

The weight of accumulated snow caused train A turbine hall to collapse, leading to a loss of voltage on this train

Since the 225 kV line was off-line at the time of the incident, train A switchboards were without voltage

The loss of voltage led to start-up of the train A standby diesel generators (LHPE and LHPF)

Diesel LHPE did not automatically couple and required local intervention

2- ANALYSIS

The incident was caused by considerable snowfall which led to the collapse of the train A turbine hall roof Part of the cladding fell on the high-voltage equipment of train A causing a zero-phase sequence on the main transformer and therefore opening of the line circuit breaker

The 225 kV was off-line and the loss of voltage led to the two diesels LHPE and LHPF starting without coupling diesel LHPE due to incorrect operation of the voltage regulator (fuse break contacts) and despite changeover to the standby regulator

Manual coupling of diesel LHPE led to re-powering of the reactor coolant pump E although this was in reverse rotation as a result of failure of a rotation sensor on this pump

The standby unloading cask (MHU) that had been stored in the turbine hall was damaged

The reactor building and steam generator access hatch was blocked in a closed position resulting in inoperability of local mode devices

3- STEPS TAKEN

- Reconnection of the auxiliary 225 kV line and repair of LHPE diesel generator
- Installation of a 6 6 kV jumper cable between the non-emergency supplied 6 6 kV switchboards of train A and train B
- Addition of special steps for clearing snow in the " cold weather " instruction
- Repair of the turbine hall
- Dimensional analysis of the roofs of the other buildings
- Repair of the MHU unloading cask
- Checks of the condition of reactor pump E after re-powering when in reverse rotation
- Decision to apply a loss of electrical source instruction throughout without seeking to use a recovered meantime source

SIMULTANEOUS OPENING OF SEVERAL CONTAINMENT BARRIERS

I.S. No.88 Date : 3 May 1995

NATURE OF THE INCIDENT

On a servicing operation on reactor coolant system argon blowers (RAA0), early dismantling of a terminal box on blower 02 CO led to opening of the intermediate containment barrier (2nd barrier) The third barrier is by-passed by this circuit

In parallel, the truck area (4th barrier) was open, leading to the loss of the feedwater containment

This situation did not meet the operating specifications in shutdown status which require availability of at least one of the containment barriers

ANALYSIS

The origin of the event is found in the failures which occurred in organisation of multi-competence maintenance and due to human elements

Analysis of the event highlighted faults in preparing, scheduling, preparing withdrawal from operation and execution of the work

The specificities of the equipment (complexity of the 2nd barrier and difficulty to identify certain of the materials which constitute this barrier) constituted an aggravating factor

STEPS TAKEN

- Henceforth, technicians will indicate " Safety, caution containment barrier " in operation documents and requests for maintenance regimes when carrying out work affecting the containment barrier

- The requirements of the quality organisation and safety culture have been recalled through the presentation of a nuclear safety memo adapted to the LMFRs, notably communicated to electricity and mechanical engineering staff A working group took into account the information gained from this incident when drafting the risks analysis guide adapted to Creys-Malville

- Lastly, when repairs require several specialist contractors, co-ordination will be henceforth the responsibility of a single person

FUSION OF THE LARGE ROTATING PLUG LIQUEFIABLE METAL SEAL

REACTOR CRITICAL

I.S. No.92 Date : 12 May 1996

NATURE OF THE INCIDENT

During neutronic tests at nominal power < 3 %, the metal to metal supports of the rotating plugs were heated 48 hours before reactor shutdown. This action, which is compatible with the current tests, also led to unintentional heating of the liquefiable metal seal (JML), the liquid status of which is incompatible with reactor criticality

ANALYSIS

This incident caused an error in an operation sheet of procedure G 8 (Operation to Handling changeover) on a newer version published in 1991. It was not detected by the quality assurance system used to draft the instructions

The line of defence concerning inadvertent re-supply to the JML power cabinets (administrative lockout G21) did not avoid the event as a result of the authorisation to remove padlocking issued by the shift engineer

STEPS TAKEN

These were of two types

- Operating instructions concerning the liquefiable metal seal

Clarification of the temperature monitoring methods for the liquefiable metal seal indicated in instructions G1 and G1-bis (conditions at start and end of monitoring, temperature at which the liquid metal seal are considered melted).

- Criticality instructions

An inspection of the liquefiable metal seal condition added to the checks before criticality

ARGON LEAK FROM THE BELL OF EXCHANGER RCPE-02-EX

NATURE OF THE EVENT

From the outset, the argon supply of intermediate exchangers has justified careful monitoring by the operator A few re-inflations per month were observed on a few intermediate exchangers including notably IHX RCPE 02 EX

On start-up in 1994, following the long BCS shutdown, a significant increase in the frequency of bell re-inflation on IHX-RCPE 02 EX was observed (up to around one re-inflation per day) As a result of the fast shutdown of 17/12/94, the leak rate from the bell increased significantly to reach a value close to the threshold set by the operator (160 mbar/day) This justified shutdown of the reactor

ANALYSIS

The investigations were carried out during reactor operation and then on hot shutdown and cold shutdown The leak was located and characterised In view of this characterisation, the absence of a generic aspect was checked for the other intermediate exchangers

STEPS TAKEN

Safety analysis

The analysis of the safety of the event led to the following conclusions

- leak evolution could be controlled with great reliability using operator instrumentation (pressure monitoring) and by monitoring the re-inflation frequency

- the new inflation procedure was the subject of a safety analysis which highlighted a sufficient number of lines of defence in terms of the identified risks

- in the event of an increase in the bell leak rate, a reactor shutdown criterion is applied This criterion, which fixes the maximum frequency of bell re-inflation at eight per day, enabled the acceptable primary sodium gassing (8 8 10^{-4} Nm^3 gas /m^3 primary sodium on average) to be complied with, including large margins This latter rate integrates the results of the tests associated with the risk of gas accumulation in the core support diagrid and transferthrough the core

- aggravation of the bell argon leak did not have unacceptable consequences for reactor safety The studies undertaken as part of the analysis of the transfer of gas into the reactor remain within the umbrella case limits In particular, the insertion of reactivity induced by sudden de-pressurising of the bell as a result of the pipe break at the level of the bell and the passage of gas into the reactor remain acceptable (less than 1$)

- as one of the lines of defence, in the event of fast build up of an intermediate exchanger leak rate, the fast shutdown alarm on bell deflation (the reliability of which has been improved) enables the reactor to be quickly brought to a sub-critical condition

Repairs

In accordance to a repair safety file, the leak was plugged by installing a metal sleeve applied and held in position by permanent deformation (expansion by passing hydraulic pressure) using two crimpings 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 were then left in the piping, and is subject to specific periodic monitoring This technique enabled the original use of this piping to be restored This return to conformity (leak plugged) has proved to be of an acceptable quality both at power (only one inflation weekly has become necessary) and on shutdown

Annex II

EVENTS WITH BEARING ON EFR

28.09.85	7	reactor trip	hydraulic coupler and primary sodium flowrate measurement	design of the primary flowrate measurement; procedure (rod insertion before changeover to pony motor)
08.10.85	8	emergency shutdown by clad break detection	core measurement conveyance	instrumentation and control (very weak currents)
22.10.85	11	emergency shutdown by clad break detection	core measurement conveyance	instrumentation and control (very weak currents)
22.11.85	12	loss of power induced reactor trip followed by coast-down of primary pumps without inertia	primary pumps inertia device	relaying design
07.01.86	18	abnormal heating of subassembly	plug left in sub-assembly leg	quality control
06.04.86	24	reactor trip due to fertile sub-assembly overheating	temperature fluctuation at sub-assembly outlet	thermocouples positioned closer to sub-assemblies; hot header thermalhydraulics
08.05.86	25	bursting of handling cask observation window	thermal effect on seal tightening	design
02.04.87	39	DVS rods fall-down due to trip signal by flowrate measurement when changing over one primary pump to pony motor	primary sodium flowrate measurement	design of the primary flowrate measurement
08.03.87	40	leakage of the fuel storage drum main	steel grade	steel grade; safety vessel anchored in reactor pit;

		vessel		refractory concrete
21.05.87	43	rupture of one DHR fan blades	fan (fatigue rupture)	quality control
11.01 98 12.01.89	53	anomaly in the fast isolation-decompression sequence of a steam generator	instrumentation and control	instrumentation and control
28.04.90	60	sodium leakage on the F secondary loop auxiliary system	tee-connection (thermal fatigue)	analysis of hot/cool sodium mixing conditions
20.06.90	61	rise in primary sodium impurities level leading to exceed the limits authorised by operational technical specifications	rupture of membrane in gas radiometer; cover gas chemical monitoring	addition of catharometer
12;05;9 1	65	sodium leakage on C-train diverse DHR plugging indicator	plugging indicator (pre-heating)	pre-heating design
15.09.93	79	rupture of E-primary pump coupling	gear coupling	gear design (allowing for shaft tilting), cog material (nitride steel)
19.04.94	82	discovery of through wall crack on rupture disks balancing line of C-train DHR loop	balancing line connections	elimination of connections on dead leg
03.05.95	88	simultaneous opening of the cover gas circuit and the reactor trucking gate	argon blower	maintenance-designed equipment and plant

XA0056254

OPERATING EXPERIENCE WITH BELOYARSK FAST REACTOR BN600 NPP

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Abstract

The main results of the seventeen-year operation of the BN600 Nuclear Power Plant are considered. The principal backfittings of the main BN600 Power Plant equipment are presented and summarised.

Introduction

In 1997 BN600 has accumulated 17 years of power generation. Connection to grid took place on April 8, 1980. The design electrical power as high as 600 MW was achieved in December 1981.

The experience from BN600 power generation shows BN600 to be reliable in control, safe and highly valuable for further commercial deployment of large fast reactor power plants at the current stage of nuclear power development.

During operation the following objectives set out at the design phase were achieved:

- demonstration of sustained, reliable and safe operation of a sodium cooled fast reactor power plant,

- long lifetime testing of the large components operating in sodium environment in high neutron and gamma fields,

- mastering of the sodium technology, optimization and improvement of the operating plant procedures, mastering of the sodium equipment replacement and maintenance technology,

Summary of Power Plant Operation

At present time the BN600 power plant is in steady operation under electrical capacity ranging from 580 to 610 MW.

During a number of recent years beginning from 1983 the load factor of the operating power plant has been in the range of 72 to 80% (the overall load factor is 73.5%) and outages are mainly made for reloads. During 77% of time the power plant generated electrical power as planned, 21% of time were spent for the planned maintenance and reload outages, the unplanned losses accounted for less than 2% of time.

Totally (as of 1/11/97) 36 reactor core reloads have been carried out at peak discharge burnup of 10% h.a. and experimental fuel burnup of 11.8% h.a.

Short-term objectives in respect of the core are further tests of various plutonium fuel sub-assemblies and the activities on the third modification of the 11% h.a. burnup core.

17 years of the BN600 power plant operation showed good agreement of the design and actual performance of the main components. The following technical and economical indicators were achieved:

Table 1

No	Indicator	Unit	Total value as of 1/11/97
1.	Electrical generation	mln kW-h	64459,1
2.	Auxiliary power	%	7,5
3.	Load factor	%	73,5
4.	Availability factor	%	75,3
5.	Gross efficiency	%	41,5
6.	Net efficiency	%	38,5
7.	Number of plant outages	unity	67
8.	Number of loop disconnections	unity	19\26\17
9.	Average output	MW	—

Steady, reliable and safe operation of the power plant and its high technical and economical performance could be provided only due to the large scope of the integrated research and development activities resulting in modification of some units and systems.

Besides it was required to develop and modify core components (fuel sub-assemblies, control rod guide tubes and control rods), to explore and modify electrical drives of sodium pumps, to modify a reactor refuelling system, to construct advanced failed fuel detection systems, to design and construct advanced reactor vessel integrity inspection systems, reactor vessel and auxiliary primary sodium pipeline displacement measurement systems, to remarkably improve water-sodium reaction detection systems of the water-sodium steam generators, to construct technical diagnostic systems of the reactor and steam generator components, to determine and extend equipment lifetime and to carry out many other activities; even this rather incomplete scope of the objectives shows a novelty and variety of the problems which have been successfully solved by the Beloyarsk NPP team in cooperation with OKBM, IPPE, GIDROPRESS design company.

During operation a high degree of the radiation safety of the BN600 power plant under the sustained operating conditions at the rated generation parameters was provided for. The results of the measurements of the radiation environment and process medium activity have shown their values to be within the design limits. The radiation levels in the attended and semi-attended areas of the reactor building have been within the authorized limits, the gamma radiation levels of the secondary circuit components have been within the natural background limits.

Atmospheric discharges have been well within the limits and within the range of 2 to 10 Ci/day (522 Ci/year in average) and they tend to decrease; there have actually been no liqiud radioactive effluents from the plant while it being operated. The average release of the solid radioactive waste is 22 m^3/year. The average collective dose of the plant personnel is 84 rem/year.

Results of 17-year Operation of Power Plant

Reactor Core

Since 1980 till 1986 the BN600 reactor operated with the core of the firstload type with the peak burnup of lowly enriched fuel being as high as 6.1% h.a. and that of highly enriched fuel 8.3% h.a.

During operation of the first type core the fuel failure occurences have been observed by the end of nearly each interval between reloads. As a result of examination of the irradiated fuel sub-assemblies it was found that the fuel failures had been generally caused by the strained operational fuel conditions (due to peak linear ratings as high as 54 kW/m and reshuffling of highly enriched fuel sub-assemblies from the outer towards the inner core locations) as well as by poor fuel cladding structural materials. It should be noted that it was the highly enriched fuel that mostly failed in the core.

During 1986 and 1987 the reactor was changed over to the modified core (01M) with the peak burnup as high as 8.3% h.a. to improve fuel performance and to increase BN600 reactor fuel burnup. The new peak burnup values were 6.5% h.a. for the lowly enriched fuel, 6.9% h.a. for the intermediately enriched fuel and 8.3% h.a. for the highly enriched fuel. The principal difference between the first load type core and the modified core was an increase in the height of a fuel fissile section from 750 to 1000 mm and utilization of three uranium 235 fuel enrichments, i.e. 17%, 21% and 26% instead of 21% and 33%.

Just after the change-over to the first modification core (the beginning of the 20th cycle) all the fuel failure causes were actually corrected, namely:

- the core: the peak linear rating was decreased down to 47.2 kW/m and the advanced structural materials more resistant to radiation were started to be used as well as the reshuffling of highly enriched fuel sub-assemblies from the outer towards the inner core locations was removed out of practice,

- the radial blanket: the fuel pin gas plenum height was increased from 160 to 310 mm and the band spacing of fuel pins was introduced.

Over a period of 1988 to 1990 the reactor operated with the first modification core.

Over a period of 1991 to 1993 the reactor was changed over to the second modification core with the new peak burnup values being 9.0% h.a. for the lowly enriched fuel, 9.5% h.a. for the intermediately enriched fuel and10% h.a. for the highly enriched fuel.

This was enabled by the utilization of the advanced fuel cladding and wrapper materials resistant to a damage dose as high as 100 dpa.

The fissile height was increased from 1000 to 1030 mm and the effective fuel density from 8.5 to 8.6 g/cm^3 to provide for the necessary reactivity margin in the second modification core by means of the 4% increase in the fuel load.
The outer radial blanket was extended by 16 fuel sub-assemblies and the in-reactor storage became respectively less by 16 fuel assemblies.

The first (01M) and the second (01M1) modification cores themselves have the same configuration, i.e. 136 lowly enriched fuel sub-assemblies of 17% enrichment, 94 intermediately enriched fuel sub-assemblies of 21% enrichment and 139 highly enriched fuel sub-assemblies of 26% enrichment.

The radiation-resistant fuel structural materials, i.e. the ЭП-450 steel as wrapper material and the ЧС-68 steel as cladding material, are used.

Table 2 shows the main characteristics of the cores and the operating fuel conditions.

319 experimental fuel sub-assemblies were tested in the BN600 reactor.

In support of the BN800 reactor core design validation 6 vibro-packed and 8 pelletized MOX fueled sub-assemblies have been tested in the reactor and 4 more sub-assemblies are being tested.

By present time the 11.3% h.a. peak fuel burnup BN600 reactor core design has been elaborated and is at the stage of approval and finalization. The activities on development of the 12% h.a. peak burnup core have been started.

Table 2 The main characteristics of the BN600 reactor cores

No	Characteristic identification	First type of the core (01)	First modernization (01M)	Second modernization (02M)
1.	Number of fuel sub-assemblies, pcs.:			
	- Low enriched fuel S/As	217	136	136
	- Intermediately enriched fuel S/As	—	94	94
	- Highly enriched fuel S/As	144	139	139
	-Depleted uranium dioxide control rods	8	—	—
2.	Weight of enriched uranium, kg (% of enrichment):			
	- Low enriched fuel S/As	20,0 (21)	27,6 (17)	28,9 (17)
	- Intermediately enriched fuel S/As	—	27,6 (21)	28,9 (21)
	- Highly enriched fuel S/As	20,0 (33)	27,6 (26)	28,9 (26)
3.	Core height, mm	750	1000	1030
4.	Peak local burnup, % h.a.:			
	- Low enriched fuel S/As	6,1	6,5	9,0
	- Intermediately enriched fuel S/As	—	6,9	9,5
	- Highly enriched fuel S/As	8,3	8,3	10,0

TABLE 2 (cont.)

5.	Peak damage dose, dpa: - Low enriched fuel S/As - Intermediately enriched fuel S/As - Highly enriched fuel S/As	 49,1 — 42,5	 53,3 51,0 54,0	 75,0 72,0 69,0
6.	Fuel residence time, efpd: - Low enriched fuel S/As - Intermediately enriched fuel S/As - Highly enriched fuel S/As	 200 — 300	 330 330 330\495	 480 480 480
7.	Height of axial blankets, mm: - upper axial blanket height -lower axial blanket height	 400 400	 300 380	 300 350
8.	Height of pin gas plenum, mm	800	660	660
9.	Effective fuel density, g/cm^3	8,2	8,5	8,6
10.	Hot spot fuel cladding temperature, ^{0}C	706	697	696
11.	Peak fuel sub-assembly power, MW: - Low enriched fuel S/As - Intermediately enriched fuel S/As - Highly enriched fuel S/As	 4,5 — 4,7	 4,4 4,5 4,6	 4,4 4,6 4,7
12.	Peak linear rating, MW/m: - Low enriched fuel S/As - Intermediately enriched fuel S/As	 53 —	 43 45	 43 44
	- Highly enriched fuel S/As	54	48	47

Fuel Handling Systems

Operating Experience from Reactor Refuelling System

As it is mentioned above 36 reactor reloads have been carried out since the power unit has been put into operation. In general the refuelling system performance during this period of time was sufficiently reliable - in fact no failures which would have caused the delay of making the reactor critical occurred. It is the refuelling system mechanism position indication and control system that has mostly been a trouble contributor. As far as the mechanical part is concerned the following operating results can be noted:

A. The fresh fuel argon-filled and the irradiated fuel sodium-filled storage drums

Since the start of operation neither failures nor faults have occurred. On the basis of the initial operation a number of the irradiated fuel storage sodium-filled drum cells have been bored to provide for the unloading of the irradiated fuel sub-assemblies with the considerably distorted hexagonal wrappers.

B. The gas gate valves of the irradiated fuel sodium-filled storage drum

At the initial stage we had to replace the sealing elements (made of fluoroplastic). After they had been replaced with the rubber ones (specific rubber) no problems emerged.

C. The sub-assembly transfer mechanism of the transfer chamber

During operation a number of the mechanism design faults have been found. For example, the strip counterbalance rope lifetime is short, indication of the strip position and the gripper directly from the function element is required, fixing of the vertical shaft of the gripper drive is insufficient. By present time the documentation has been elaborated and fabrication of the mechanism backfitting parts is under way.

D. The rotating plugs

The plugs have caused no problems during initial 2 or 3 years of operation. On the basis of the feedback from BN350 and taking into account the experience gained at the BN600 reactor from the monitoring of the protective liquid and hydraulic seal eutectic the former was decided to be periodically replaced and cleaned to remove mechanical impurities.

At present time the protective liquid is replaced once a year in accordance with the reactor maintenance schedule. Up to 1994 pre-operation driving of the rotating plugs caused no major problems - it used to take a number of hours and the time was spent mostly to better heat up the eutectic. Since the autumn outage held in 1994 the rubbing of the large rotating plug and since May 1995 the rubbing of the small rotating plug have been observed and as for the rubbing of the large rotating plug it has never been observed any longer. The rubbing consisted in higher force on the handwheel of the hand drive and respectively in higher drive motor current with the largest increase related to one particular sector. The recurrence of this situation takes place up to now and the level of the rubbing is somehow higher. At present time different options to mitigate and correct the rubbing are being tested. The most likely reason is deposition of sodium in the gaps between the plugs due to violation of the their cooling conditions.

E. The refuelling mechanisms

Over a total period of operation the refuelling mechanisms have caused no major problems.

Operating Experience from Sub-Assembly Cleaning System

Over a total period of operation the core sub-assembly cleaning system performance was sufficiently reliable. Like with the fuel handling system it is the mechanism position indication and control system that has mostly been a trouble contributor with an impact from the process control and monitoring system. The process configuration itself is unsatisfactory since the manifold of the steam-water cleaning cells is combined with the equipment cleaning system (large capacities, risk of overflows and leaks, etc...).

As far as the mechanical part is concerned the following operation results can be noted:

A. The gas gate valves of the irradiated fuel sodium-filled storage drum and irradiated fuel discharge pit

Since the accumulated running time of these gate valves is much more than that of the similar gate valves located on the side of the transfer chamber the intervals between replacements of the gate element rubber sealing rings and between inspections of the gate valve drives are shorter.

B. The cleaning cell wedge gate valves

So far these valves have caused no serious problems.

C. The irradiated fuel discharge pit mechanisms

Of all the mechanisms the conveyor has naturally accumulated the longest running time. All the irradiated fuel discharge pit mechanisms have their design faults (each mechanism has its own design faults) although up to recent time their performance has caused no serious troubles (to provide for this the necessary measures on the periodicity and scope of the maintenance and technical inspection have been taken just after the beginning of operation). Now fabrication of new drives for all the irradiated fuel discharge pit mechanisms is in hand.

Reactor and Primary Heat Tramsfer Systems

In general the performance of the reactor and its in-vessel components is successful.

Control Rod Drives

The control rod drives and the ion chamber hangers are under normal operating conditions. There are 27 control rod drives installed in the reactor.

The specified lifetime of the control rod drives is set up to be 120000 hours. At present time a complex of the material tests in support of the extension of the lifetime is under way.

Sodium-Sodium Intermediate Heat Exchangers

Intermediate heat exchangers are filled with sodium on primary and secondary sides and provide for the necessary heat removal at nominal power.

The performance of the intermediate heat exchangers has been trouble-free. Their lifetime was specified to be 20 years.

In support of IHX replacement a complex of activities including correction of exsisting technical documentation and elaboration of new technical documentation, inspection of a spare IHX and fabrication of new IHXs has been planned. The first IHX has been planned to be replaced not later than 1999.

Primary Sodium Pumps

The primary sodium pumps at the BN600 reactor are of the centrifugal vertical type with a lower hydraulic bearing.

During the power unit commissioning period until March 1981 the pump performance has been trouble-free but during a routine increase in pump speed when reactor power was built up the pump shaft - motor engagement element failures causing trips of the pumps have been observed.

Over period of 1982 to 1984 higher vibration of the pupms, cracking of the shafts, damages to the half-couplings and unreliable performance of the electric drives have been observed.

The damages of the half-couplings were of the fatigue nature. In order to provide for the normal primary pump performance the following activities have been carried out:

- shaft strain measurements that allowed torsional shaft fluctuations within the operating speed range and the coincidence of the drive power output fluctuations with the natural frequency of the torsional shaft fluctuations to be detected,

- replacement of the shafts with the modified ones less in diameter and differing in design from the old shafts,

- redesign of the motor rotor - pump shaft engagement half-couplings,

- introduction of the primary and the secondary pump vibration monitoring systems,

- change-over to the non-controllable operation of the motor (cage rotor) after achievement of the nominal power of the reactor.

The shaft seals preventing gas leaks, the brush-contact unit and the electric drive tachometer generator attachment unit have been also modified. Owing to the actions taken the failures of the pumps causing the abnormal operating conditions have been avoided since the 15th cycle (since December, 1985).

The lifetime of the removable components of the pumps after modification of the impeller was specified to be 50000 hours.

Primary Sodium and Irradiated Fuel Storage Drum Sodium Purification Systems

During operation the purification systems were used to provide for the sufficiently effective cleaning of sodium up to the required standards both on-load and off-load.

As of 1/11/97 accumulation of the sodium oxides in the cold traps has amounted to the following:

cold trap 3ФЛ-1А: 0 kg
cold trap 3ФЛ-1Б: 1264 kg
cold trap 3ФЛ-1В: 214 kg
cold trap 3ФЛ-1Г: 907 kg
cold trap 3ФЛ-БОС: 696 kg

while the rated capacity of each cold trap is 1830 kg.

The cold trap 3ФЛ-1А installed in the primary sodium purification is not filled with sodium and is stored under standby conditions while the cold trap 3ФЛ-1Г is temporally removed out of service after a sodium leak event occurred on October 7, 1993.

Irradiated Fuel Sodium-Filled Storage Drum Cooling System

The cooling system is used to remove the decay heat from the fuel sub-assemblies being located in the irradiated fuel sodium-filled storage drum.

In order to comply with the regulations the backfitting activities including installation of the additional valves and outfitting of the pipelines with guard jackets up to the second gate valves inclusive have been completed on the system.

Steam Generators and Secondary Heat Transfer System

Steam Generators

Over a total period of BN600 operation 12 water-sodium reaction events have been observed in the steam generator modules: 1 in the evaporator module, 5 in the superheater modules and 6 in the reheater modules.

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 precised design concept of a 'Large leak' event and the steam generator emergency protection system operation algorythms 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 5И-А3 has accumulated 7280 trouble-free running hours after the intervention and has been replaced with the new one for material testing.
The restored reheater module 5ПП-Б3 has accumulated 53854 running hours after the intervention and has been removed out of service on indications of water-sodium reaction having totally accumulated 68366 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.

By now the steam generators have accumulated about 115000 running hours each. During the spring outage the planned replacement of all the evaporator modules with the expired lifetime was completed. The lifetime of the replaced modules was extended from 50000 running hours as forseen by the design up to 105000 running hours.

The operating experience from the power plant shows the sectional modular steam generators in use to have high operational reliability. The feasibility of the steam generator operation at the rated power with 7 sections connected has been experimentally validated. Owing to this among 12 water-sodium reaction events occurred only two events have required a loop disconnection and one event necessitated power plant shutdown.

Examination of the failed modules has shown that the most likely reason of a sodium-water reaction is a manufacturing fault failed to be found during anufacturer's tests using applicable inspection methods.Over a total period of operation a series of the following activities on the steam generators has been fulfilled aiming at improvement of their reliability:

- optimization of pre-start and re-agent cleaning procedures,
- correction of the design faults of the main sodium valves,
- modification of the reheater module cover seal units,
- backfitting of the drain and blow-off pipelines,
- modification of the module sodium-water reaction detection systems.

Description of Secondary Circuit Equipment Performance

Over a total period of operation the secondary circuit equipment performance has been sufficiently reliable. The planned scope of work in accordance with the maintenance and in-service metal inspection schedules has been fulfilled on the secondary circuit components and pipelines. During 1995 and 1996 the sodium feeding units of all the cold traps have been repaired.

The overhaul of the main steam generator sodium gate valves has been carried out following the developed procedures.

The work on regeneration of the secondary cold traps in accordance with the procedure developed by the nuclear reactor research centre is in hand.

Table 3 shows the data on the impurity accumulation by the cold traps.

Table 3

Cold trap in-house identification	Accumulated impurity weight, kg
4ФЛ-2А	447,1
4ФЛ-2Б	733
5ФЛ-2А	849
5ФЛ-2Б	888
6ФЛ-2А	604
6ФЛ-2Б	698

The design capacity of each cold trap is 1830 kg of impurities.

Secondary Sodium Plants

Over a total period of power plant operation 2 failures of the pumps have occurred both leading to the power reduction of the plant with a loop disconnection, i.e. in 1980 the motor and tachometer generator shaft coupling failed and in 1985 the electric motor brush unit failed.

The lifetime of the removable components of the pumps was specified to be 100000 hours.

Turbines and Water-Steam Heat Transfer System

In general the turbine performance is successful.

The main pipelines have brought no serious troubles throughout the entire operation period.

The electrical generators are commercially manufactured and proven components.

However over a total period of their operation the events involving the loss of integrity of the stator water cooling system have been observed. Several times these events caused unplanned generator trips. The reason is an imperfect design of the stator rod seals.

Instrumentation and Control System

This system has caused the largest number of the plant generation losses this being accounted for by incomplete applicability of the typical designs used by the designer of the fast reactor power unit. This system had been remarkably modified before it stopped to be a source of the troubles.

Analysis of Power Plant Equipment and System Failures
Categorization of Failures

For the analysis the abnormal operation events related to the reduced and limited power have been selected. The failures of the following eqipment and systems of the plant has been scrutinized:

- the reactor,
- the fuel sub-assemblies,
- the primary and secondary pumps,
- the primary circuit equipment including the intermediate heat exchangers,
- the steam generators,
- the water-steam circuit equipment,
- the turbines,

-the electical equipment,
-the monitoring, control and protection systems of the reactor, sodium pumps and turbine.

Besides equipment the errors of operators and maintenance staff have been taken into account.

The largest percentages of the abnormal plant operation events due to equipment failures are as follows:

- 22.1% (of the total number of events): failures of the monitoring, control and protection systems,
- 12.8%: the electrical equipment failures,
- 12.7%: failures of the primary and the secondary sodium pumps,
- 11.6%: failures of the water-steam circuit equipment,
- 10.4%: failures of the fuel cladding and wrappers.

Consequences of Failures

Totally 104 failures have occurred over a total operation period. The abnormal operation events due to equipment and system failures resulted in 28 plant shutdowns of which 18 shutdowns involved reactor scrams (5 events involved manual reactor emergency shutdowns). In the remainder the plant power reductions took place.

Sodium Leaks

Over a total operation period 27 leaks into the environment of which 5 events involved radioactive sodium leaks have occurred. 14 events have involved active burning of sodium. Five leaks have been caused by the personnel errors.

The main causes of the leaks have been the following:

- insufficient thermal compensation and manufacturing faults of pipelines,
- imperfect valve designs,
- loss of leaktightness of the flange joints of the system of sodium reception from tank cars.

The most serious leak occured in the primary sodium purification system pipeline on October 7, 1993. The volume of the spilled sodium was about 1000 litres but owing to sufficiently effective performanceof the leak suppression system the release of radioactivity outside the plantwas only 10 Ci. The increased exposure of the plant and public did not occur.

The description of the sodium leaks is as follows:

Table 4

No	System	Number	Sizes,l (sodium burning events are underlined)	Number of sodium burning events	Radioactivity release, Ci
1.	Reactor	—	—	—	—
2.	Primary citcuit	5			
2.1.	Gas purification system	1	0,1	—	—
2.2.	Sodium purification system	4	0,3--3,0--0,2--1000	1	0--0,2--0,5--10
2.3.	Sodium storage system	—	—	—	—
3.	Steam generator	1			
3.1.	Leak detection system	1	2	1	—
4.	Sodium reception system	3	10--50--10	3	—
5.	Secondary circuit	17			
5.1.	Main pipelines	—	—	—	—
5.2.	Main valves	4	1--300--30--10	3	—
5.3.	Drain and blow-off lines	9	0,2-1-10-600-300-100-0-1	6	—
5.4.	Drain line valves	1	0	—	—
5.5.	Sodium storage system	3	1--0--0	—	—
	TOTAL:	27	~ 2500	15	10,7

Abnomal Operation Event Causes

The abnormal plant operation events can be divided into 2 categories:

- due to personnel errors,
- due to process equipment failures.

Percentages of these two groups for a total period are as follows:

- events due to staff errors 20.8% of which 12.5% are operator errors, 8.3% are maintenance staff errors;
- 79.2% are equipment failures.

The failures observed in the plant have not caused the operational safety limits to be exceeded and are of level 0 or 1 on international nuclear event scale.

FAST REACTOR OPERATING EXPERIENCE GAINED IN RUSSIA: ANALYSIS OF ANOMALIES AND ABNORMAL OPERATION CASES

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Abstract

Review of various anomalous events and abnormal operation experience gained in the process of Russian fast reactors operation is given in the paper. The main information refers to the BN-600 demonstration reactor operation.

Statistical data on sodium leaks and steam generator failures are presented, and sources of these events and countermeasures taken to avoid their appearance on the operating reactors as well as related changes made in the BN-800 reactor design are considered.

In the paper, some features of impurities behaviour are considered in various modes of the BN-600 reactor operation. Information is given on the impurities ingress into the circuits, on abnormal situation emerged in the process of the BN-600 reactor operation and its probable cause.

Information is presented on the event related to the increased torque of the BN-600 reactor central rotating column and repair works performed.

INTRODUCTION

Three sodium cooled fast reactors are currently in operation in Russia, namely BR-10 and BOR-60 experimental reactors and BN-600 demonstration reactor NPP. NPP with the BN-350 prototype reactor is now on the territory of Kazakhstan Republic. However Russian institutions and enterprises which participated in the design development and construction of the BN-350 reactor are now involved in its operation.

Considerable experience has been gained by the Russian specialists on tests and operation of sodium cooled fast reactors (over 100 reactor-years). Based on this experience, modifications were made of systems and components of the reactors in operation, as well as of the BN-800 reactor design.

In this respect, the experience gained in the process of operation of the BN-600 which is the largest in Russia fast reactor, is of the highest value. Therefore, in this paper, events occurred on the BN-600 reactor are mainly described.

1. ANALYSIS OF SODIUM LEAKS

1.1. Experience gained on sodium leaks

During the operation of domestic fast reactors the following quantities of sodium leaks took place:

BR-10	19
BN-350	15
BN-600	27

At the BOR-60 practically no sodium leaks occurred.

According to their initiation causes the leaks have been distributed as follows:

BR-5/10:

pipe burning-through by electric heaters	2
failures of pump-vessels level indicator sensors	6
sodium valve failures	7
improper procedure of sodium unfreezing	2
manufacture defect	1
crack formation on a pipe	1

BN-350:

flange joint defects	2
improper procedure of sodium unfreezing	6
intercircuit leaks in steam generators	2
mechanical formation of holes as a result of direct actions by personnel	4
uncertain (may be, corrosion)	1

BN-600:

steam generator sodium valve seals	5
flange joint defects	5
improper procedure of sodium unfreezing	4
mechanical formation of holes as a result of direct actions by personnel	2
manufacture defects	3
sodium valve failures	2
crack formation on pipes	6

Burning-through by electric heating

Burning-through of pipes by electric heating occurred at an early stage of BR-5 reactor operation when experience in designing and operation of sodium systems was insufficient, and incorrect decisions were often made. These burns-through occurred due to earthing to the frame of an electric heater and formation of short circuit and electric arc between the electric heater and piping. Later the scheme of heaters' power supply was changed: a transformer with insulated neutral wire was used. This scheme has been used at all domestic fast reactors. No losses of tightness due to this cause occurred any more.

Defects of valves

At the BR-5/10, sodium valve leaks took place as a result of two causes: during the initial period of operation - because of bellows seal defects, and in the subsequent period - because of introduction of valves with improperly designed casings. In the first group of causes sodium did not leak outside the boundaries of a back-up gacket seal (i.e., leak volume did not exceed several cm^3), and in the second group the valve casing was crushed by sodium expanding at heating up but the leaks were also very small. At the BN-600 in one case the leak occurred because of wastage of packing between the casing and the bellows (less than 1 kg of sodium leaked out, no fire occurred) and in another case the leak was caused by a poor-quality joint weld in which some craters developed (also less than 1 kg of sodium leaked out, no burning occurred).

Improper procedure of unfreezing

Rather high percentage is made up by leaks taking place as a result of improper procedure of unfreezing of sodium that was frozen within some section of the system. Unfreezing should be carried out from the free level of sodium, by switching the heaters in a strict sequence, after melting of sodium in the preceding section of the pipe. At sodium expansion within a closed volume as a result of phase transition some seal ruptures of sodium valves, pipes, electromagnetic pump vessels took place. The causes of these events were improper arrangement of heaters, errors in operating instructions, personnel errors. As a rule, leakages took place in the presence of personnel, at changing operating conditions of systems, mainly at start-up or repair work.

Manufacture defects

Four leaks occurred because of poor-quality manufacture of sodium system units. So, at the BR-5/10 immediately after mounting of the primary circuit impurity cold trap a leak occurred at the trap nozzle through a microcrack in the joint weld at sodium heating up in the pipe. The cause of the leak was poor quality inspection of joint welds after completion of assembling. Similar situations took place at the BN-600 as well. Such leaks were detected immediately after putting into operation of a failed section of the system.

Flange joint defects

In sodium systems the joints are made, as a rule, by means of welding. Some exceptions include flange joints of the sodium preparation system. Tanks for sodium transport (for example tank cars) are connected to sodium circuits with the use of removable sections. Often after carrying out of the connection operation some leaks through flange joints appeared. These leaks were immediately detected by personnel carrying out the operations and (or) by monitoring systems.

Sodium gate valves cutting off the BN-600 steam generator modules from the circuit have flange joints which are backed up by welded sealing lips. It was initially supposed that the flange joints would assure tightness, and no due importance was attached to weld joints quality. After appearance of sodium leaks through these seals the sealing lips were welded and weld quality was carefully checked. Subsequently no leaks on gate valves occurred.

Intercircuit leaks in steam generators

At the BN-350 two rather large sodium leaks occurred as a result of defect formation in steam generator heat transfer tubes. In the first case after "sodium-water" reaction occurrence the operation of the steam generator safety system and sodium and water draining from its spaces took place. However, because of personnel errors the cutting-off of the steam generator third circuit and sodium draining were incomplete and into the secondary circuit the ingress of water continued which interacted with sodium remaining in the circuit. Approximately in 5 hours after carrying out activities on steam generator draining a sodium leak from the sodium drainage pipe-line was detected. The pipe break and leakage of sodium with products of its interaction with water occurred as a result of the effect of this reaction products upon pipe material and of high temperature. Leaking out took place through an opening of 10-15 mm in diameter formed in the area of a joint weld at the butt joint of pearlitic and austenitic steel. In the second case leaking out of sodium through the evaporator vessel of other steam generator type took place. As in the previous case, the cause of the leak was a loss of tightness of an evaporator tube and sodium-water reaction in the intertubular space. The failed tube was located adjacent to the evaporator vessel and the leaking-out jet of water was aimed directly at the vessel. The effect of reaction products and of high temperature resulted in the development of the opening in the vessel and sodium leakage.

Mechanical formation of holes as a result of direct actions by personnel

Several sodium leaks were the result of immediate personnel actions. For example, cuts on pipelines at carrying out repair work followed by erroneous supply of sodium at the place of cut. In one case, the leak occurred as a result of a gross error of personnel who started the withdrawal of the BN-350 secondary-circuit sampler-distillator level indicator without cutting off the distillator from the circuit. All these leaks were recorded immediately.

Crack formation

One leak at the BR-5/10 and six leaks at the BN-600 took place through cracks developed at pipelines in the process of operation. All of them were the result either of improper design solutions, or of improper assembling and were related to insufficient temperature self-compensation of pipelines. So, at the BR-5/10 a leak of sodium in the region of drainage pipeline adjacent to the main pipeline took place. Leak volume did not exceed several tens of cm^3. Sodium did not leak outside the heat insulation boundaries. At the inspection of the leak site there was detected a crack in solid metal of the drainage reducer pipe connection. The crack was ~0.5 mm wide and spread to about one half (40 mm) the diameter of the pipe. Adjacent to the through crack on the inner surface of the pipe connection there were microcracks up to 0.4 mm deep. The cause of crack formation was insufficient freedom of drainage pipe movement at changing temperature conditions of the main circuit. Cracks at the BN-600 had an analogous cause and were of the same character.

Defects of level indicators sensors

The BR-5/10 reactor pump level indicators sensors are made of stainless steel tube 20 mm in diameter and with wall thickness of 0.2 mm. They are divided into sections to which current collectors are welded. Leaks took place through cracks formed in joint welds of these sections. After improving the joint welding technology of current collectors there were no cases of level indicating sensors. On the rest of the reactors the level indicators of other design are used. No leaks due to such cause occurred.

Uncertain cause

The cause of one of the leaks remained uncertain. At the BN-350 a rupture of the electromagnetic pump channel wall of the secondary circuit auxiliary system. A possible cause of this event was corrosion resulting from prolonged operation of the pump at pumping of coolant strongly contaminated with sodium-water interaction products.

Thus, of the total number of 61 considered leaks about one third of all the cracks that occurred resulted from erroneous actions of operating or repair personnel. About one half of cracks occurred at repair or start-up activities or at the sodium preparation system (i.e., at the system which in no way is related with reactor safety).

From 27 sodium leaks which occurred during the BN-600 NPP operation the most of them have been small leaks: in 21 cases the amount of poured out sodium did not exceed 10 kg. In the rest 6 cases, the amounts of sodium released were 30, 50, 300, 600, 650, and 1000 kg. Table 1.1. gives the main characteristics of large (over 10 kg) sodium leaks.

Table 1.1. Main characteristics of large sodium leaks in BN-600

Date of leak	Location	Detection method	Cause of leak	Amount of Na leaked
13.01.80	Sodium receipt system	Ionization smoke detectors	Flange joint failure	50 kg
11.08.81	SG valve sealing	Electric heaters monitoring, ionization detectors	Flange joint failure	300 kg
02.07.82	SG valve sealing	Visually by personnel	Flange joint failure	30 kg
31.12.90	SG drain pipeline	Electric heaters	Manufacturing failure	600 kg
07.10.93	Primary sodium purification system	Electric heaters, radioactive aerosol monitoring	Insufficient self-compensation of pipelines	1000 kg
06.05.94	IHX drain pipeline	Visually by personnel	Pipeline cutting before sodium is frozen	650 kg

Sodium fires occurred in 14 cases. All leaks were detected in time by either detection systems or the operators.

The total number of leaks was distributed between the components as follows:

- sodium receipt system 5
- SG modules shut off valves 5
- secondary auxiliary systems 12
- primary auxiliary systems 5

In order to confine and extinguish fires of non- radioactive sodium, powders were used. It was only in case of large leak and fire of primary (radioactive) sodium when the algorithm of sodium burning consequences confinement adopted in the power unit design was realized. This algorithm proved its value: the radioactivity release was much lower as compared to the permissible value. The draining type fire extinguishing systems were not used since there was no need.

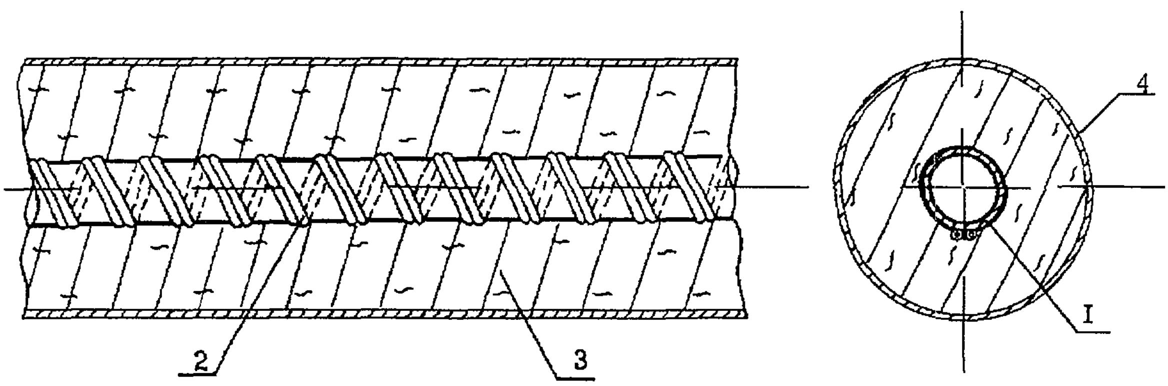
1.2. Sodium leak and fire detection systems

Several different systems of the sodium leaks and fires detection used at our domestic reactors can be considered as an example of the BN-600 reactor where the positive experience of such systems operation has been taken into account.

In the BN-600 several leak and fire detection system types are used, namely:

- the electric heater earthing detection system;
- the radioactive sodium aerosol detection system,
- the smoke detection system;
- the system for gaseous medium temperature measurement within sodium-containing premises.

In Fig. 1 the lay-out of electric heaters and thermal insulation on sodium piping is shown. The heaters have 100% reserve. Earthing of the main and stand-by heaters is continuously monitored. Leaking sodium creates an electric contact between the heater and the pipe wall. Simultaneous failure of both the operating and stand-by heaters is indicative of a sodium leak. Such systems have been installed at all our fast reactors. Their operation experience has shown high reliability and sensitivity of such systems.



1. Pipeline
2. Electric heaters
3. Thermal insulation
4. Jacket of thermal insulation

FIG. 1. Layout of electric heaters and thermal insulation on pipeline.

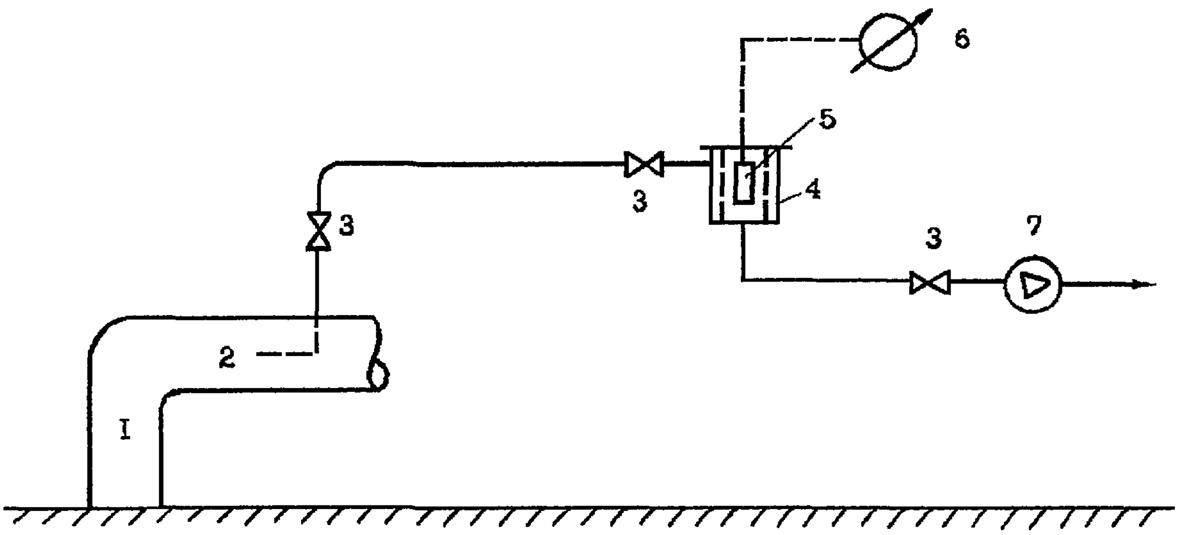
There are two versions of the sodium leak detection systems based on the radioactivity detection principle:

- taking gas samples from the monitored area, pumping them through the aerosol filter and measuring the filter radioactivity;
- measuring radioactivity of air at the exhaust ventilation pipe of the monitored area (radiometer sensor is placed directly on ventilation piping).

The first version of such system is presented in Fig. 2. The air samples from the exhaust ventilation pipeline are pumped through a special aerosol filter where sodium aerosols are accumulated. The radiometer measures the filter radioactivity and generates a signal at an excess of some specified level. This system showed very high operating characteristics.

Large sodium fires can be also detected by the gaseous-medium temperature monitoring systems of the rooms.

Besides, for monitoring of sodium ingress into some cavities the spark-plug-type sensors are used.



1. Pipe of exhaust ventilation system
2. Sampling tube inlet
3. Valves
4. Aerosol filter
5. Sensor of radiometer system
6. Secondary radiometer system
7. Fan

FIG. 2. Sodium aerosol detection system.

1.3. Sodium fire extinguishing systems

Sodium fire prevention and extinguishing measures are mainly provided by the passive means. First of all, these are reactor guard vessel and jackets covering pipeline sections attached to the reactor up to the shut off valves, as well as the main secondary sodium pipeline sections from the IHX to the reactor cell wall. The space between the main and guard vessels is filled with the inert gas preventing sodium from burning in case of leak.

Self-extinguishing method based on the oxygen burning out resulting in the termination of burning of the sodium released into the air filled closed space is widely used. In order to realize this method, pans with covers are installed in the critical cells. The pans are equipped with the devices allowing sodium to penetrate inside but preventing the air oxygen from penetration.

The self-extinguishing method is applied conformably to the whole cells. These cells are made to some extent leak tight (the integrity of the cells is characterized by the leak rate of to 100% or 20% of the cell volume per hour if the cell vacuum is respectively 50 mm and 10 mm water column). This prevents oxygen leaking to the cell from outside so that its content in the cell atmosphere is lowered down to the fire extinguishing value. Using certain algorithm of the ventilation system operation pressure rise in the cell caused by the sodium fire is limited, and directed release of the aerosols to the atmosphere through the special filtering system is provided. This algorithm is as follows (see Fig. 3):

- valves on the incoming ventilation ducts are closed;
- valves on the exhaust ducts of the emergency ventilation system are opened, while the regular ventilation valves are closed;
- emergency exhaust system fan is put into operation.

Besides, draining type systems are provided for the sodium fire extinguishing. This approach is based on the use of a tank located below the cell where the large sodium leaks are probable. A membrane made of material with low melting point is provided over the tank inlet nozzle. The poured out sodium melts the membrane and flows by gravity to the drain tank to be extinguished because of the lack of oxygen.

The active methods of fire extinguishing use the forced delivery of extinguishing media and materials to the area of fire origin. In some cases, for the small cells nitrogen delivery is envisaged. In the cells with non-radioactive sodium transportable fire-extinguishers are used. Also, fixed powder fire extinguishing systems are provided.

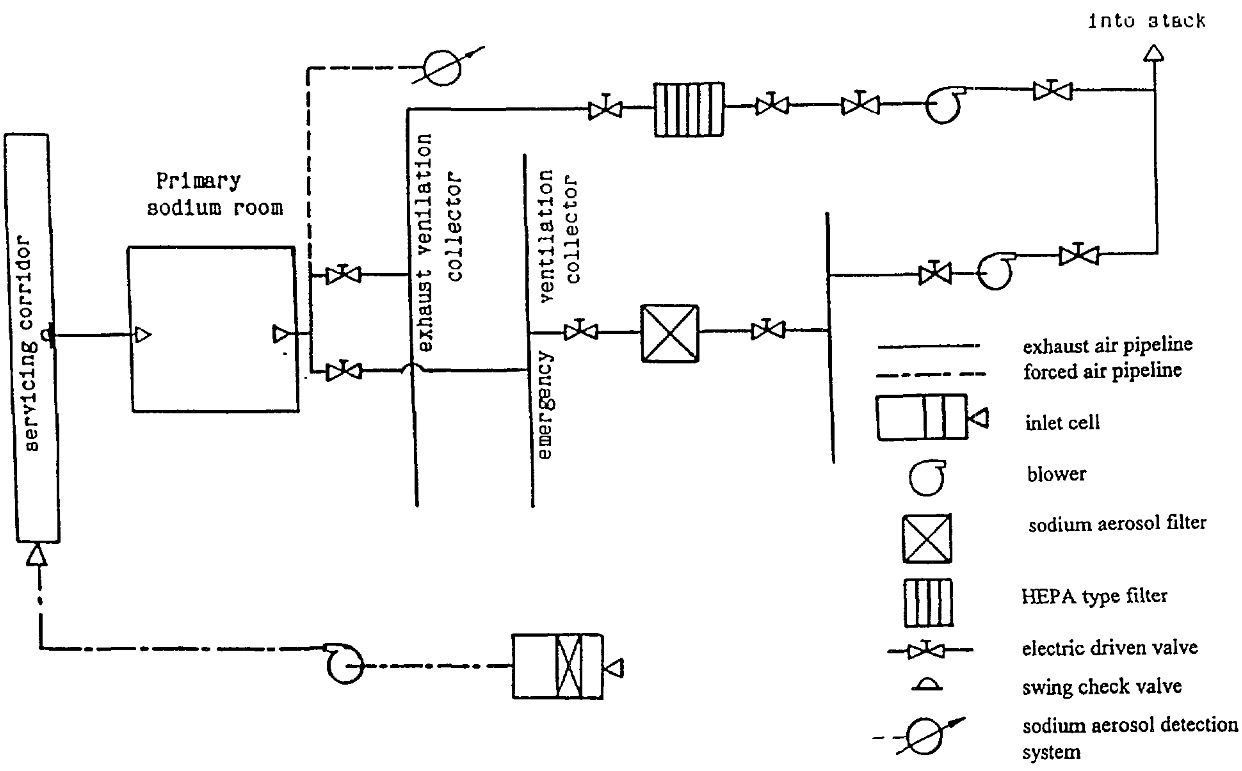


FIG. 3. Scheme of primary sodium room ventilation system.

In order to eliminate direct contact of sodium with concrete structures, inner surface of the cell walls is lined with steel. Thermal effect of the sodium on the concrete is moderated by the thermal insulation layers, placed between the concrete and steel lining.

1.4. Sodium leak and fire detection systems operation

Experience with sodium leaks at our domestic reactors indicates that all leaks without exception, were timely detected. All leaks that occurred on the electrically heated sodium system sections were registered by heaters control systems. In addition, primary sodium leaks were sensed by the radioactive sodium aerosol detection systems. Calculation and experimental analysis of the radioactive sodium aerosol detection system has revealed its high sensitivity and fast response.

In some cases (mainly at sodium valve bellows leakages) the contact leakage detectors (of a spark-plug type) came into action.

Operation experience of sodium leak and fire detection systems indicate that design solutions on these systems have been made properly.

2. ANALYSIS OF LEAKS IN STEAM GENERATORS OF THE BN-600 NPP

2.1. Steam generator of the BN-600 NPP and protection system against secondary circuit overpressure

The steam generator (SG) of the BN-600 NPP is once-through sectional heat exchanger. It involves eight parallel sections, connected with each other by the sodium, water, high pressure steam and reheated steam pipelines. SG is equipped with auxiliary pipelines for sodium filling and draining, gas relief, etc. Shut off valves are provided on the inlet and outlet sodium and water-steam pipelines of each SG section in order to isolate the failed section with the rest ones remaining in operation if necessary. In accordance with the "Requirements...", SG operation with six sections is permitted.

Each section consists of three modules: evaporator (EV), superheater (SH) and reheater (RH) (see Fig. 4). All modules are vertical, straight tube heat exchangers with the lens type compensators of thermal expansion provided on the vessels.

Also, SG involves compensator tank (CT), start-up equipment (two sets for each SG), SG protection system (SGPS), as well as auxiliary technological systems ensuring SG operation in all modes. The SG is equipped with electric heaters, thermal insulation, and necessary instrumentation.

SGPS was designed to provide safe and reliable operation of the steam generator and the whole power unit in case of water/steam leak in the steam generator.

It was taken into consideration when SGPS design was developed to use to the maximum extent the advantages of SG sectional design, namely the possibility of detection of failed section or module at the early stage of the leak development, and isolation of the failed item, i.e. keeping the rest part of the steam generator in operation, on condition that of the isolated item safety is guaranteed.

SGPS includes several subsystems to perform the following tasks:

- leak detection and failed section identification using the set of appropriate devices;

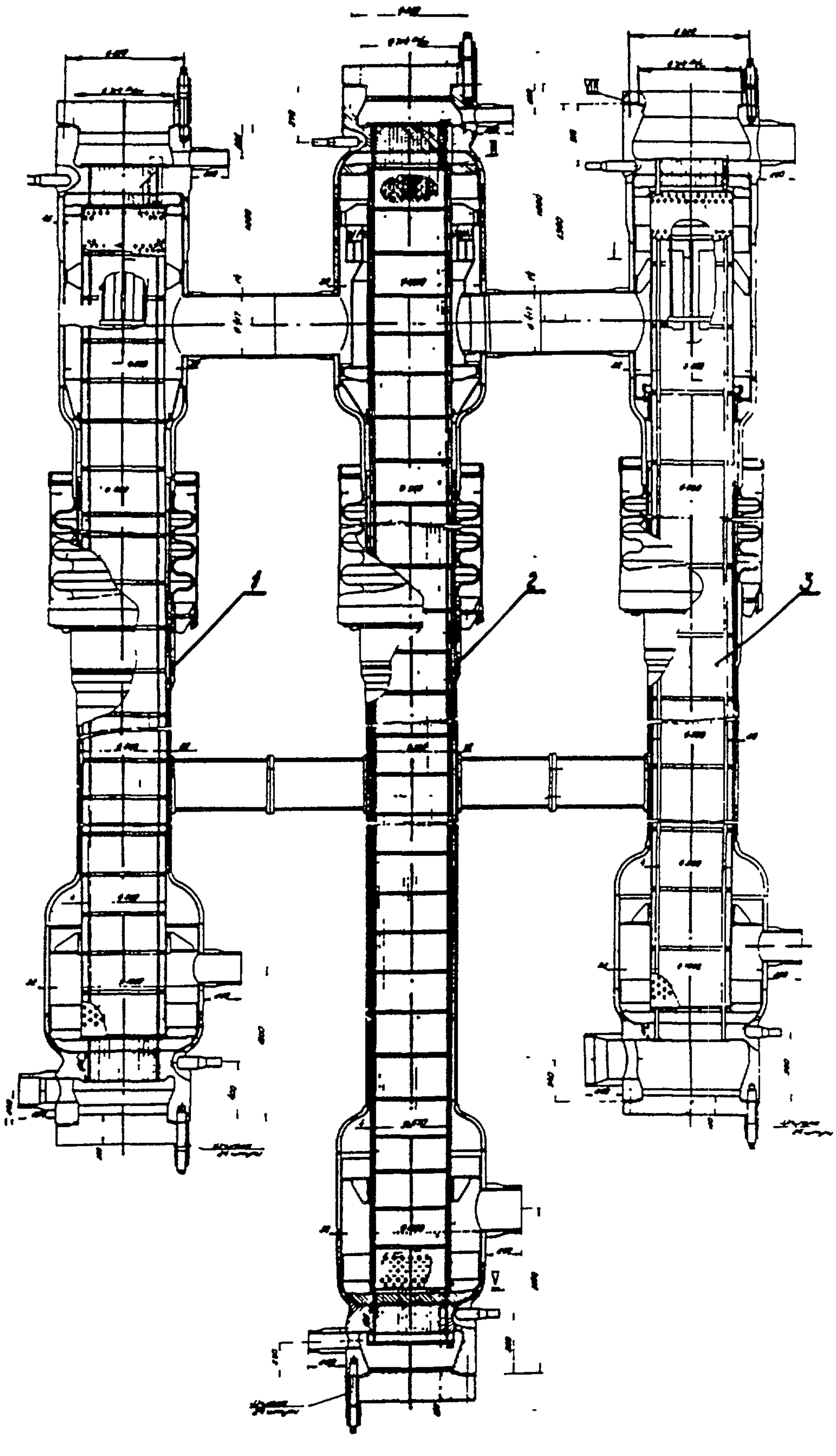


FIG. 4. Section of the BN-600 steam generator.
1 — superheater, 2 — evaporator, 3 — reheater

- isolation of the failed section from the rest part of SG remained in operation, taking measures for limitation of water (steam) amount penetrating into the sodium circuit;

- discharge and separation of reaction products;

- drying of failed section on the tertiary circuit side and filling with nitrogen.

Leak detection in the steam generator is made using the following methods and instruments:

- IVA-1 indicator of hydrogen content in sodium at the outlet of each section;

- KAV-7 conductivity apparatus monitoring hydrogen content in the CT gas volume;

- ITI [and ISHIT , systems detecting gas phase appearance in the sodium flowing through the SG relief pipelines and ИВА-1 inlet sodium pipeline;

- standard pressure sensors indicating pressure rise in the CT;

- standard magnetic flowmeters (300 mm diameter) indicating the increase of the sodium flow rate at the section outlet.

On the basis of the data received from sensors, the indication subsystem forms "large leak" and "small leak" emergency signals using special algorithms.

Also, the recorded information on technological parameters is used in addition to the above data when analyzing leak in the SG.

The isolation of failed section is carried out by closing the following valves:

- gate valves on the sodium inlet pipelines of the superheater and reheater modules and sodium outlet pipelines of the evaporator modules;

- gate valves on the feed water and steam pipelines of the evaporator modules and on both inlet and outlet steam pipelines of the superheater and reheater modules.

Discharge and separation of water-sodium interaction products is carried out by using set of tanks after operation of rupture disc devices (УПМ-200) installed in the CT. Two such devices with the flapping rupture discs using passive action principle, are provided for each SG. The disc is broken spontaneously when 0.245 MPa gauge pressure value is achieved in the CT. The separation of gaseous and condensed reaction products is made in CT and first stage discharge tank (DT-1), separated gas entering the second stage discharge tank (DT-2) through 600 mm diameter pipeline. Two pulse safety valves of 400 mm diameter are installed on the DT-2 in order to release gas into the atmosphere if the permissible pressure value (0.15 MPa) is exceeded. These valves can be urged into the open position by increasing pressure of the working medium in the special piston chamber over 0.03 MPa. The working medium taken from DT-2 is supplied to the piston chamber by opening the auxiliary 25 mm diameter valve with electromagnetic drive. The medium is taken from DT-2 to the pulse valve.

After the evaporator and superheater modules have been isolated, their drying on the tertiary circuit side is carried out by forced opening of the discharge devices with the subsequent filling of dried modules with nitrogen.

In case of the SG shutdown, steam is removed from the reheater module to the turbine condenser via intermediate pressure cylinder, while in case of single section shutdown, draining pipelines are used for this purpose.

Protection of the steam generator and secondary circuit against the overpressure in case of water/steam leak is provided by the following measures:

- in case of "small leak" signal formation, isolation of failed section on the operating steam generator is made by an instruction from the operator, with its drying on the tertiary circuit side;

- in case of "large leak" signal formation, shut-down of steam generator is made by an instruction from the operator with its drying on the tertiary circuit side;

- in case of CT pressure exceeding 0.23 MPa value, automatical shut-down of the steam generator on the operating power unit is made with its drying on the tertiary circuit side;

- in case of CT pressure exceeding 0.245 MPa value, spontaneous passive break of the rupture disc and reaction products discharge through CT to DT-1 are realized.

Since late 1981, the steam generators have been operated on the rated power level with design parameters. Now, scheduled replacement of evaporator modules having been in operation over 100 000 hours (i.e. those with expired life time) by the new items have been complete in 1997. Operating experience of SGPS has demonstrated its sufficiently high reliability. Water into sodium leaks (including large ones) occurred in the steam generators only resulted in replacement of SG structural elements in which the reaction took place. There haven't been any more serious consequences of the steam generator failures. No serious failures of SGPS affecting SG safety have occurred during the whole period of operation.

2.2. Review of water/steam-to-sodium leak events in steam generators of the BN-600 NPP

The sectional-module SGs have demonstrated high operating robustness in the event if inter-circuit leaks. During the entire period of SG operation there have been twelve water/steam-to-sodium leak events, half of which occurred in the first year of operation and were caused by development of latent manufacturing defects. Inter-circuit leaks were mainly in the superheater modules (six events) and in the reheaters (five events), while in the evaporators there was only one leak event (Table 2.2).

In spite of the leaks occurred in the steam generators module and section SG design has proved its advantages by providing planned rate of the reactor power gaining and high performance. Average value of load factor for the period from reactor initial start-up till 1996 was 69.3%. Steam generators with related systems have not caused any load factor decrease during the last five years.

2.3. Description of large leak in the BN-600 superheater on 19 January, 1982

The power unit was put into operation after some scheduled maintenance about two months before the leak occurred, and its operation was stable on 90-100 % power level. By the time of the leak appearance three loops were in operation, the unit power being equal to 93.6%.

No.5 steam generator performance at the time of leak:

- sodium temperature at the SG inlet	492 °C
at the SG outlet	298 °C
- fresh steam pressure	11.2 MPa
- fresh steam temperature	489 °C
- SG steam capacity	463 t/hour

Reheater module of 5B1 section of the steam generator was shut down. All detectors of water-into-sodium leaks were serviceable and they were in operation.

Table 2.2. Water into sodium leaks and leak detection system operation on the BN-600 steam generators.

No of leak / Parameters at the time of leak	1	2	3	4	5	6	7	8	9	10	11	12
1 Modules	RH	SH	RH	SH	SH	SH	SH	SH	EV	RH	SH	RH
2 Leak size	L	L	S	S	S	S	L	S	S	S	S	S
3 Date of leak	24 06 80	04 07 80	24 08 80	08 09 80	20 10 80	09 06 81	19 01 82	22 07 83	06 11 84	10 11 84	24 02 85	24 01 91
4 Time of work before failure, hours	1000	968	1145	1454	950	1640	4019	19584	26032	14512	26078	44000*)
5 NPP electric power, MWe	270	65	313	362	332	210	550	606	240	600	400	596
6 Secondary Na temperatures (SG inlet/ outlet), °C	460/300	314/299	465/300	468/298	460/299	401/300	500/301	506/304	510/305	510/305	480/300	513/315
7 Tertiary circuit parameters, - feet water temp , °C	156	-	162	163	159	159	164	164	238	240	163	240
- temp of live/reheated steam °C	440/432	-	450/453	461/453	456/447	307/187	490/483	501/493	504/497	506/496	470/462	504/499
-SH/RH pressure (MPa)	11 2/1 0	-	10 8/0 92	10 3/1 31	11/1 03	5 5/0 36	11 2/2 2	12 1/2 2	12/2 2	12 1/2 2	11 9/2 8	11 9/2 1
8 Time to reach emergency value setting, min	-	-	4	5	8	5	2	7	9	-	5	4 5
9 Water into sodium leak flow rate,g/s	0 02-6	0 1-0 615	0 09-15	0 2-0 3	0 0064- -0 23	140	250	-	0-3	0 02	0 14	4 6
10 Amount of water escaped into secondary circuit, kg	40	17 87	7	0 18	0 78	40	20 3	2 77	1 8	0 75	0 73	8 3

*) After repair work had been completed on the module failed in case 10

Background values of hydrogen content in sodium and in gas were respectively (0.08 – 0.15) $\cdot 10^{-6}$ and 0.016 vol.%.

At 16:11 on January 19, 1982, signal of IVA device indications increase was observed in 5B1 section. No indication changes were detected in other monitoring devices. Operator was sent to the IVA device area in order to check its indications.

At 16:15, abrupt increase of indications of all IVA devices installed on the outlet of the SG-5 all sections started (it should be taken into account that the period of sodium flow through the circuit is about 2 minutes). However indications of ITI, ISHIT and KAV-7 devices did not change, causing some doubt with respect to the IVA's indication growth, and the latter was explained by erroneous actions of I&C man sent to the local control board.

Later, it was revealed that ITI device had indicated "leak" twice (at 16:12 and at 16:20), but these were two outline points on the recorder, missed by the personnel. Though, according to the instruction in force at that time signals of such a kind should not be taken into account.

At 16:23, ISHIT system showed "leak" in 5B1 section. At the same time, pressure increase started in the pressure compensator vessel. All these signals were perceived by the personnel as a large leak indication, and SG-5 was shut down at 16:24 using "large leak" switch with automatic realization of the emergency algorithm (shutdown of the main sodium pump and drying out of SG on the water side).

At 16:30, indications of KAV-7 device increased abruptly beyond the scale (by 5%). According to the indications of IVA device installed on the failed section, hydrogen content there reached $30 \cdot 10^{-6}$ value, while it was (3-5) $\cdot 10^{-6}$ in other sections.

After decreasing its power down to 64% the reactor remained in operation on two loops, but 4.5 days later the third loop (except for the failed section) was put in service.

Retrospective analysis of indications of SG-5 leak detection devices and other system parameters made it possible to recreate the following picture of the process:

- at 16:10, steam into sodium leak of 60- 100 g/min flow rate occurred in the superheater, resulting in the increase of the SG section pressure drop on the secondary sodium side and about 2% sodium flow rate decrease from the initial value; IVA indications growth in this section began 1 minute later (Fig. 5);
- during 13 minutes, steam leak flow rate increased up to 500 g/min, while sodium flow rate in the section continued to decrease (see Fig. 6), maximum reduction being 16.5% from the initial value, and sodium flow rate through the other sections increased by 30% as compared to the initial level;
- at 16:23, an abrupt increase of the leak up to 15000 g/min occurred resulting in the rapid formation of hydrogen bubble in the module and pushing sodium into the headers. Sodium flow rate at the failed section outlet increased by 1.35 times; sodium level in the pressure compensator increased by 250 mm. Rupture disc installed on the pressure compensator burst spontaneously at 0.18 MPa pressure value;
- after "large leak" automatic algorithm had been initiated, both SG and secondary loop were shut down and SG was dried out.

In the course of shutdown, steam pressure in the section was reduced down to 0.6 MPa during 50 s using safety valves, with their subsequent closure.

Six minutes after SG shutdown both inlet and outlet sodium valves of the suspected section were closed, and its draining started. During 1 hour, 17 m^3 of sodium was drained, and plugging of the evaporator drain piping was revealed.

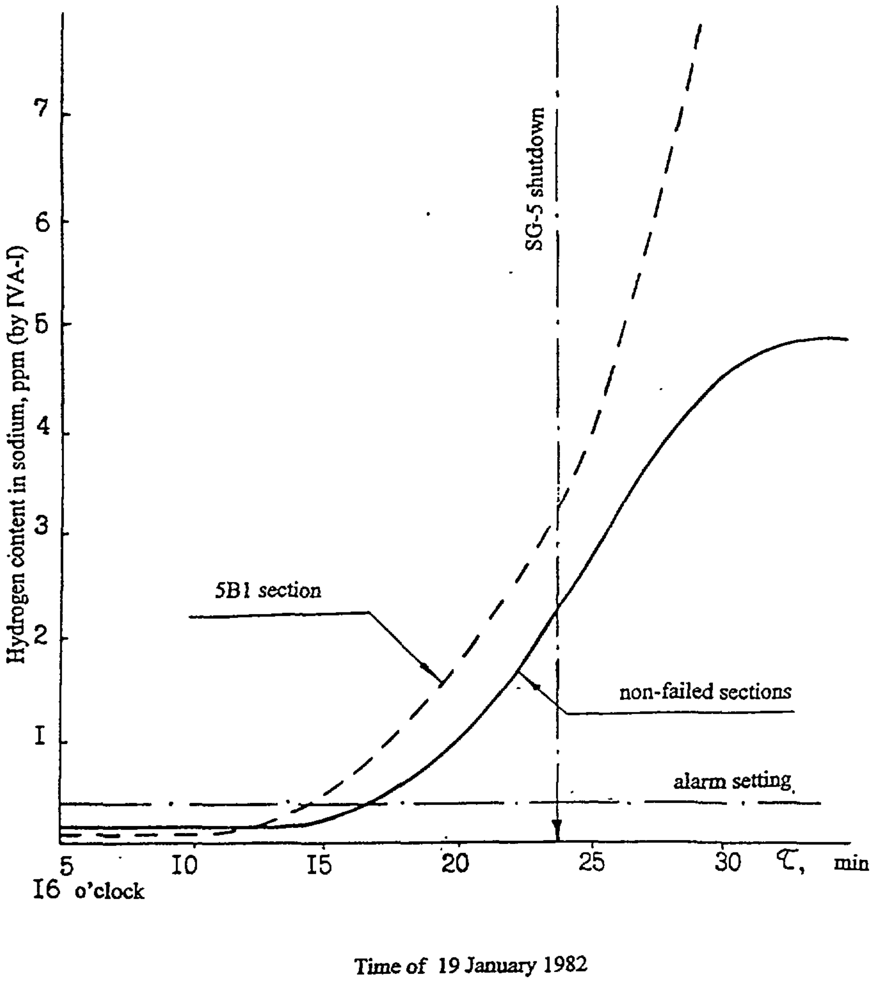


FIG. 5. Behaviour of hydrogen content in sodium caused by SG-5 leak.

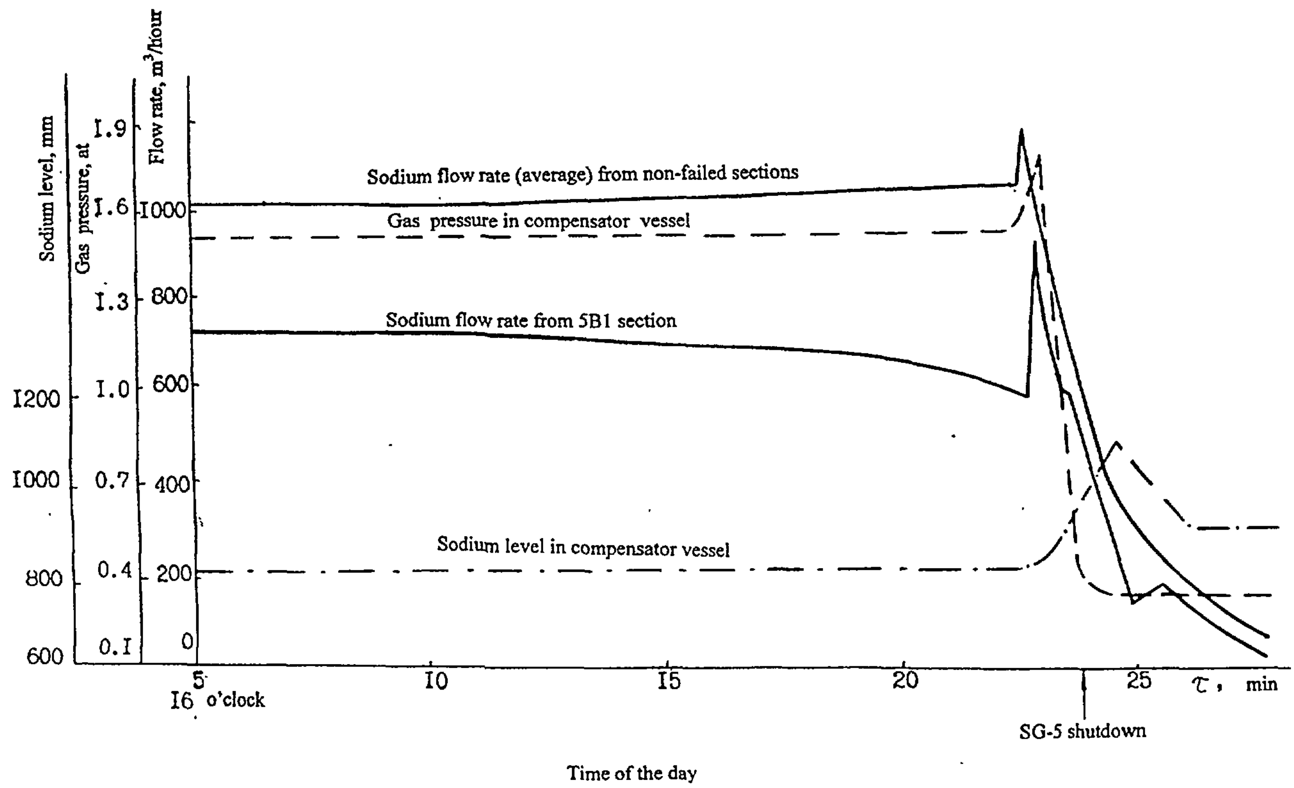


FIG. 6. Change of secondary circuit parameters caused by SG-5 leak.

Eight minutes after SG shutdown, water valves at the module inlet were closed, while outlet valves were closed in ten minutes. Then steam replacement with nitrogen was initiated.

Evaluation of steam amount penetrated into the secondary circuit was made using final values of hydrogen content in sodium and gas as well as gas volume increase. This steam amount was 20.3 kg, including 5.4 kg of water penetrated during initial 13 minutes after leak start and the rest 14.9 kg penetrated during SG outage. Besides, in spite of rapid supply of nitrogen into the water circuit, about 200 kg of sodium entered the module lower chamber and outlet steam piping.

All IVA-1 devices of SG-5 detected steam ingress in sodium and leak evolution. Other systems failed to operate, that resulted in hampering and delaying decision on the SG shutdown:

- KAV-7 gas analyzer reaction to hydrogen appearance in pressure compensator vessel was correct, but sufficient hydrogen content was reached there already after the SG had been shutdown;
- both ITI and ISHIT systems responded to the steam leak, although their response was short, so neither "leak" signal was formed, nor any signal was put out to the operator's display. This in particular was because of the imperfection of the signal processing algorithm.

Experience gained on this and some other leaks has shown that ITI system operation strongly depends on hydraulic mode of the secondary circuit and leak location. Later on, experimental studies and analysis of this system performance were carried out, thus allowing to refine its role in the set of leak indication devices.

"Large leak" signal based on gas pressure increase in the pressure compensator vessel was not put out because of burst of the rupture disc that had happened before pressure safety settings were reached in the compensator vessel. Absence of displayed information and KAV-7 gas analyzer and ITI system responses which were habitual in connection with the previous leaks, as well as unusually rapid growth of IVA-1 indications similar to that in case of device failure - all this made personnel doubt in the correctness of these devices. As a result of this, personnel did not manage to understand the situation and decision on the SG shutdown was made with delay.

After additional attributes had appeared confirming leak occurrence, personnel began to act in more positive, proper and timely manner, so that no further propagation of the leak was allowed.

Identification of the failed module was made by in-turn pressurization of the SG modules water side with nitrogen, its pressure decrease being monitored. Tests revealed superheater failure of 5B1 section.

After its draining the module was pressurized with air, and 7 failed tubes were revealed in the central area of the tube bundle (Fig. 7). In the course of dismantling of the tube bundle it was shown that through defects, i.e. local thinning of the tube walls caused by metal dissolution were found on tube sections between spacer grids at about 6500 mm distance from the upper tube plate (Fig. 8). Wall thickness decrease was revealed on two more tubes. Metallurgical defect of the tube wall, which had not been revealed on the stage of incoming gauging, and which manifested during operation, could be the cause of the initial leak in this case.

Repair works were devoted to the replacement of failed module with the new one.

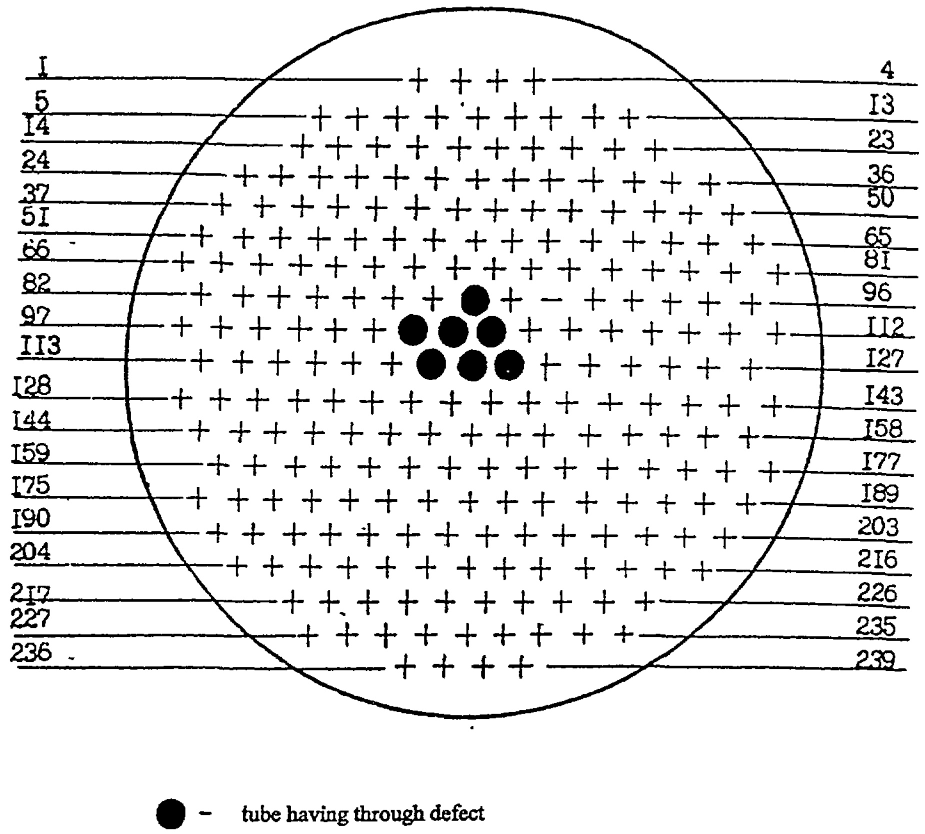


FIG. 7. Cartogram of failed tube locations in 5SB1 module.

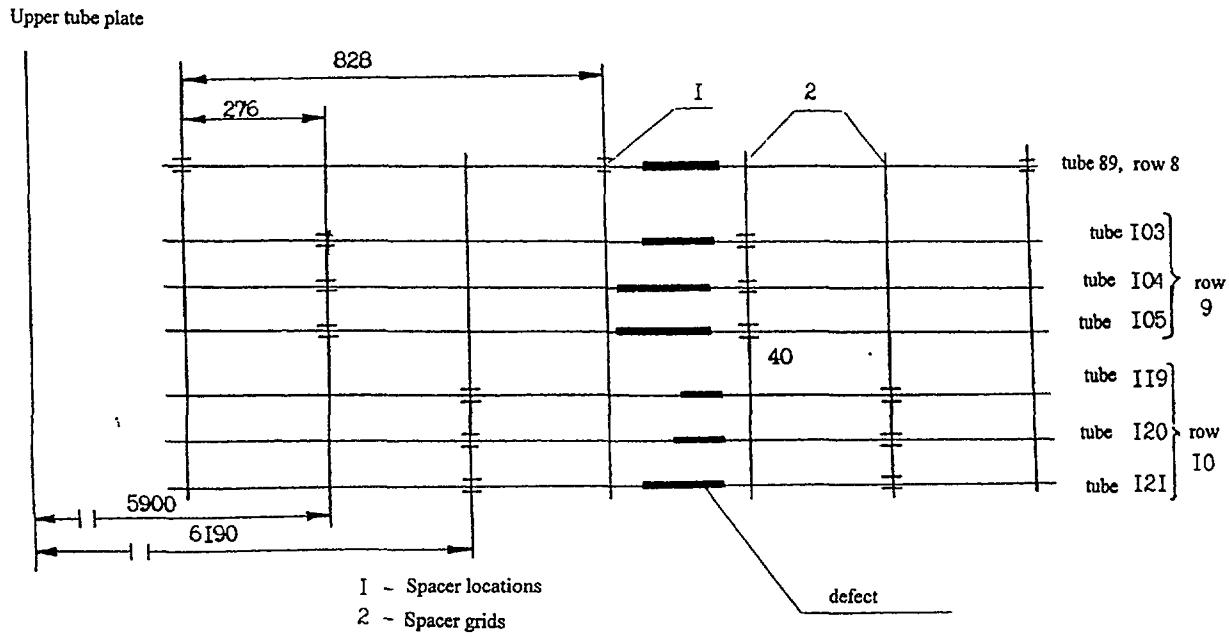


FIG. 8. Locations of through defects in the tubes of 2SB1 module.

2.4. Improvement of steam generator protection system

Considerable modifications of SGPS in BN-600 NPP were made during the operation period:

- in order to prevent gas entrainment, sodium level in CT was increased, resulting in decrease of the gas compensating volume from 18 m^3 to 14.5 m^3;
- flapping rupture discs based on passive principle were installed instead of forced rupture discs;
- number of discharge pipelines from CT to DT-1 was decreased from four to two;
- pipelines of sodium relief from SG modules were modified in order to provide continuous sodium flow;
- ISHIT ' sensors were installed on the modified relief piping to monitor water into sodium leaks in each module (especially, in the vicinity of upper tube sheets);
- algorithms of "small leak" and "large leak" signals formation were modified (in particular, the readings of 300 mm diameter magnetic flow meters installed at the outlet of each SG section, were introduced for making "large leak" signal);
- in order to decrease transport time of the analyzed gas sample to the sensor, the gas flow rate through the KAV-7 device was increased from 3 l/hr to 12 l/hr.

Experience gained in BN-600 SG operation has been taken into account in the BN-800 SG design. It is for the first time in the home design practice that the BN-800 steam generator is equipped with double-level safety system for the case of tube failure causing water leak

The first level system which is usually called automatic safety system (ASS) is actually preliminary protection stage. It is designed for failed SG section location on the early stage and its shut-down, steam generator being kept in operation on full or lowered power (depending on specific situation). This system provides automatic draining of isolated section on both sodium and water sides. Algorithm of automatic shut-down of the failed section provides its independent simultaneous isolation on both sodium and water sides, followed by opening of discharge paths.

In case of rated (design) leak process evolution, only the first level safety system should operate.

The second level, i.e. the general protection stage provides the whole steam generator isolation and its drying out on the water side, as well as protection of the secondary circuit against overpressure.

In case of emergency, SG is shut down using automatic algorithm. Related signal is formed on cover gas pressure increase in the pressure compensator vessel (PCV) up to the alarm setting. If one of safety rupture discs bursts spontaneously before the alarm setting is reached, the algorithm is initiated by the gas pressure decrease signal.

Algorithm initiation and SG shut-down control are fulfilled automatically. In case of failure of algorithm automatic operation, SG should be shut down by operator.

Algorithm of the SG emergency shut-down includes the following procedures:

1) shut-down of secondary sodium pump and closure of quick-acting and shut-off valves installed on the SG water circuit common pipelines;

2) automatic steam discharge from SG through the safety valves, installed on the discharge header - after valves closure according to 1);

3) automatic or remotely controlled drying out of both SG and feed water route using draining pipelines - after valves closure according to 1);

4) automatic closure of sodium valves installed at the inlet of SG sections;

5) remote nitrogen supply to the water side of the SG.

Analysis of the BN-800 reactor SG and related safety system design have shown that there are some approaches adopted to prevent propagation of dangerous situation similar to that occurred on the PFR:

- devices for leak detection, using various principles and duplicating each other to some extent (this is to facilitate leak detection on the early stage);
- most components of the SG safety system are backed up, and in case of preliminary safety system stage failure to shut down SG section, the whole steam generator is shut down;
- reliable coolant release from both secondary and water circuit is provided.

It is very important, that according to the established rules of operation of both BN-350 and BN-600 power units, safety devices and related instruments can only be switched off for a short time with the intensification of the SG performance monitoring by the operator for this period.

3. ANOMALOUS EVENT ON THE BN-600 REACTOR CAUSED BY IMPURITIES TRANSFER IN THE PRIMARY CIRCUIT

3.1. Analysis of impurities behaviour in the primary circuit

Maintaining required purity of the coolant and cover gas is a necessary condition for successful operation of any reactor facility. In case of sodium cooled reactor these requirements are determined by Specifications on sodium coolant for nuclear reactors (ТУ – 6 – 01 – 788 – 73) and Rules of the BN-600 reactor operation. They are most rigid for the primary circuit, since nuclear safety of the reactor is immediately concerned.

Content of impurities in the coolant is determined by the source/release rate relationship in the system. Not only ingress and removal of impurities from the coolant (in more general sense - from the system) should be understood as sources and releases, but all processes influencing forms of impurities existence.

Under normal operating conditions of the reactor on power, the main sources of impurities entering the coolant are as follows:

- structural materials corrosion products;
- cover gas supplied to the system in the process of system replenishment or necessary procedures;
- lubrication system of centrifugal pumps;
- hydrogen and tritium, produced in the steam generator and in the core, entering the coolant owing to diffusion.

Analysis of behaviour of structural materials corrosion products in the BN-600 type commercial reactor has shown that the main amount of corrosion products (which can reach ~100 kg/year if oxygen content is ~10 ppm) is not removed by the cold traps but accumulated in the circuit as deposits to all appearance mainly on the heat exchanger tubes. Their genesis in the process of long-term operation of the reactor plant requires carrying out special studies. The main attention should be paid to the issue of their possible mass ingress into the coolant and development of measures preventing from negative consequences of such event if there is any probability of it.

During 1981-1984 period, average amount of argon supplied to the reactor cover gas plenum (CGP) was $\sim 4 \cdot 10^3$ nm^3/year, while during maintenance work period, from 2.5 to 68 m^3 of air entered the circuit (evaluations were made on the basis of nitrogen content increase in the CGP). As a result of this, up to 6 kg/year of oxygen entered the circuit, water amount being ~ 0.5 kg/year. The most portion of these impurities was brought when some works were carried out on the reactor.

According to the evaluations made by the pump designers, rate of the oil ingress into the circuit does not exceed 60 g per year from each pump.

The range of sources of the impurities extends if any maintenance or emergency works are carried out, and their capacity sometimes increases several orders.

Impurities release is possible through the following equipment:

- purification systems, such as cold traps;
- circuit sections, where accumulation of impurities is possible because of hydrodynamic and temperature conditions (thermodynamic parameters).

In the latter case the following systems can be taken as examples: stagnation zones in the bottom of reactor vessel and sodium/structural material and gas/structural material interfaces.

In the analysis of probable distribution of impurities in the primary circuit, complicated structure of the BN-600 reactor should be taken into account, namely the fact that several hundreds of the components having various operating temperatures and thermodynamic parameters, are located in the reactor vessel. Under nominal operating conditions, temperature difference in the sodium circuit reaches 200°C value. Taking into account that in spite of the designers' efforts some stagnation zones still exist in such complicated system, and these zones are very slightly involved in the process of mass transfer with the bulk of sodium, conclusion can be made on that the real temperature difference in the reactor vessel exceeds 200°C.

When there is no coolant flow in the circuit, temperature difference along the main route of sodium is insignificant. However under these conditions, the risk of stagnation zones appearance in the vicinity of the vessel bottom is high. It should be taken into account in the analysis of this phenomenon, that "cold" sodium from the cold trap enters lower section of the reactor vessel and both reactor pit and reactor guard vessel are cooled permanently by air.

These factors create prerequisites for impurities accumulation in the zones with limited mass transfer to the main sodium flow, namely surface of cover gas plenum of the reactor and possible stagnation zones. At the same time, corrosion products would deposit on the non-isothermal surfaces of the intermediate heat exchanger until the certain moment determined by the critical thickness of deposits layer.

These points should be taken into account when analysing unusual behaviour of impurities, considered below.

Some features of impurities behaviour in the BN-600 reactor primary circuit are observed during reactor transients, such as increase of temperature and power, shut-down/start-up modes of the loops, reactor scram etc.

In November-December 1981, one of the early cases of abnormal behaviour of plugging meter temperature (referring to the content of impurities soluble in the coolant) was detected. Parameters of the system prior to this event and their further evolution are shown in Fig. 12. The increase of the coolant temperature was preceded by various procedures related to the replacement of the primary components. These resulted in the ingress of large amount of air into the CGP. In the course of this work, sodium was purified using cold trap, and the plugging temperature was always lower than that of the flowing sodium, this difference being over 100°C just before the increase of temperature.

Stable growth of the plugging temperature was detected after 325°C value had been reached. Further temperature increase resulted in several times higher rate of the plugging temperature growth. During the first half of the day of December 1, ingress rate was over 150 g/hour in terms of hydrogen. After purification system capacity had been increased over two times, plugging temperature started to decrease, and it took eight days to complete purification of the total amount of the coolant which passed four times through the cold traps. No abnormalities have been detected during further on-power operation of all primary system components.

By now, monitoring of non-metal impurities behaviour in the primary circuit and their possible effect on other parameters of the sodium coolant (temperatures, flow rates and levels) has been set up on the regular basis.

For instance, in some cases of the cold condition reactor start-up, hydrogen content in the reactor cover gas increases from 0.0005 to 0.3 vol. % within 220 - 380°C temperature range, followed by its stabilisation in the range from 380 to 400°C and subsequent gradual decrease down to 0.05 vol.% with further temperature increase up to 530-540° C. This type of hydrogen behaviour can be explained in the following way: hydrogen content growth is obviously caused by the hydroxide decomposition and sodium hydride dissociation, taking place at considerable rate in the temperature range from 280 to 360°C, while the reduction of hydrogen content in the cover gas is resulted from the reverse process, since the rate of hydrogen interaction with sodium surface increases at temperatures over 380 - 400°C.

Hydrogen containing impurities can penetrate into sodium during refuelling operations when fresh fuel is loaded into the core, and also in the course of maintenance and components replacement works. Specific reactor operating modes and stagnation zones favour deposits accumulation in the line ends, although sodium purification in the cold traps is carried out permanently during reactor refuelling.

If at least one loop or the reactor is shut down, hydrogen and nitrogen content increase is sometimes observed in the cover gas. For instance, in February 1989, when one loop was shut down followed by the reactor scram, contents of hydrogen and nitrogen increased respectively from $2 \cdot 10^{-4}$ vol.% to $3.2 \cdot 10^{-3}$ vol.% and from 0.03 to 0.05 vol. %. The origin of these phenomena has not yet been discovered, and studies on impurities behaviour in the reactor are going on.

3.2. Description of abnormality

During the BN-600 reactor operation, spontaneous abrupt changes of the parameters occurred related to some operating modes. One of the sources of these parameter changes could be the changes of hydraulic characteristics of the primary circuit caused by the process of non-metal impurities transfer in sodium.

It was shown by the BN-600 reactor operating experience, that not a single change of the reactor parameters having been observed since the reactor was put into operation resulted in the effect on the core processes, except for the event occurred on 21 January, 1987. However, there were no abnormal operating conditions in all these cases including the above-mentioned event.

On January 21, 1987, No.3 power unit of Beloyarskaya NPP was in operation on the power level of ~ 96.6%.

During the period from 14:29 to 14:36, spontaneous change of some reactor parameters was detected (Figs. 9-11). This resulted in redistribution of power between the primary loops without any change of reactor thermal power (there were no changes in the current readings of ionisation chambers). This, in its turn, caused slight redistribution of electric load of the turbine generators. There were the following changes of the reactor parameters during this period (6 minutes):

- automatic insertion of AP-1 control rod into the core to the depth of 90 mm (~ 65 mm during the first minute and ~25 mm during the next ~4 minutes);

- increase of the cover gas pressure by about 5% with its further decrease to the initial value;

- sodium level fluctuations in the reactor vessel within ± 50 mm, followed by its stabilisation on the level of 25 mm below the initial value;

- increase of sodium level in the primary pump tanks by 50÷140 mm;

- reduction of power consumed by the primary pumps by 1.5÷2%;

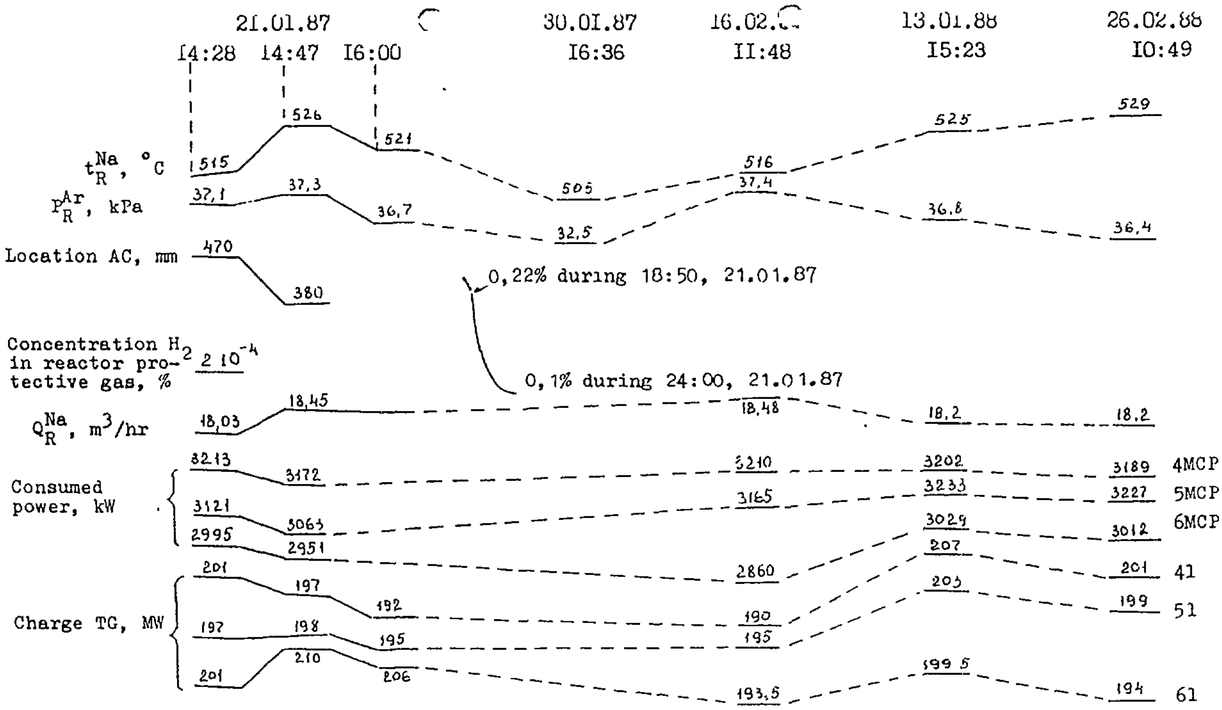


FIG. 9. Reactor BN-600 parameters variations 21.01.87–26.02.88.

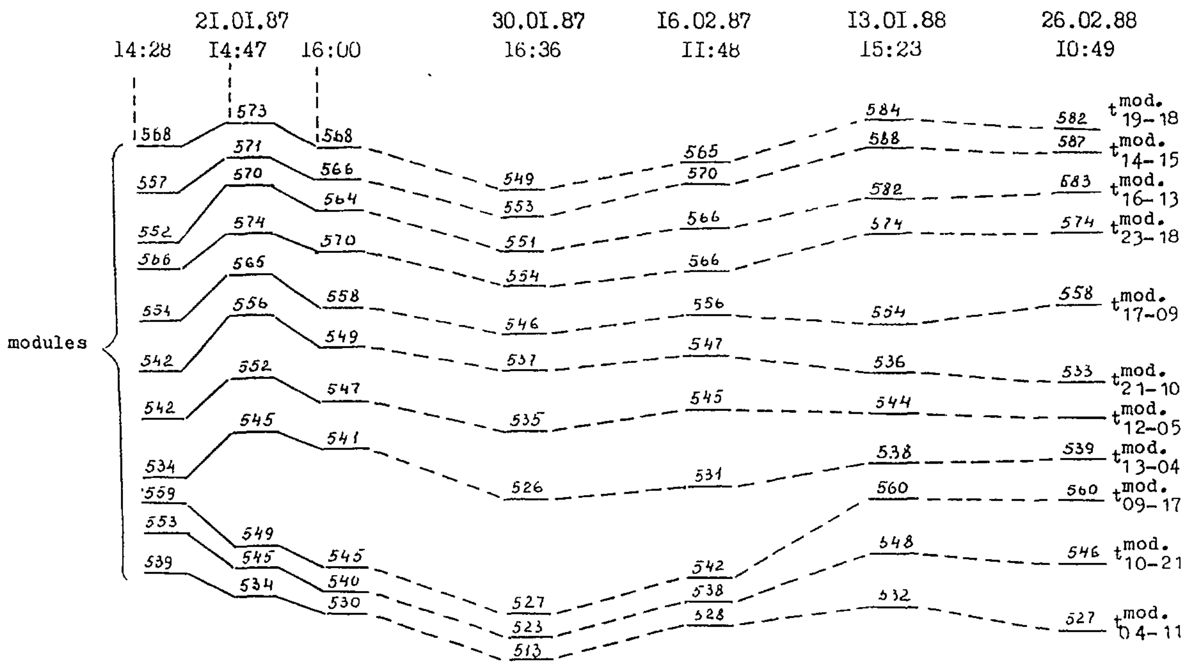


FIG. 10. Reactor BN-600 parameters variations 21.01.87–26.02.88.

- changes of sodium temperature at the SA outlet within ± 10°C (temperature increase was observed in the third sector, while it decreased in the first sector);

- changes of sodium temperatures at the intermediate heat exchangers inlet and outlet.

Further, reactor power was decreased down to 94.7% and then to 86% from the rated value.

The analysis of parameters showed, that ~0.031 % Δk/k positive reactivity had been inserted into the core during this period, while the core pressure drop increase and sodium flow rate decrease by ~2.5÷3% took place.

All parameter changes were within the operational permissible limits, and they were easily compensated by the automatic control systems without intervention of the operators. Personnel support was only required on the water-steam circuit in order to bring parameters of No.6 turbine generator back to the normal values after all the transients were over (pressure regulator of TG No.6 was opened for pressure reduction in the controlled stage chamber of the turbine).

Impurities content measurement system operated on the periodical basis, so the first measurements of hydrogen content and plugging temperature were made respectively ~4 hours and ~ 2.5 hours later. Content increase values for hydrogen, nitrogen and helium were respectively ~ 0.22 vol.% (against usual value of $\sim 2 \cdot 10^{-4}$ %), 0.06 vol.% and 0.03 vol. %, while the plugging temperature increased from 120°C to 155°C and then to 170°C (34 hours after start of the event). It is necessary to notice that this was the only case of changing contents of hydrogen, nitrogen and helium in the CGP under steady state conditions.

According to the evaluations, ~90 g of hydrogen entered CGP, while ~ 110 g was added to the primary sodium, the total amount of hydrogen ingress being ~200g.

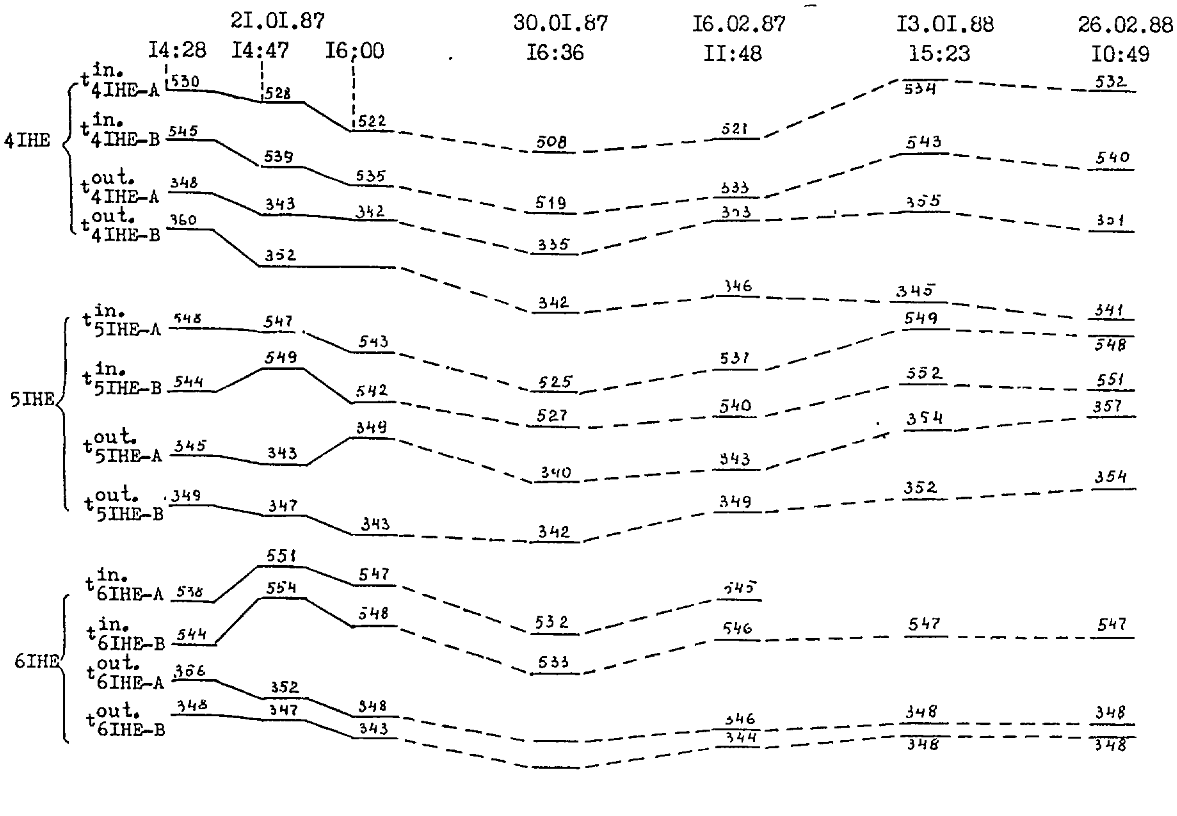


FIG. 11. Reactor BN-600 parameters variations 21.01.87–26.02.88.

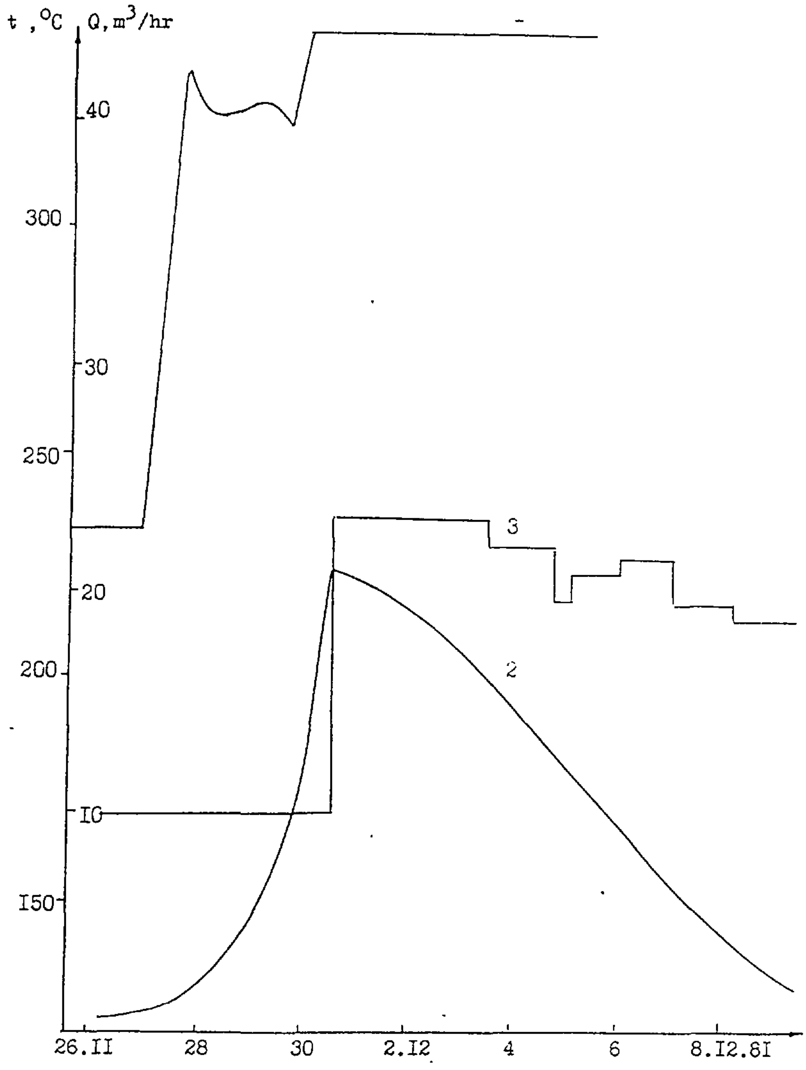


FIG. 12. First circuit main parameters variations from 14.11.81–10.12.81
1 — reactor sodium temperature
2 — plugging temperature
3 — sodium flow rate for purification

Analysis of reactivity change showed that ~70 g of hydrogen was in the core (in order to release reactivity value of ~β(0.7 %Δk/k), ~2000 g of hydrogen should be introduced to the core).

Just after penetration of the impurities into the primary sodium, redundant cold trap was put into operation. After ~10 days of two cold traps operation, the plugging temperature was decreased down to 130°C. According to the evaluations made, about 6 kg of impurities in terms of sodium hydride was removed by the cold traps from the circuit. This was in a good agreement with the evaluated amount of hydrogen brought to the primary circuit (~90 g –into cover gas plenum, and ~110 g – into sodium). Positive reactivity was disappearing with sodium purification. By January 29, 1987, hydrogen content in the CGP decreased down to ~3.08 · 10^{-3} vol.%.

Since the number of signals from the system of fuel element cladding integrity monitoring (CIM) based on gas activity and delay neutrons detection was monotonously increasing, the reactor was shut down in March 1987 in order to detect SA with failed fuel elements and to measure sodium flow rate through the subassemblies located in the core and in-vessel storage (IVS).

Below are the results of flow rate measurements. Sodium flow rate decreased by 2-12% in some subassemblies of the third sector, while in the most of tested IVS subassemblies flow rate decrease was within 10-72% range. Considerable decrease, namely over three times as compared to the design flow rate was revealed in eight IVS subassemblies. These subassemblies were withdrawn partially (by 800 mm) using the refuelling mechanism and then installed in their cells, by this recovering their flow rate. Subassemblies were tested for integrity using special mechanism, and cladding failures in the fuel and gas sections of the fuel elements were detected respectively in three and five subassemblies. Testing subassemblies in the hot cell did not show any traces of impurities deposits (these impurities might be removed during SA washing from sodium).

In the previous years of the reactor operation, no events similar to that of January 21, 1987, occurred. Reactor hydraulics and neutronics and primary system parameters are within design limits.

3.3. Analysis of causes and countermeasures for their elimination

Analysis of parameters changes showed that carbon and hydrogen containing substances penetrated into the primary sodium and deposited on the walls of SA feet and core diagrid. This was the cause of the flow rate decrease through SA in some sections of the reactor core.

Hydrogen containing substances can penetrate into the sodium circuit during maintenance operations and cover gas replenishment.

The following versions of these substances location were considered:

- solid impurities deposited on the walls of the primary circuit cover gas plenums;

-accumulation of both solid and gaseous impurities in the stagnation zones under support ring structure etc.

In order to reveal specific causes of the anomalous event, a program of studies was developed. This program envisaged sampling and analysis of impurities in the deposits over the CGP walls and in the primary sodium, measurement of caesium isotope activity in the reactor, studies on the impurities behaviour under conditions of experimental facilities etc.

Sodium sampling using standard device did not reveal any significant deviation of impurities content in sodium from the normal value, except for one of four measurements of carbon content (~150 ppm). In the course of measurements of thermodynamic activity of carbon in sodium using method of equilibrium standard samples, increase of activity from the background value of $2 \cdot 10^{-2}$ to $5 \cdot 10^{-2}$ was detected after the event of January 21, 1987.

It was rather difficult to make inspections and deposits sampling in the reactor cover gas plenum because of its complicated configuration and small free space. It was only possible to examine gas plenum of the refueling channel. The deposits over the whole length of the channel shielding plug were revealed as well as the "beards" of 20 to 100 mm length hanging from its bottom. Analysis of these deposits samples showed that they consisted mainly of sodium carbonate (80-85%), sodium hydroxide (4-5%)

and iron oxide (up to 2%). Since the sampling was made without using any special protection means, some pollution of samples with impurities from air was probable.

In order to make measurements of caesium-137 activity, special device containing graphite pellet was installed in the loop connected in parallel to the primary sodium purification system. Just after the event of January 21, 1987, some decrease of the system indications was observed. Further, indications of the system came back to the previous level, however, some feature in caesium behaviour was noticed. While in the period preceding described events the decrease of the system indications in case of the reactor shutdown and sodium temperature reduction down to 250°C was 30-40%, later this decrease became as large as 10-12 times. After full power of the reactor had been reached, sodium temperature being as high as 530-540°C, indications of caesium activity measurement system returned to the rated level.

Besides, these events were followed by redistribution of caesium-137 related activity in the vicinity of the reactor vessel bottom.

Probability of gas accumulation under the reactor support structures was evaluated to be rather low.

The most probable version is solid impurities deposition in the reactor cover gas plenum during the previous 7 years of the reactor operation. This supposition was confirmed by:

- analysis of direct and accompanying indications of parameters and system behaviour;
- analysis of the primary sodium samples;
- results of inspection of some structural elements and surfaces of gas plenums of the primary components;
- results of the hot cell SA inspection;
- measurements of sodium flow rate in irradiated and fresh SA.

However, in order to make final confirmation of this supposition additional studies are required.

The following countermeasures are taken against penetration of hydrogen containing substances into the primary circuit:

- drying of replenishment argon has been carried out;
- devices for permanent monitoring of impurities content (and their analysis) in the reactor CGP and primary sodium are under development;
- some measures have been made in order to decrease gas leakage from the reactor CGP, thus allowing minimisation of CGP replenishment rate.

4. INCREASE OF TORQUE OF CENTRAL ROTATING COLUMN OF THE BN-600 REACTOR

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 (Fig.13). This phenomenon was interpreted as a result of sodium vapour 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

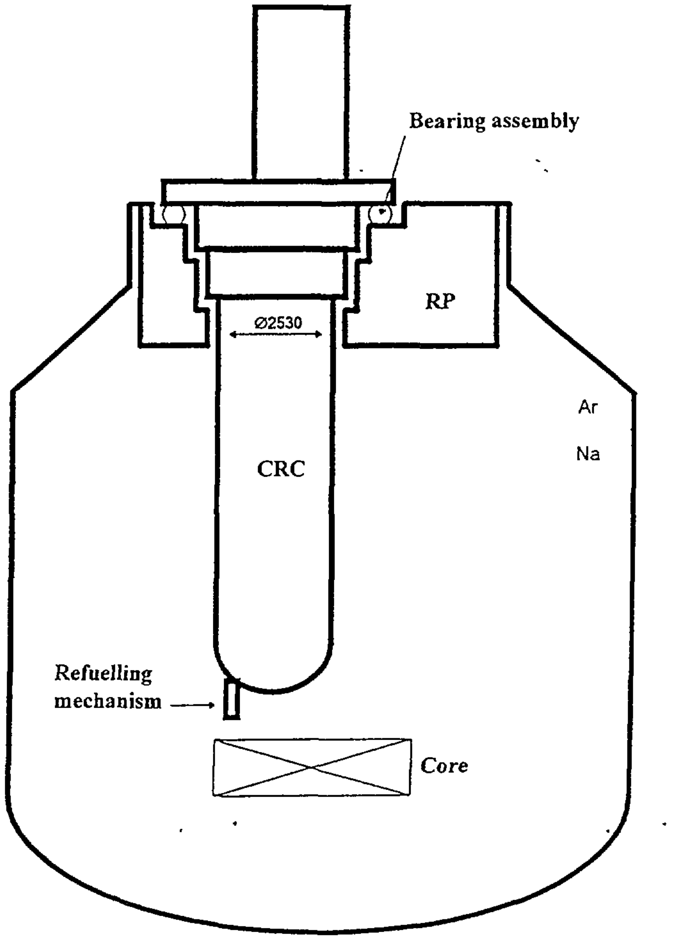


FIG. 13. Scheme of the BN-600 reactor vessel with rotating plugs.

appropriate preparations were made. This task was unique, since no procedures of such a kind had ever been implemented on this type reactors 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 (Fig. 14) 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 analysed.

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.

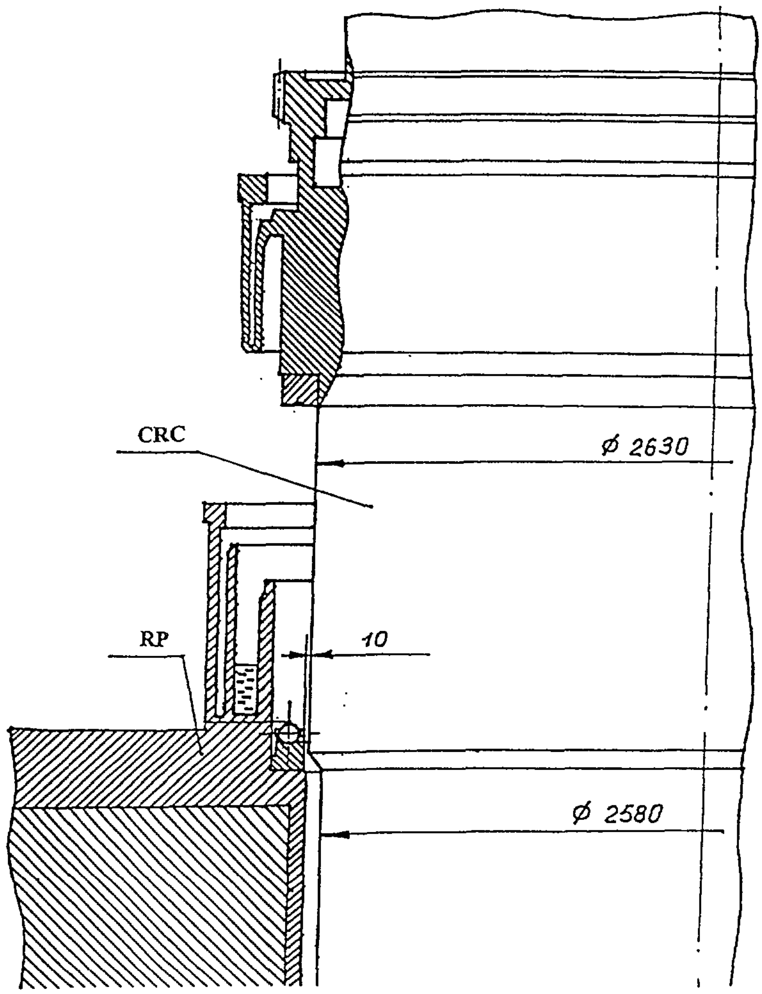


FIG. 14. Relative location of CRC and RP after CRC withdrawal.

CONCLUSION

In this paper, various anomalies and abnormal operation events occurred during BN-600 reactor operation which are the most important from the safety viewpoint, are reviewed. Successful overcoming of these events consequences and elimination of their sources, as well as reliable and stable operation of the BN-600 reactor during last years makes the ground for affirming large potential and prospects of this direction of the nuclear power.

We hope that fast reactor operating experience gained in Russia will undoubtedly be of use for the specialists in other countries, where this direction is under consideration.

UNUSUAL OCCURENCES IN FAST BREEDER TEST REACTOR

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Abstract

Fast Breeder Test Reactor (FBTR) is a 40 MWt/13.2 MWe sodium cooled mixed carbide fuelled reactor. Its main aim is to generate experience in the design, construction and operation of fast reactors including sodium systems and to serve as an irradiation facility for the development of fuel and structural materials for future fast reactors. It achieved first criticality in Oct 85 with Mark I core (70% PuC - 30% UC). Steam generator was put in service in Jan 93 and power was raised to 10.5 MWt in Dec 93. Turbine generator was synchronised to the grid in Jul 97. The indigenously developed mixed carbide fuel has achieved a burnup of 44,000 MW·d/t max at a linear heat rating of 320 W/cm max without any fuel clad failure.

The commissioning and operation of sodium systems and components have been smooth and performance of major components, viz., sodium pumps, intermediate heat exchangers and once through sodium heated steam generators (SG) have been excellent. There have been three minor incidents of Na/NaK leaks during the past 14 years, which are described in the paper. There have been no incident of a tube leak in SG. However, three incidents of water leaks from water / steam headers have been detailed.

The plant has encountered some unusual occurrences, which were critically analysed and remedial measures, in terms of system and procedural modifications, incorporated to prevent recurrence. This paper describes unusual occurrences of fuel handling incident of May 1987, main boiler feed pump seizure in Apr 1992, reactivity transients in Nov 1994 and Apr 1995, and malfunctioning of the core cover plate mechanism in Jul 1995. These incidents have resulted in long plant shutdowns. During the course of investigation, various theoretical and experimental studies were carried out for better understanding of the phenomena and several inspection techniques and tools were developed resulting in enriching the technology of sodium cooled reactors.

FBTR has 36 neutronic and process parameters initiating reactor trip and has encountered large number of trips since first criticality. The paper also highlights several modifications affected in safety related systems for improved performance and safety reviews to reduce the parameters initiating reactor trip.

The lessons learnt from the analysis of these incidents and safety reviews have been significant not only in improving FBTR performance but also as an important input for the design of future fast reactors.

1.0 INTRODUCTION

Fast Breeder Test Reactor (FBTR) is a 40 MWt/ 13.2 MWe sodium cooled, mixed carbide fuelled, loop type reactor. It has two primary and secondary sodium loops and a common steam water circuit, which supplies high pressure, high temperature superheated steam to turbine generator (TG). Heat is rejected in cooling tower (Fig 1). A 100% capacity dump condenser is provided for reactor operation even when the TG is not in service. The main aim of the reactor is to generate experience in the design, construction and operation of sodium cooled fast reactors and to serve as an irradiation facility for the development of fuels and structural material for fast reactors. It achieved first criticality in Oct 85 with Mark I core

SCHEMATIC FLOW DIAGRAM

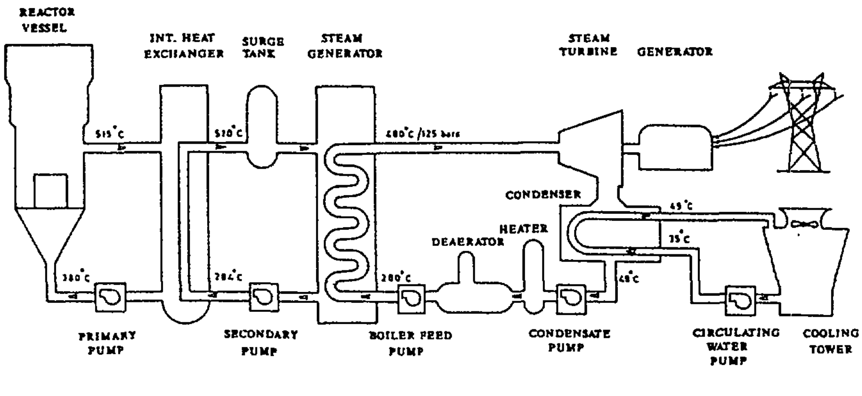


Fig.1

(70 % PuC - 30% UC). The steam generator was put in service in Jan 93, power raised to 10.5 MWt in Dec 93 and 12 MWt in Jul 97 when TG was synchronised to the grid. The reactor is presently in its 6th irradiation campaign. The reactor has operated for more than 20,000 h so far with 6600 h operation at high power. Fig 2 gives the operation histogram since Jan 93 and fig 3 gives the present core configuration.

2.0 PERFORMANCE OF THE FUEL

Indigenously developed Pu rich mixed carbide fuel is chosen as driver fuel for Mark I core[1]. Since the fuel is new, it is proposed to ascertain its performance through Post Irradiation Examination (PIE) and increase the reactor power in a phased manner. Detailed PIE has been carried out in inerted shielded cells on one irradiated fuel subassembly (SA) at a burnup of 25,000 MW·d/t at a peak linear heat rating (LHR) of 320 W/cm (Fig 4). PIE included visual examination, dimensional measurement, leak testing, eddy current testing, X-radiography and metallography. The observations were; shining appearance of the fuel pins,

HISTOGRAM OF REACTOR OPERATION

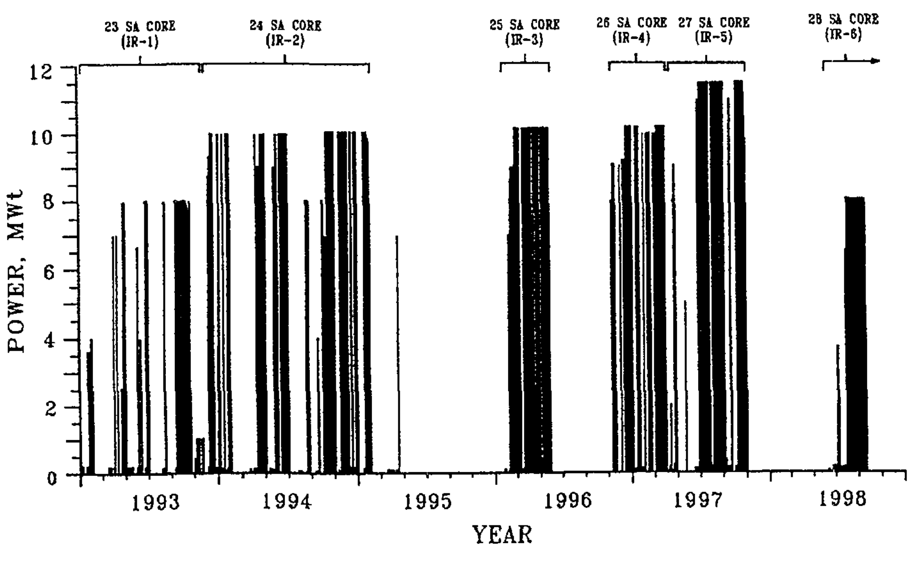
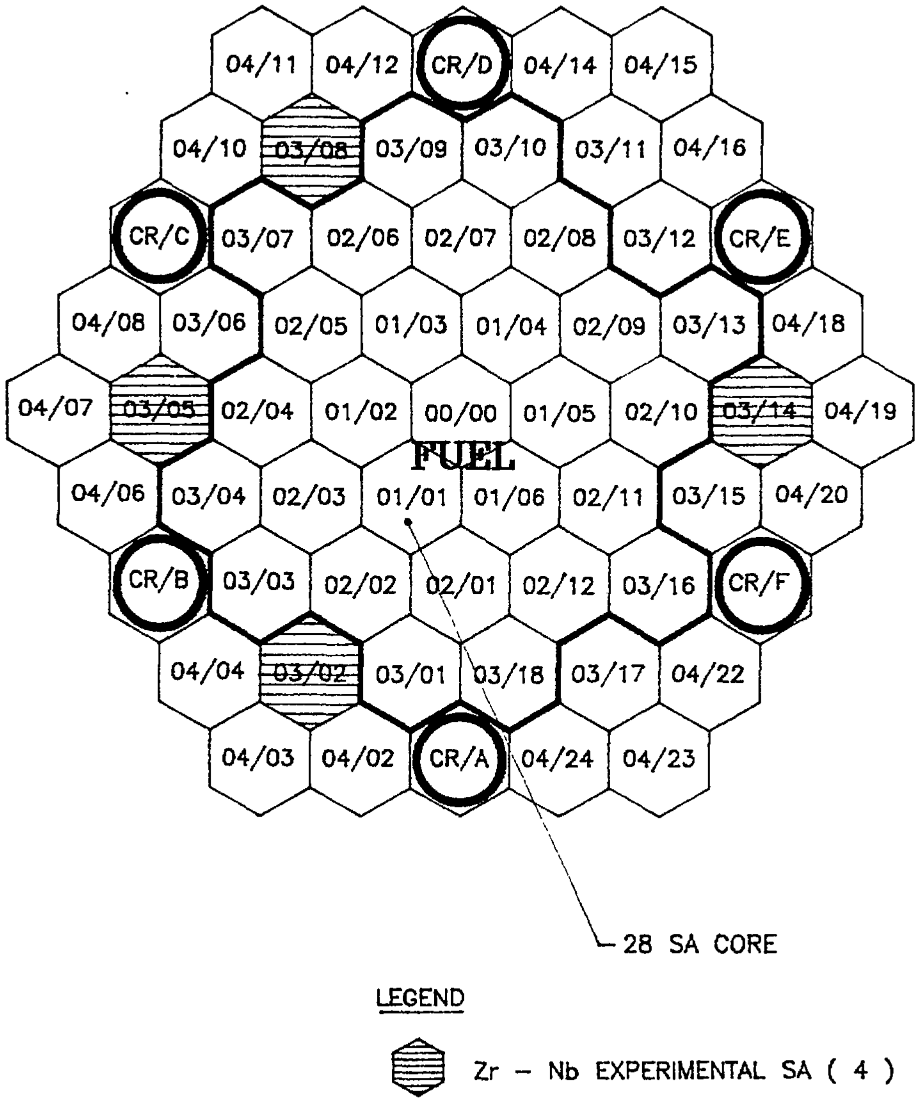


Fig.2



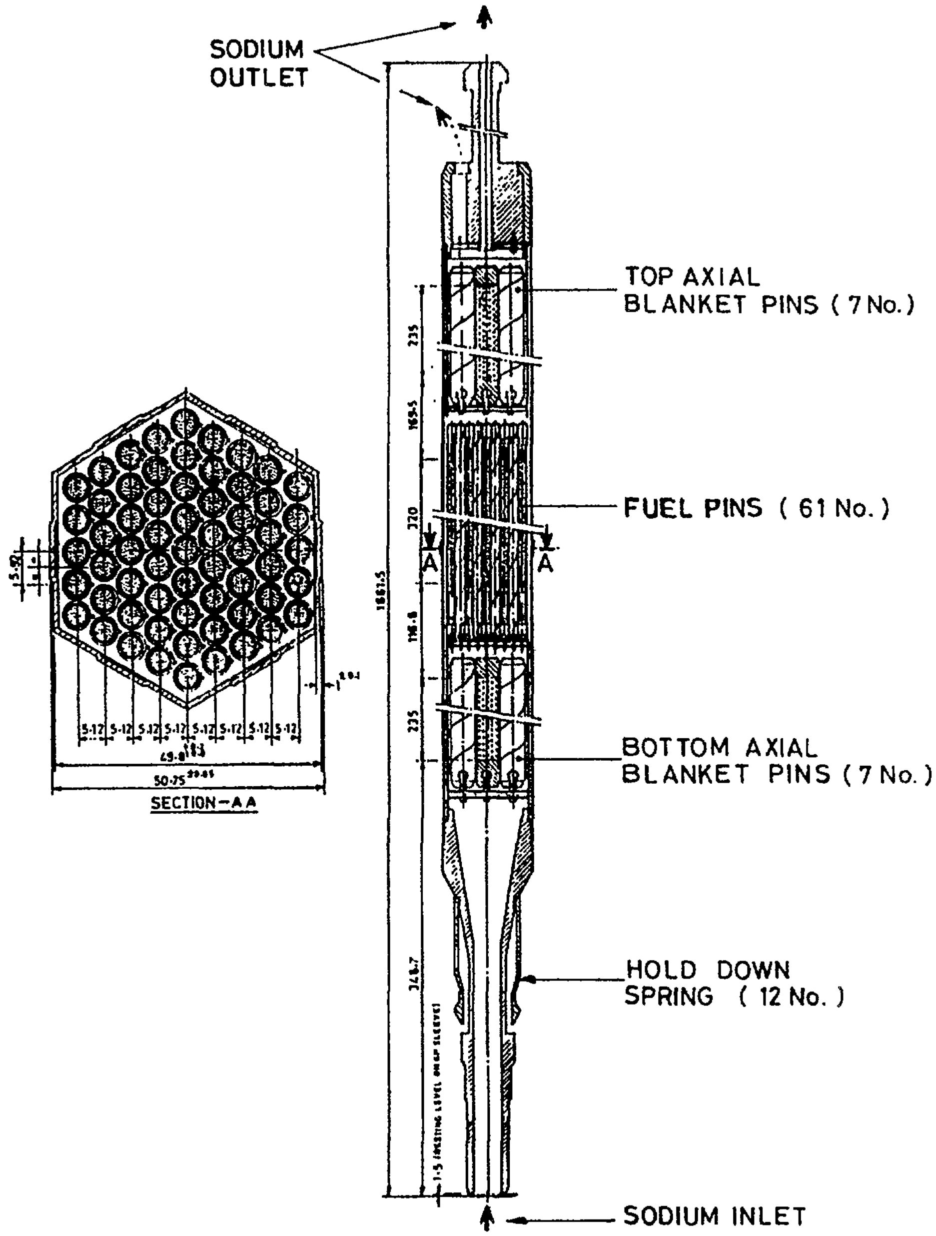
CORE CONFIGURATION

Fig.3

maintenance of clad integrity, non-closure of fuel clad gap and fuel swelling rate being less than predicted. From these observations, fuel performance has been inferred to be excellent[(2)(3)]. The fuel has since achieved a maximum burnup of 44,000 MW·d/t at LHR of 320 W/cm without any fuel clad failure. Clearance has also been obtained to enhance the LHR and fuel burnup to 400 W/cm and 50,000 MW·d/t respectively. This is proposed to be achieved shortly.

3.0 PERFORMANCE OF SODIUM SYSTEM

Primary and Secondary sodium systems are in service for the past 14 years at a maximum temperature of 485°C at the outlet of core and 420°C in sodium circuits and the performance of the sodium circuit components has been satisfactory.[(4)(5)(6)] The sodium purity has been well maintained and there has been no incident of any radioactive sodium leak from the primary circuit. Once-through steam generator (SG) has been in service for about 6600 h and there is no incident of steam generator tube leak.



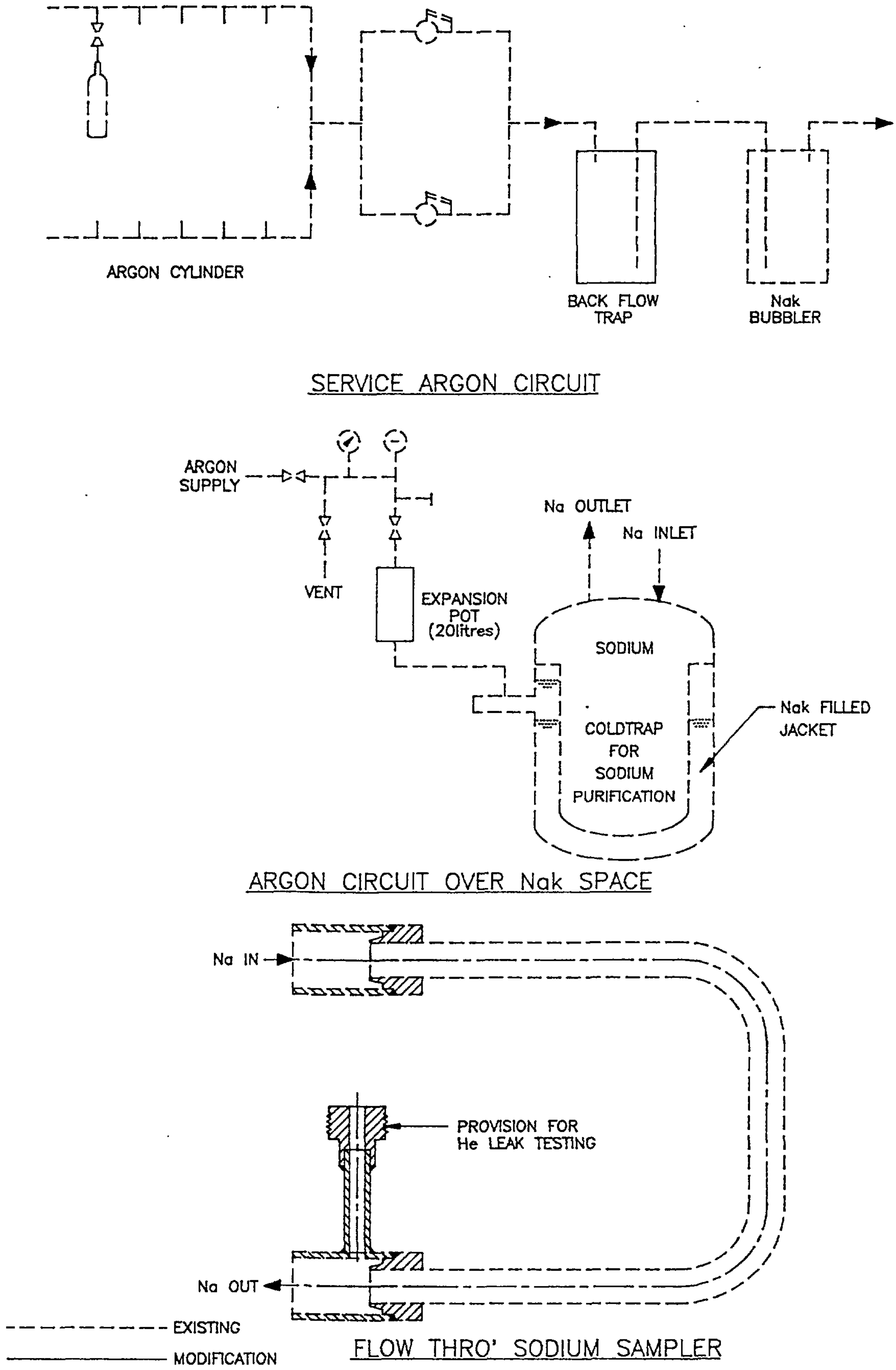
FUEL SUBASSEMBLY

Fig.4

Three minor incidents of Na/NaK leak in the secondary sodium circuit and three minor incidents of water leak from the water / steam subheaders of SG are described here.

3.1 Sodium/NaK leak incidents (Fig 5)

3.1.1 While preheating secondary cold trap during initial commissioning (Oct 84), about 2.5 l of NaK leaked out from the NaK jacket through spark plug type high level probe. Investigation revealed the failure of the level probe due to high pressure during preheating because of nonavailability of adequate expansion space in the jacket. The fire was put out effectively in 15 min by dry chemical powder (DCP). Modifications were done to prevent recurrence by capping of level probes and providing an argon pot of 20 l capacity to allow free



SODIUM/NaK LEAK INCIDENTS

Fig.5

expansion of NaK during preheating phase. Surface thermocouples were also provided to follow NaK temperature during pre-heating.

3.1.2 Secondary sodium sample is required to be taken twice a year for chemical analysis. About one litre of sodium leaked out (Sept 87) through a swagelok coupling while putting in service the secondary sodium flow through nickel sampler. The fire was fought using DCP and CO_2 extinguishers and was put off in 30 min. To prevent recurrence, provision was made for helium leak testing of swagelok coupling joints after installation of the sampler. Over flow type sampler with conoseal joints for better leaktightness and more representative sampling was also installed in one of the secondary loops.

3.1.3 During adjustment of pressure setting of the regulating valve in supply argon system (May 88) about 2 l of NaK backed up and leaked out from the NaK bubblers provided for supply argon purification. The NaK leak was carefully collected in a tray covered with DCP and safely disposed off within 30 min. As a remedial measure, a backflow trap of 50 l capacity was introduced on the upstream side of bubbler and a pressure equalising valve was provided across it.

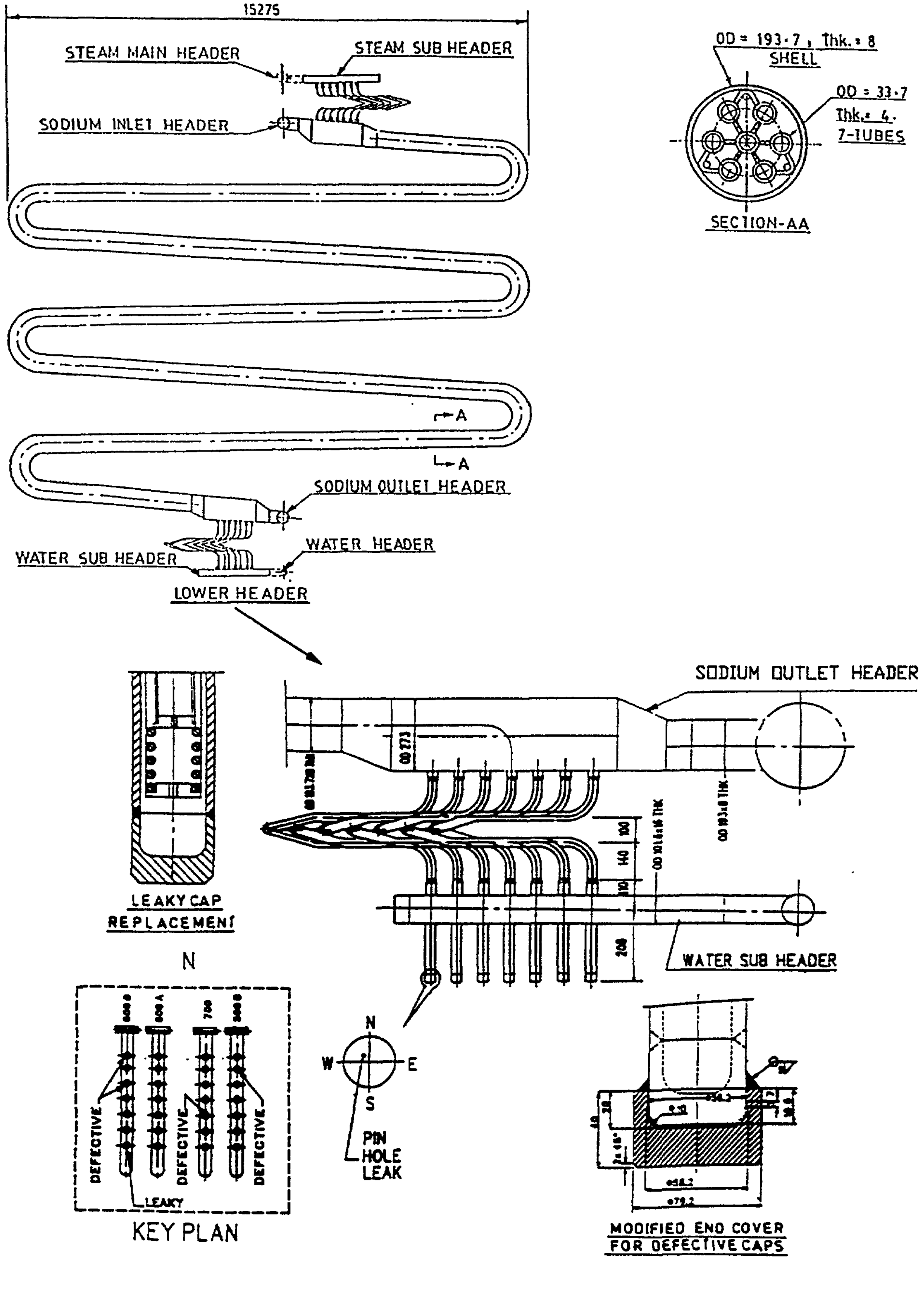
3.2 Water leak incidents in SG subheaders

The SG is a counter current once through type and of modular construction in which sodium flows on the shell side and water / steam flows in the tubes (Fig 6). There are two such modules in each of the secondary sodium loops and the 4 modules are housed in an insulated SG casing.

3.2.1 In Jan 93 when SG was put in service for the first time, after 70 h of operation at 4 MWt, a water leak took place due to a linear pin hole defect in the end cap of one of the orifice assemblies at SG inlet. All similar caps (35 numbers) were ultrasonically inspected and four more were found having indication of linear defects. The leaking cap was replaced and additional covers were welded on the defective caps and SG modules requalified (Fig 6). This was attributed to inspection (having less capability to detect such defects) before accepting the material for fabrication of these caps.

3.2.2 Water subheaders of the 4 modules of the SG are provided with flanged orifices located inside the SG casing for flow measurement to study SG stability. In Aug 93, when reactor was operating at 8 MWt, feed water was found leaking through the orifice flanges. Investigation revealed that the leaktight orifice flanges under ambient conditions tend to develop leak under operating conditions as a result of differential thermal expansion between the water subheaders and the SG modules. All the orifice flanges were replaced with welded spools with integral orifices.

3.2.3 In Feb 98, when reactor was in shutdown state, while readjusting the settings of SG safety valves, water leak was observed in one of the bosses in experimental thermowell in the steam subheader of one of the SG modules. These thermowells are also provided for SG stability studies. The leaking thermowell boss and plug were replaced with a dummy piece of similar dimensions to have the same flow restriction in the path. Liquid penetrant inspection (LPI) was carried out on all similar welds in the four SG steam subheaders and no defective indications were noticed. Investigations revealed that this was due to lack of heat treatment of this part during fabrication.



STEAM GENERATOR MODULE

Fig.6

3.3 Major sodium circuit components replaced during 14 years of operation include; lower parts of two control rod drive mechanism due to metallic bellows failure, central canal plug due to failure of two thermocouples for measurement of outlet temperature of central fuel SA, one cold trap in secondary sodium loop due to impurity loading during SG commissioning, two reheaters of improved design in steam generator leak detection circuit and two secondary sodium service bellows sealed valves due to bellows failure. Three sodium impurity monitors viz.; electro-chemical hydrogen meters, electrochemical carbon meter and cover gas hydrogen meters developed in the centre have been added in the secondary sodium system.

4.0 UNUSUAL OCCURRENCES

This section describes the four major unusual occurrences which had safety implications and resulted in very long plant shutdowns viz.; fuel handling incident, main boiler feed pump seizure, reactivity transients and malfunctioning of core cover plate mechanism.

4.1 Fuel Handling Incident[4]

Fuel handling is carried out in shutdown state, with sodium at 180°C, with the help of two charging/ discharging machines and two rotating plugs. Since a core SA has to be handled 6 m below 0 elevation, 6 m long, guide tube of 113 /67 mm diameter is introduced into the fuel handling canal to guide the fuel handling gripper and to prevent lifting of adjacent SA.

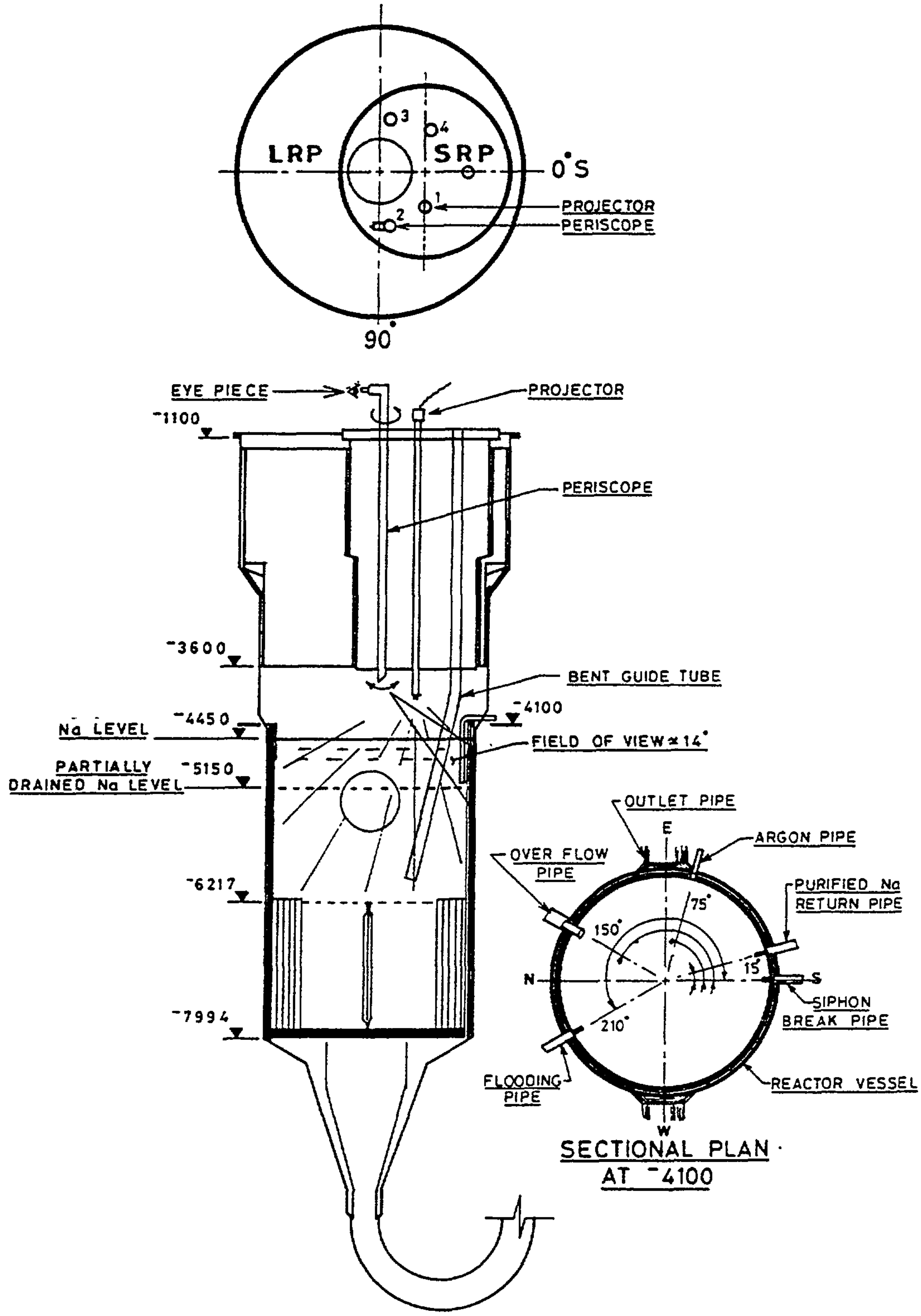
4.1.1 Incident Description

During an inpile transfer operation in May 1987 when a fuel SA was being transferred to the periphery from the core, difficulties were experienced in releasing the SA in its new location. Manoeuvres were done to install the SA at various locations at the periphery, but to no avail. Finally it was decided to discharge the SA, but the gripper mechanism was getting stuck midway in the guide tube. The SA was forcibly extracted through the guide tube. Examinations revealed bend in the head and foot and bow in the body of the SA but no fuel pin failure. The fuel handling machine gripper was also found bent. When the guide tube was being removed by normal procedure, it was not coming out. Attempts to remove the guide tube along with its outersheath (which is fixed to the fuel handling canal of the reactor) also proved futile. It was then obvious that the guide tube had got bent beyond the limit, which will allow its removal through the canal. During the various manoeuvres to overcome the problem, a complex mechanical interaction seemed to have taken place with the components within the reactor vessel causing mechanical deformation to the fuel handling gripper, the fuel SA and the guide tube.

At this juncture, all further operation on pile was suspended. A quantitative measure of the bend in the guide tube and extent of deformation to various other components in the reactor vessel became necessary.

4.1.2 Investigations

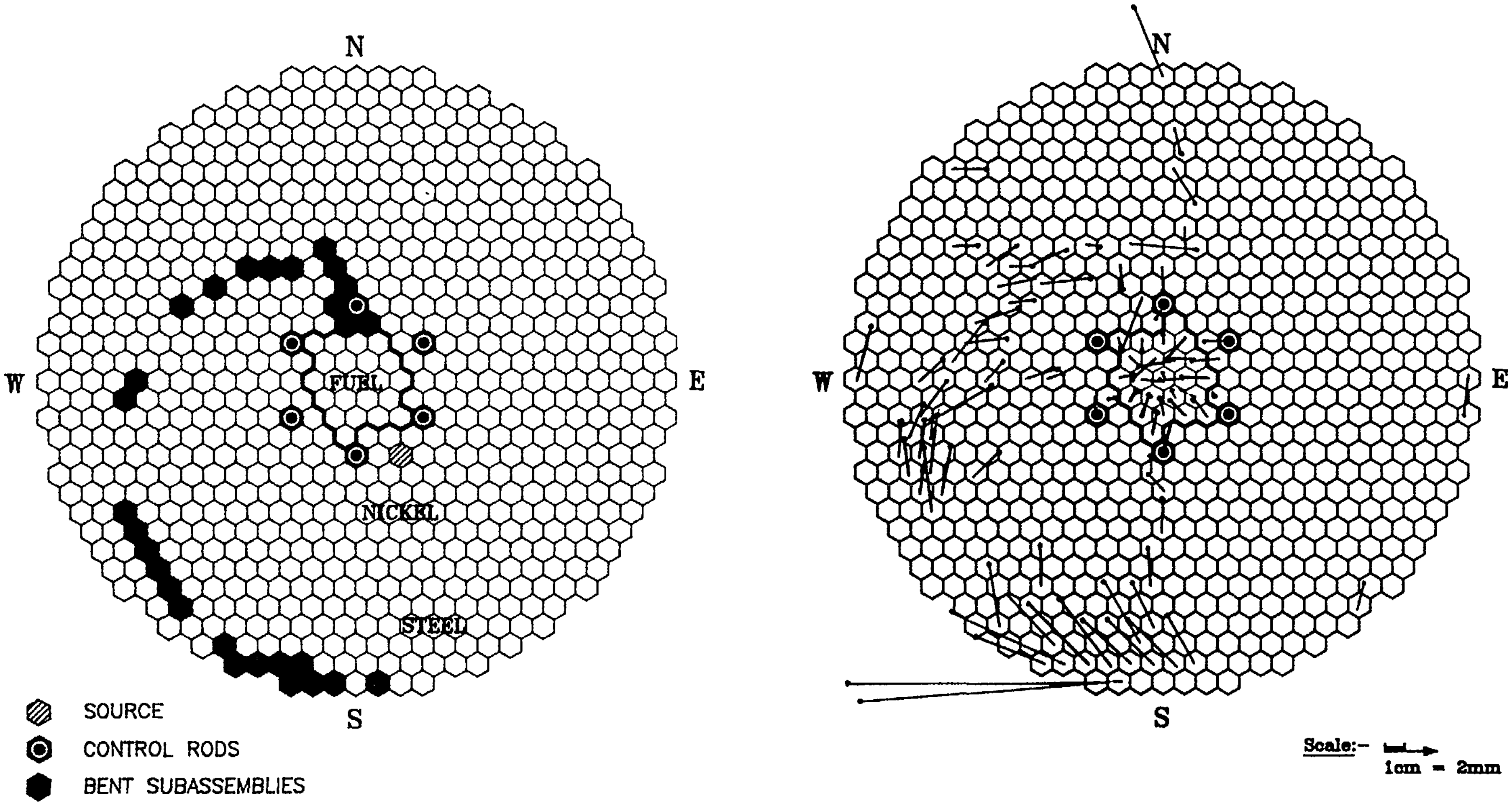
Three techniques were developed and utilised to assess the bend of the guide tube, viz., optical inspection, ultrasonic air gauging and mechanical disc gauging. Optical inspection was carried out with a periscope/projector system (Fig 7). Sodium was drained to expose the heads of SA and the bend of guide tube was measured by finding out the radius of sweep of its



REACTOR VESSEL INTERNALS INSPECTION THROUGH PERISCOPE

Fig.7

bottom tip with reference to heads of SA when it is rotated on its own axis. This inspection also indicated a lifted SA at the periphery where manoeuvres to lower the fuel SA were earlier made. Slight deformations of the heads of some of the reflector subassemblies along two spiral paths were also seen (Fig 8). Ultrasonic air gauging method involved lowering of an



OPTICAL INSPECTION

RESULTS OF CCMD INSPECTION

DAMAGED SUBASSEMBLIES DURING FUEL HANDLING INCIDENT

Fig.8

ultrasonic probe in a specially designed carrier, directing its beam towards the guide tube and measuring the time of flight. The profiling was done under perfect leaktight conditions with sodium partially drained to expose the complete length of the guide tube. Mechanical disc gauging involved lowering of discs of different diameters inside the guide tube and measuring the depth at which each disc stops entering further. Since the bend was much larger than the inner diameter of the guide tube, gauging could be done only upto a certain depth and the bend value had to be estimated by extrapolation. All these techniques were successful in assessing the bend with a high degree of reliability and it became obvious that the guide tube should be cut in-situ for its removal from the reactor. The profile also indicated that the cutting had to be done at a depth of around 3 m below zero level.

In the light of the sighting of a lifted SA during visual inspection, to authorise rotation of plugs, it was important to rule out any lifted SA below the core cover plate housing thermocouples for measuring the outlet temperatures of 85 SA. For this inspection, an under sodium ultrasonic scanner was developed and it was confirmed that there was no protruding SA below the core cover plate.

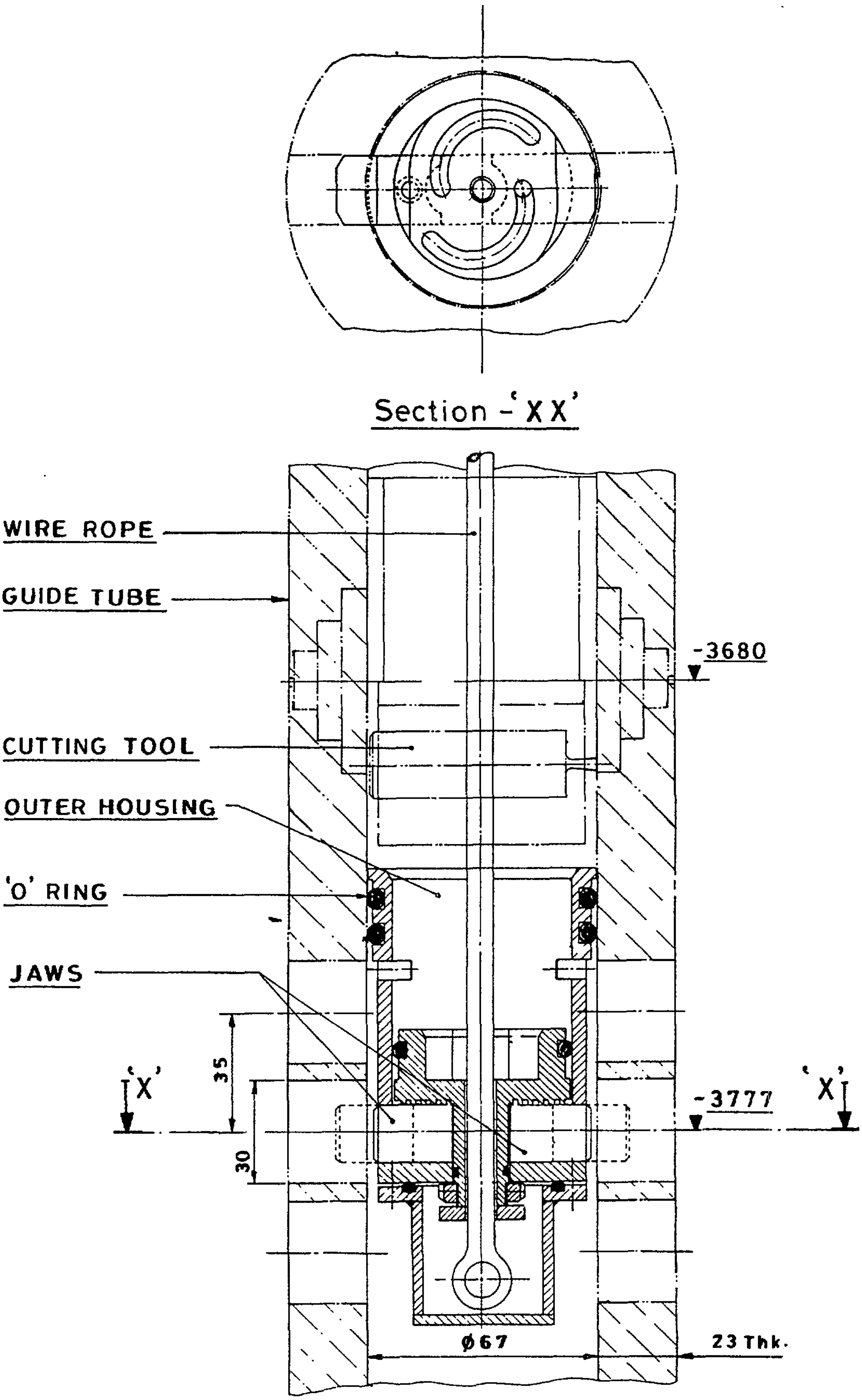
4.1.3 Retrieval of damaged components

The design of leaktight remote cutting tool had the following special features; The cutting to be done at a depth of 3 m, the tool to be accommodated within the guide tube bore of 67 mm and provide a depth of cut of 23 mm without tool chatter, the bottom portion of the cut guide tube to be held in position till the top part is removed, no cutting chips should fall into the reactor and leak tightness should be maintained during the cutting operation. An elegant remote cutting tool having these features was developed and it consisted of assembly of telescopic tubes for transmitting the rotational movement to the tool and controlling and monitoring the axial travel and depth of cut from the top. No lubricant was used and special features were provided to recover all the chips generated. This tool also employed a leaktight plug anchored to the bottom part of the guide tube to maintain leaktightness w.r.t. reactor cover gas, to collect the chips generated and to hold the bottom part during complete process of cutting and retrieval (Fig 9). The cutting sequence was carefully chosen to permit radial entry of the tool holder along the guide tube thickness during the last stages of cutting, thus avoiding tool chatter. After successful mockups, the tool was perfected and with microprecision the in-situ cutting and retrieval of the damaged guide tube was successfully completed in May 88. Measurement of the profile of the cut guide tube indicated a deformation of 350 mm, bearing full testimony to the reliability of the various remote methods employed to measure the bend.

The retrieval of the damaged SA was another work successfully executed on the pile. Since the core was relatively new, radioactivity levels were very low. Hence all the SA were directly viewed and mapped through a transparent plate fixed on top of the fuel handling canal and 18 numbers of steel and nickel reflectors SA were identified for replacement. The retrieval operation was done using a two-finger gripper mechanism especially engineered for this purpose. Subsequently core coordinate measuring device (Tube de visee), received from France, was utilised to inspect 88 SA in the vicinity of the damaged path and 10 more reflector SA having minor bends in their heads were also replaced (Fig 8).

4.1.4 Incident Analysis and Remedial Measures

The sequence of events leading to the incident was reconstructed from all available evidences. The incident was found to have originated in excessive friction in the fuel handling



SCHEMATIC OF THE CUTTING TOOL

Fig.9

gripper assembly, either due to O-rings, or sodium aerosols or both, resulting in the inability to open the fingers and release the SA. During the process of rotation of the plugs, the SA seems to have slipped down slightly in the transfer flask resulting in damages to its own foot and to the heads of reflector SA in its paths. During the manoeuvre to lower it in the periphery, the bent foot ejected out an adjacent reflector SA. A complex mechanical interaction took place between the ejected SA and the guide tube during subsequent rotation of the rotating plugs, resulting in damage to the guide tube.

Based on these findings, appropriate remedial measures, including mechanical stopper for the fuel handling gripper and redundant interlocks for authorising plug rotation, were implemented. Proper maintenance and operating procedures for the fuel handling mechanisms were evolved. It took two years to recover from this incident and the reactor was restarted in May 89. The remedial measures were so effective that about 300 fuel handling operations carried out for the past 9 years involving charging, discharging and inpile transfer have been smooth and trouble free.

4.2 Main boiler feed pump (MBFP) seizure[7]

Steam water circuit (Fig 10) consists of condensate extraction pump (CEP) taking suction from main condenser (MC) and dump condenser (DC) through low pressure flash tank (LPFT), two contact type low pressure (LP) heaters, a deaerator working at 13 Kg/cm^2 pressure to provide 190°C feed water and MBFP to supply water to SG at 125 Kg/cm^2. MBFP is a 10 stage barrel type pump designed for a feed water temperature of 196°C, delivering 89 m^3/h flow at a head of 1770 mlc with operating speed of 5700 rpm and input horse power of 580 kW. The required NPSH is 6.9 mlc (actual test value being 5 mlc). As per design, the balancing leak off from the pump discharge is fed to the pump suction.

4.2.1 Incident description

In Apr 92, this pump was being used at a flow of around 17 m^3/h for preheating feed water by its own power for putting SG in service. At 165°C feed water temperature, abnormal noise was heard from the pump with large fluctuations (110 to 170 kg/cm^2) in its discharge pressure gauge. The motor current crossed full scale of 75 A and there was reduction in feed water flow. The pump was immediately stopped and on inspection motor drive end thrust pads and most of the stages of impellers, were found damaged. Water lubricated hydrostatic bearing sleeve was found seized. Balancing piston and sleeve had scoring marks.

4.2.2 Investigation

With the site location of deaerator from which the pump takes suction, the available NPSH varies from 8.69 to 8.52 mlc, which under normal operating condition for full flow is adequate. During investigation, it was found that prior to the incident condensate system had to be shutdown to attend to some instrumentation problem. MBFP was kept in recirculation mode and system preheating was continued. Restoration of condensate system later on resulted in admission of cold condensate to deaerator causing collapse in deaerator pressure and hence reduction in available NPSH resulting in severe cavitation and flashing at the pump inlet and causing damage.

4.2.3 Remedial Measures (Fig 10)

Modifications to improve available NPSH at pump suction were carried out viz.; balancing leak off line which was earlier heating the suction was routed to the deaerator, continuous cold injection was ensured at pump suction to improve transient performance, additional recirculation line was added to avoid pump operation at low flows. Operating procedures were modified and feed water heating by package boiler steam was strictly adhered to. This resulted in a delay of 8 months to put the SG in service.

One of the two MBFBs has since been replaced with indigenous make. This pump operates at 3600 rpm with same Q- H characteristics, without any hydrostatic bearing and having lower NPSH requirements (4.3 mlc).

4.3 Reactivity Transients

4.3.1 Incident Description (Fig 11)

In Nov 94, when the reactor was operating at 10.1 MWt, power was found to be slowly increasing. Though the control rods were lowered one by one, the power continued to rise and reached 10.4 MWt in about a minute. The control rods were cumulatively lowered by 7.6 mm and the power was brought down to 10.1 MWt. The reactivity meter registered a spike of 3 pcm during the incident. Reactivity before and after the incident did not reveal any measurable permanent reactivity gain.

In Apr 95, during a startup, when the reactor was at 7.1 MWt, power increased sharply by 450 kWt in 7 s. No control rod movement was made at this time. Reactor underwent a scram on high positive reactivity and the recorder indicated a spike of + 10 pcm. Criticality measurements before and immediately after the incident revealed a reactivity gain of about 24 pcm. However, a reduction of this value to 14 pcm was observed upon subsequent measurements. The measurement error itself is of the order of ± 10 pcm.

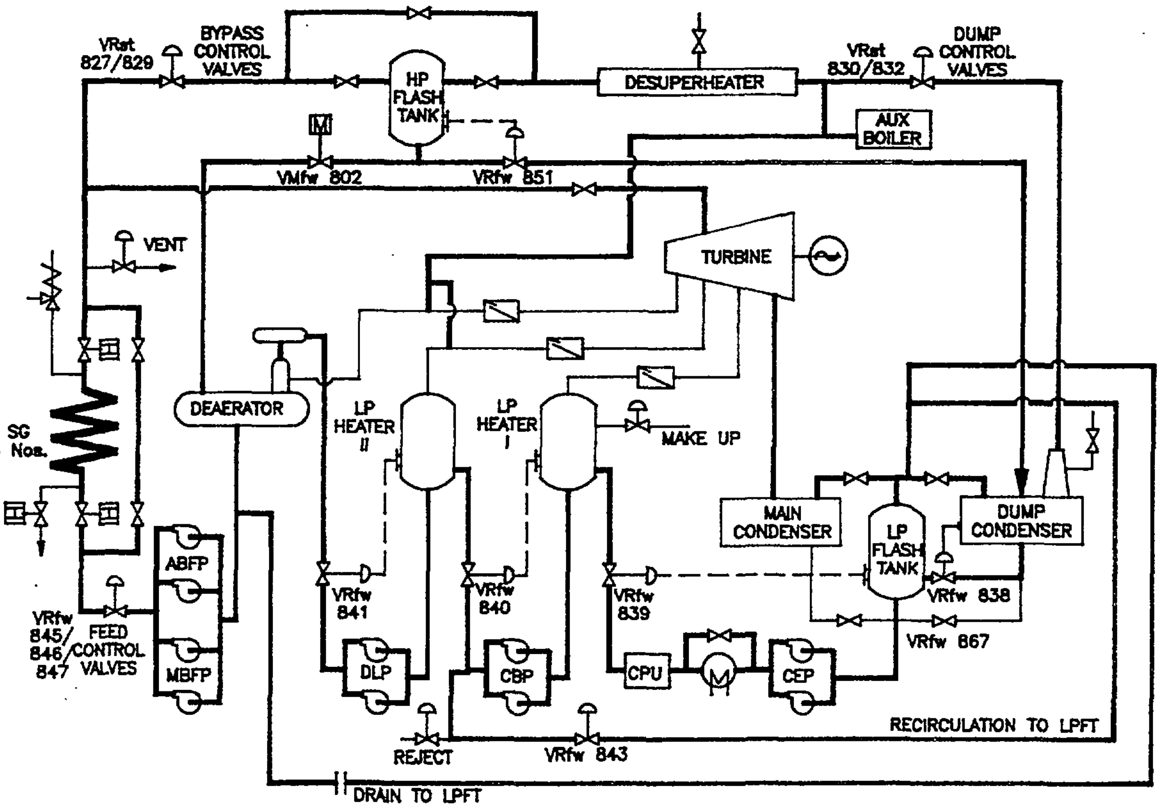
4.3.2 Investigations

To identify the probable cause of the transients, 20 postulates were studied and 14 of them were tested during reactor operation. Based on recorded observations, calculations, tests and analysis, inadvertent raising of CR by operator just before the incident was ruled out. The postulates studied can be broadly classified into the following five categories: viz. process parameter changes, absorber movement, voids collapse and sodium filling, fuel movement and moderator ingress[8].

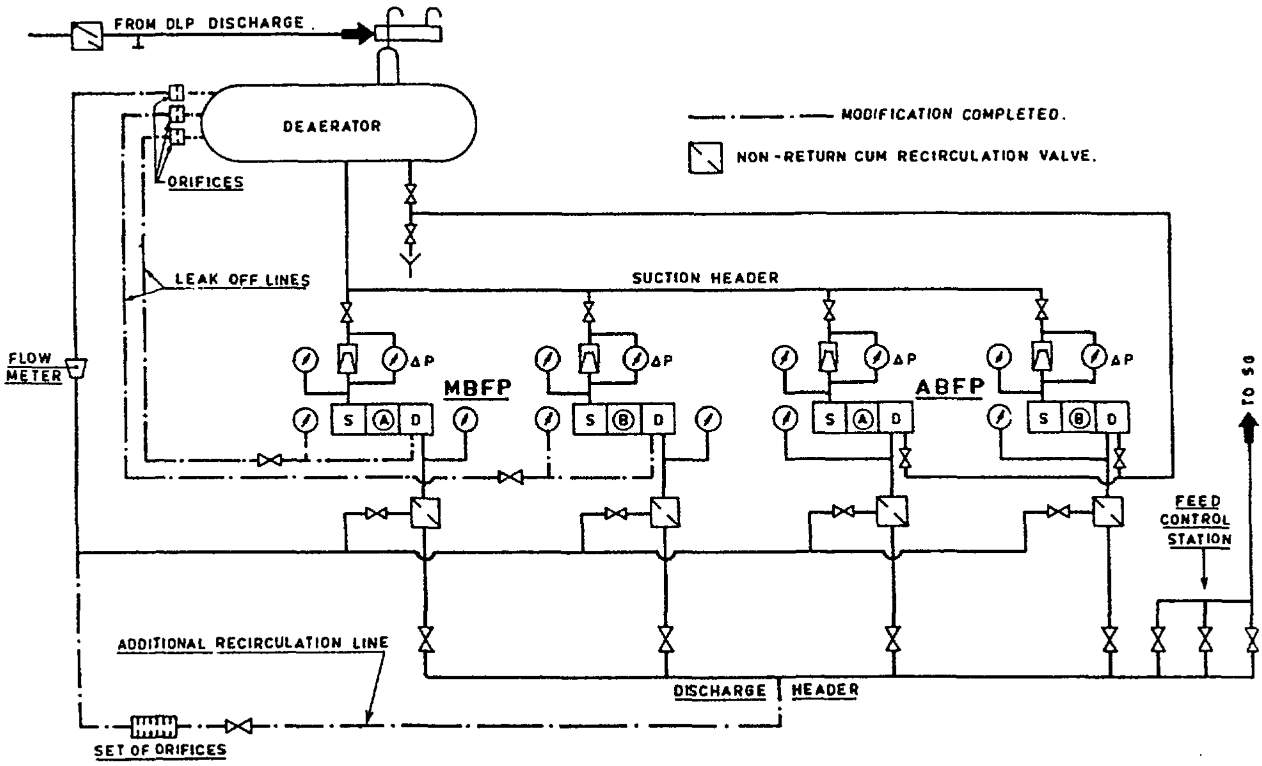
4.3.2.1 Process parameter changes

The changes in process parameters resulting in decrease in core inlet temperature causing reactivity transient were studied. The required inlet temperature change to cause the two transients were estimated to be -3.1 and -4.9°C respectively. Extensive tests on the influence of changes in primary, secondary and feed water flow and steam pressure were studied at 9.5 MWt power and the results are given in Table 1.

All these experiments have shown that these events are reversible as well as recordable by various chart recorders provided in the control room. As no change in any process

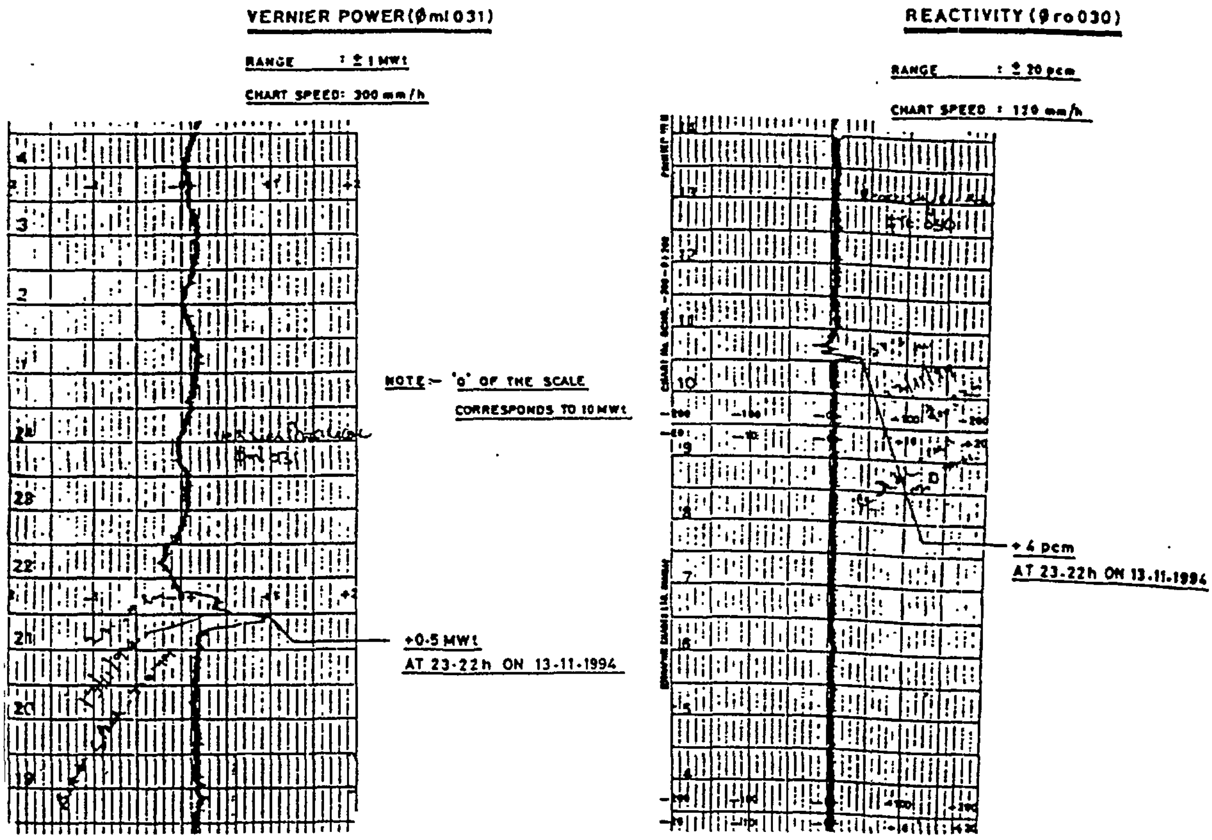


STEAM-WATER SYSTEM FLOW SHEET

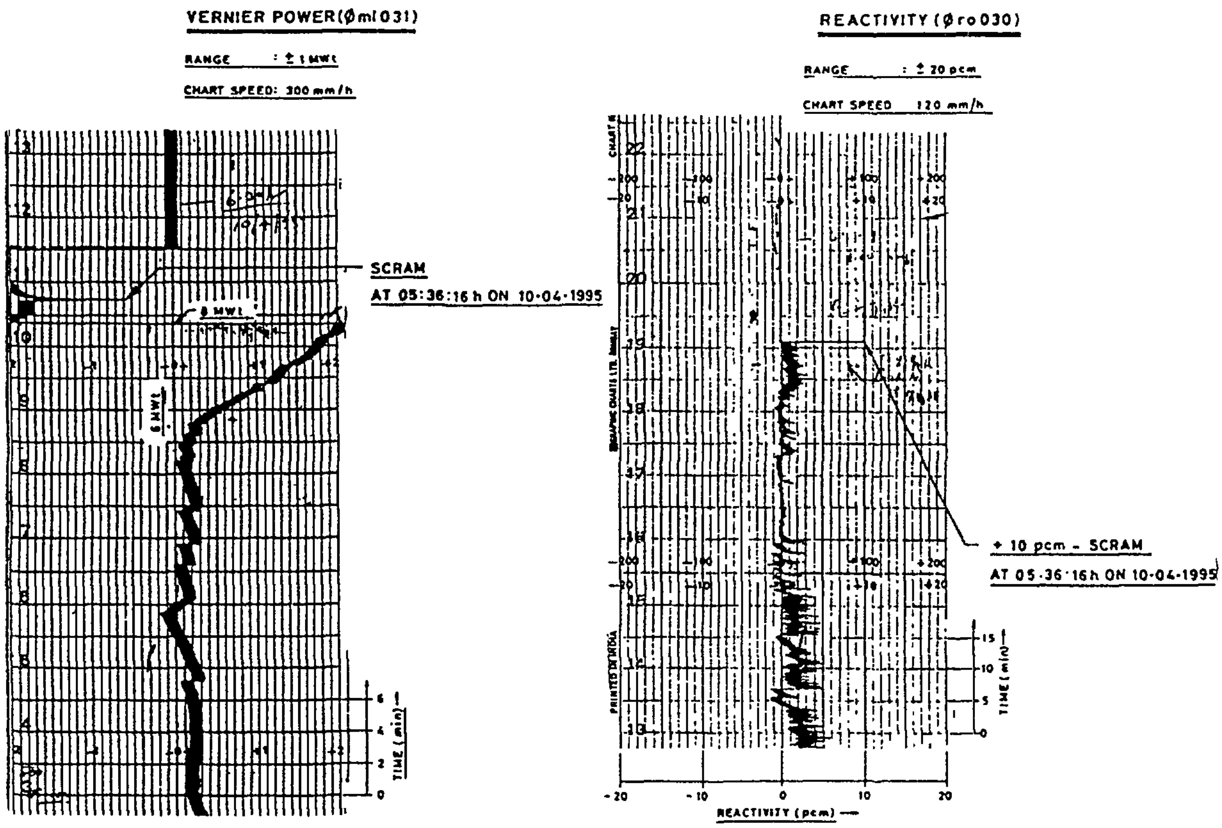


STEAM WATER SYSTEM MODIFICATION

Fig.10



REACTIVITY TRANSIENT OF 13.11.94



REACTIVITY TRANSIENT OF 10.04.95

Fig.11

parameter was observed during the course of transients, this being the cause for the incidents was ruled out.

Table 1 Process Parameter Changes

Event	Observations
Primary sodium flow change	4% increase results in power increase of 100 kWt
Secondary sodium flow change	9.5% increase results in power increase by 350 kWt
Feed water flow change	13.4% increase results in increase of power by 200 kWt after a time delay of 200 s.
Steam Pressure change	Reduction of steam pressure from 114.7 to 97.4 Kg/cm^2 results in increase of power by 500 kWt after a delay of 60 s.

4.3.2.2 Absorber movement

Inadvertent movement of absorber away from core can result in reactivity increase. Possibility of stuck contacts resulting in continuous raising of control rod, improper gripping of control rod by control rod drive mechanism resulting in its relative movement and movement of boron carbide pellets inside the control rod, were also studied and ruled out. Also the remeasurement of control rod reactivity worth done in June 95 compared well with the earlier values. Loss of antimony, a neutron absorber from auxiliary neutron source can give a gain of 14 pcm. This requires breach of double containment to get into sodium and should also affect the shutdown counts, which was not observed. Hence absorber movement being the cause was ruled out.

4.3.2.3 Voids collapse and sodium filling

Three scenarios were studied, viz., sodium voiding subassembly due to boiling and sudden collapse of these voids, sudden release of accumulated argon gas from the core and sudden release of accumulated helium from control rods. Filling of the void space by sodium can result in reactivity gain. For such an incident 16% of the volume in a SA is required to boil and this should have resulted in scram by the plugging detection subroutine. It was also estimated that release of 41 cc of gas and its displacement by sodium, can explain the transient. Tests were carried out to vary primary sodium free level in IHXs upto ± 100 mm for argon entrainment but no perceptible reactivity change could be observed. Detailed analysis revealed that sodium voiding by helium generation in control rods and their subsequent displacement by sodium has to take place simultaneously in six control rods to cause the transient which is highly improbable. Studies of all the three scenarios indicated that these could not have caused the transients.

4.3.2.4 Fuel movement

Any fuel restructuring leading to axial contraction could have positive gain in reactivity. The fuel had seen a maximum / average burnup of about 13,000 /10,000 MW·d/t at the time of the transients in Apr 95. Central SA discharged in July 96 after a burnup of 25,000 MW·d/t indicates no axial contraction. It is also observed that fuel swelling rates are less than predicted which however is an irreversible phenomena and the reactivity changes will be permanent. As the incident took place after loading 25th fuel SA, its worth was remeasured in Aug 95 and it compares well with the earlier values. Geometric changes in the core due to weight of core cover plate mechanism while it is resting on the top of the core for accurate measurement of fuel SA outlet temperature and sudden reversal during normal operation causing the transient, was also postulated. Experiment was done at low / high power and no perceptible changes in reactivity could be observed. Hence fuel movement being the cause was ruled out.

4.3.2.5 Moderator ingress

Ingress of moderator in the core causing such a transient was considered most likely. Three scenarios were considered viz. oil ingress from mechanical seals of primary sodium pumps, hydrogen/hydrate ingress through cold trap or through cover gas and ingress of Be from the auxiliary neutron source. A special micro filter SA was loaded in the core and after circulating 520 times the primary sodium inventory through this SA to trap sodium oil reaction products, the filterate was analysed for carbon content and found to be very small (30 mg). It was considered inadequate to cause the transient (30 g). Any ingress of hydrate from cover gas to cause the transient of this nature would have increased the plugging temperature to 290°C. Plugging temperature was found to be well maintained at 105°C during both the transients. An experiment was also carried out to observe the washing of hydrate from cold trap by increasing the cold point temperature from 120 to 130°C for 12 h. No reactivity changes could be observed. Efficacy of NaK bubbler for purifying argon and helium was also checked and found OK. Visual inspection of reactor vessel surfaces and bottom of rotating plugs did not indicate any buildup of hydrate deposits on the surfaces which could get loosened and fall in the core. Ingress of Be was ruled out as there was no change in the shutdown count. Hence moderator ingress may not be the cause for the transients.

4.3.2.6 The influence of any parameter causing reactivity changes being more perceptible at low power, the reactor was kept in subcritical state at 180 and 400°C for about two weeks for observation but no reactivity transient was observed.

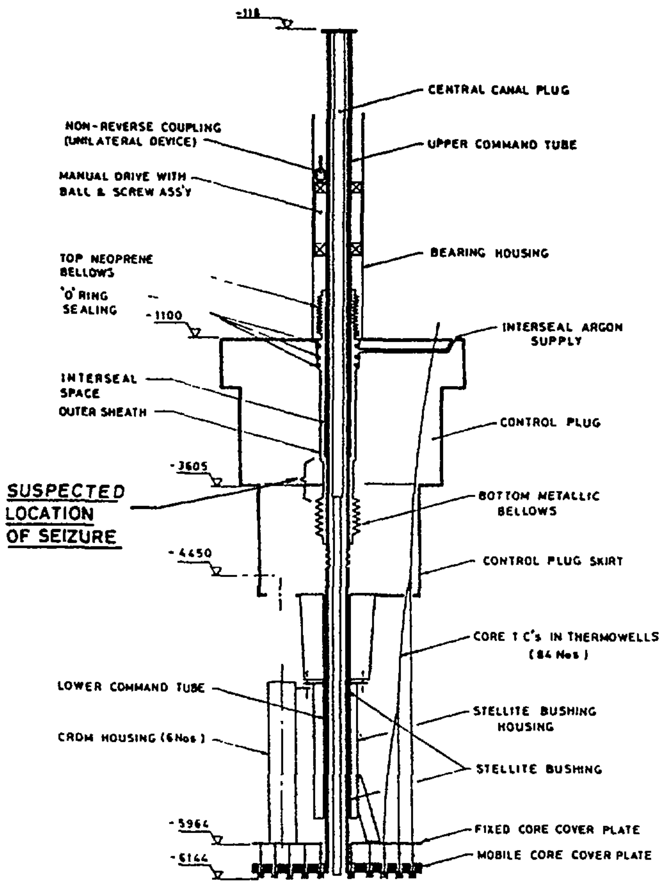
4.3.2.7 Present Status

Results of detailed investigations carried out did not reveal the cause of these two transients. Due to hydrodynamic coupling of the two primary sodium loops, investigations regarding possibility of introduction of cold slug of Na into the core are planned. Since the transients have occurred during high power operation and reactivity inputs were very small causing no undue safety concern, permission was sought to continue reactor operation with enhanced scram threshold for reactivity (± 30 pcm) and augmented data acquisition system to acquire sufficient data for analysis in case the incident recurs. Investigations carried out in various spells during 1994 to 96 cumulatively lasted for about 12 months.

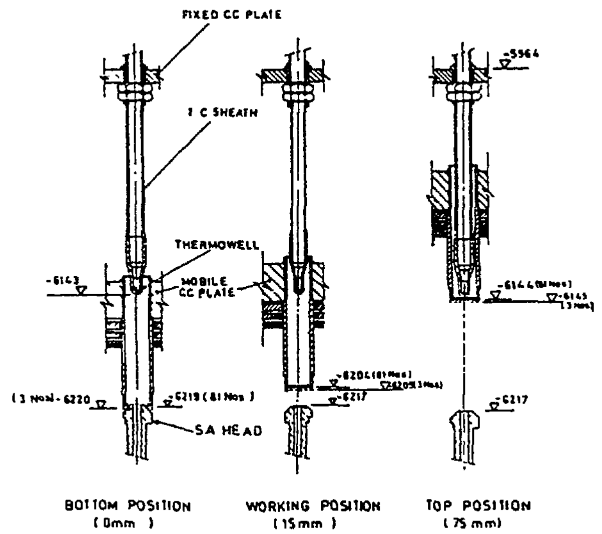
Since the incident in Apr 95, reactor has operated for about 6800 h with 4600 h at high power and the transient has not recurred.

4.4 Malfunction of Core Cover Plate Mechanism(CCPM)

The outlet temperature monitoring of 84 core SA is done by means of thermocouples housed in Core Cover Plate Mechanism (CCPM). The fuel SA thermocouples are scanned by central data processing system (CDPS) to generate trip signals. The CCPM (Fig 12) is a 6 m



SCHEMATIC OF CCPM ASSEMBLY



THREE POSITIONS OF THE MOBILE CORE COVER PLATE

CORE COVER PLATE MECHANISM

Fig.12

long, 82.5 mm dia component centrally located in the control plug and consists of a fixed and a mobile core cover plate which is translated manually by means of a command tube & ball screw assembly. The command tube houses a central canal plug having 3 thermocouples for measuring the outlet temperature of central fuel SA. The fixed plate houses the thermocouples wells (having 2 thermocouples each) and mobile plate houses the sleeves for directing the jets of sodium from the outlet of SA to the thermocouples. The mobile plate has three positions viz., fuel handling position (75 mm above SA heads), normal working position (15 mm above SA heads) and bottom position (resting over the SA heads). The leaktightness in CCPM is achieved by means of a primary barrier (SS bellows) at the bottom and secondary barrier ("O" rings and neoprene bellows) at the top. The interseal space between primary and secondary barriers is supplied with fresh argon at a higher pressure to prevent release of radioactive gas to RCB in case of a breach in the barriers.

4.4.1 Incident description

During normalisation of pile after fuel handling operations in Jul 95, CCPM could not be lowered to normal working position from fuel handling position. Various operations resulted in its getting stuck at 81 mm position above the top of SA heads. The likely causes were attributed to mechanical obstruction at the top, below the core cover plate or within the mechanism. Based on systematic investigations viz., checking for obstruction by dismantling the ball screw mechanism, scanning the space below the core cover plate and above the top of SA heads by ultrasonic under sodium scanner, ensuring leaktightness of bottom metallic bellows and checking for free movement of CCPM at the stellite guide bushes fixed to the control plug, it was confirmed that the sticking is in the interseal space having an annular gap of 1 mm between the command tube and the outer sheath. Based on safe load analysis, a jacking down force of 780 kg was applied to release the sticking and make the CCPM functional. Precise cause for malfunctioning could not be identified [(8)].

During normalising of pile after next fuel handling operation in Jul 96, CCPM again could not be lowered from 80 mm position to its normal working position.

4.4.2 Investigation

After carrying out similar checks as in 1995 and confirming the sticking in interseal seal space, a safe jacking down force upto 1000 kg was applied with sodium temperature upto 400°C but CCPM could not be moved down. Following investigations were carried out to identify the location and the nature of seizure more precisely;

- Introduction of 0.35/0.8 mm dia, 1.5 m long hypodermic needles in the annular gap between command tube and outer sheath. It could be introduced freely upto the step in the outer sheath.
- High pressure argon injection into the interseal space through these hypodermic needles to dislodge any foreign matter.
- Introduction of circular SS shim cutter (0.1 mm thick, 1.6 m long) into the annular gap. It went down freely upto the step in the outer sheath.

No smear of sodium oxide or any other foreign matter could be found by these three techniques. Load deflection measurement in the horizontal direction indicated the possible area of seizure to be below the step in the outer sheath and above the SS bellows (Fig 12) and the most likely reason could be mechanical interference. Further investigations are pursued. Cumulatively, about 8 months were spent for various investigations.

4.4.3 Implications and present status

Experiments were carried out on power to measure the fuel SA outlet temperature with CCPM stuck at 80 mm position and a temperature attenuation of 7% average was found in Mark I SA. However this attenuation is large for SA having lesser flows than the Mark I fuel. Proper monitoring of likely entry of cold slugs of Na causing reactivity transients is also affected with CCPM at 80 mm position. 3 D analysis of outlet plenum thermal hydraulic, although being a complex subject, was carried out to establish the level of plugging that can be detected viz. a viz. allowable plugging for fuel clad integrity. PSA studies based on available data for plugging of any SA during operation has also been carried out. Based on these studies, clearance was obtained for reactor operation with suitable lowering of scram thresholds generated by CDPS on fuel SA outlet thermocouples.

CCPM remains stuck at 80 mm position. This component is neither easily amenable for dismantling nor inspection below the step in the outer sheath. To improve flow and temperature measurement capability in the core, the following is planned:

- Development of an eddy current flowmeter which can be lowered through the fuel handling canal during shutdown to measure flow at the outlet of a SA as a periodic surveillance.
- Fabrication of a longer central canal plug for positioning it at 15 mm normal working position to accurately measure the outlet temperature of central fuel SA.

4.5 System modifications to improve plant availability

It is observed that in about 20,000 h of plant operation, there have been 270 trips (142 LOR / 128 scrams) which is rather large. These trips have originated mainly from neutronic instrumentation, sodium pump drive systems, uninterrupted power supply (UPS) system and steam water system. It was also noted that the plant is having 36 parameters initiating reactor trip which are also very large when compared with other fast reactors. Hence it was decided to adopt two pronged approach to improve plant availability viz.; to improve system engineering to avoid trips due to component failure / malfunction in critical systems and to carry out systematic incident analysis for eliminating unnecessary trip parameters without compromising safety.

4.5.1 Improvements in the following critical systems have been carried out;

- Replacement of neutronic and delayed neutron detector (DND) channels by state of art system. The salient features include pluggable modules facilitating on line maintenance, easy on-line testing and calibration features, microprocessor based reactivity computation and better noise immunity by use of super screened cables and opto isolators.
- Replacement of UPS with state of art system. The salient features include higher rating, regulated backup source, synchronous transfer from inverter to backup source and vice versa and inverter system following the grid frequency.
- Ward Leonard speed control system for sodium pumps was improved by eliminating pump drive trip parameters, improve speed control accuracy to ± 1 rpm and working environment by providing air conditioned enclosures for the control panels.
- Steam water system needed several modifications for improved performance viz., better design of steam bypass control valve to work in two phase flow and improvements in

hydraulic system for control valves. Modification related to replacement of contact to surface type feed heaters is planned.

- Commissioning of prestartup channels of high sensitivity to enable reactor restart after a prolonged shutdown.
- Strain gauge system for friction force measurement of control rods and provision of control rod exercising on power to always ensure its availability for safety function.
- Duplication of central data processing system with one operating and the other on auto standby.

4.5.2 Based on detailed study and incident analysis, the following safety parameters were either modified or found redundant and removed.

- 3 s interlock on control rod raising was removed to reduce startup duty demand on CRDM motors.
- Inhibition provided for reactivity trip during startup and power raising.
- Trip on negative reactivity incorporated after the reactivity transients.
- LOR on low current in CRDM electromagnetic coils removed.
- Threshold for control rod level discordance increased.
- Log P scram threshold being lowered to 10% of nominal power to ensure takeover by Lin P during power raising.
- Class II LOR (resulting in power setback) on thermal parameters in the core removed.
- Inhibition on plugging detection subroutine (PDSR) raised to 2 MWt.

5.0 CONCLUSION

- Pu-U monocarbide fuel performance has been excellent.
- Operation of sodium system and components has been very good.
- Remedial measures implemented after detailed analysis of the incidents of NaK/Na leaks in secondary circuit, water leaks in SG steam/water subheaders, fuel handling incident and MBFP seizure have been very effective.
- Inspite of detailed investigation of reactivity transients and malfunctioning of CCPM incidents, the cause could not be identified. Further efforts are in progress.
- To improve plant availability and reduce shutdowns, a large number of improvements in critical systems and safety logic have been carried out.

ACKNOWLEDGEMENT

Authors acknowledge the investigative work carried out by various Task Forces to systematically analyse the unusual occurrences in the plant. Field work carried out by the personnel of Reactor Operation and Maintenance Group is also gratefully acknowledged.

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UNUSUAL OCCURENCES DURING THE WHOLE OPERATION OF BN-350 NPP

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Abstract

Unusual occurrences during the whole operation BN-350 NPP.

1.Oil ingress in high pressure receiver for the not reveled reason, 12.05. 1994.

2.Increase of water radioactivity of circulating water supply system due to heat exchanger leak of spent fuel assembly washing out system, 17.09.1993.

3.Lack of passableness of sodium drain header of primary circuit reveled during inspection on scheduled preventative maintenance, 28.11.1996.

4.Destruction of the blow-off line of MCP-6 due to corrosion damage of the pipeline while unit was being operated at rated power, 23.04.1993.

5.Lack of passableness of blow-down pipeline connecting reactor gas cover with gas-type pressurizer while unit was being operated at rated power, 17.11.1994.

6.Sodium ingress in blow-down pipeline of loop-5 intermediate heat exchanger while loop-5 was being fed of sodium during scheduled preventative maintenance, 27.06,1994.

7.Resistance deterioration of electroheating zones of loop-4 due to heat exchanger leak and water ingress in air-pipeline of primary circuit boxes recirculating air system, 02.05.1997.

8.Resistance deterioration of electroheating zones of sodium drain header of secondary circuit was sopped in the water for the extinguishing the fire of blowing ventilation oil-strainer, 23.12.1994.

9.Sodium ingress in gas-type pressurizer through pipeline of primary sodium cleanup system and blow-down pipeline of failed MCP-2 while primary sodium cleanup system was being connected to the primary circuit, 17.08.1976.

As a rule, the main reactor systems are scrutinized more carefully than the auxiliary reactor systems and the order actions are existed for eliminating and mitigating of consequences of main reactor system fails. Therefore the auxiliary reactor system fails may impact on the main reactor systems through places of its contact in significant measure. The influence of auxiliary reactor system fails on main reactor systems and its possible consequences for behavior of the main reactor systems have been analyzed on the basis of the above-mentioned occurrences. Significance of the above-mentioned occurrences for nuclear safety BN-350 NPP have been analyzed too.

INTRODUCTION

The occurrences selected for this paper have took place only one time during the whole operation BN-350 NPP and are features for Liquid Metal- cooled Fast Breeder Reactor. At that time, when these occurrences took place, the order actions for these occurrences were not existed and identifying appropriate actions directed to eliminating and mitigating of consequences of these occurrences was enough difficult.

The first group of occurrences includes three occurrences, which were caused deficiencies of design.

The first occurrence of them is «Lack of passableness of sodium drain header of primary circuit reveled during inspection on scheduled preventative maintenance» (FIG.1). The steel disk was welded to primary circuit sodium drain header of loops-4, 5, 6 for sealing of penetration of primary circuit sodium drain header of loops-4, 5, 6 between equipment primary circuit box of loop-6 and primary circuit sodium storage system box. Freezing of primary circuit sodium and plugging of primary circuit sodium drain header of loops-4, 5, 6 throat in that place due to heat transfer augmentation connected with setting of that steel disk was reveled during inspection on scheduled preventative maintenance. Normal operational system for temperature control of primary circuit sodium drain header of loops-4, 5, 6 is thermocouples. Really controlled temperature on these thermocouples was within the limits of order threshold of electroheating (approximately 250°C). That freezing of primary circuit sodium and plugging of primary circuit sodium drain header of loops-4, 5, 6 throat was reveled on lack of passableness of sodium drain header of primary circuit while loop-4 was being fed of sodium during inspection. The primary circuit drain system is the part of the emergency core cooling and residual heat removal system. The emergency core cooling and residual heat removal system is put into operation in non-automatic mode and it is constantly in standby mode. Auxiliary electroheating zone and auxiliary thermocouple were assembled in that place for eliminating of recurrence of this occurrence. Transference of primary circuit sodium from sodium storage tank of primary circuit to free (empty) volume of sodium drain header of primary circuit one time per week was included in test program for verify of passableness of sodium drain header of primary circuit with direct method.

The second occurrence in that group is «Destruction of the blow-off line of MCP-6 due to corrosion damage of the pipeline while unit was being operated at rated power» (FIG.6). Because of local overheating of pipeline section of blow-off of MCP-6 there was accelerated corrosion damage of the pipeline. The blow-off line of MCP-6 was unsealed. This unsealing was detected only on decrease of pressure in the MCP-6 gas cover, because of the outflow of protective gas from MCP-6 gas cover was insignificant and the growth of sodium levels in MCP-6 tank and leakage drain tank-6 was not watched. The increase of aerosol activity was within the limits of norm. After separating MCP-6 gas cover from reactor gas cover, it was organized feeding of protective gas for equalization pressure between MCP-6 gas cover and reactor gas cover. Really controlled temperature on thermocouples, which control this section of blow-off line, was within the limits of order threshold of electroheating. The defect was eliminated after ending of reactor microcampaign.

The third occurrence in that group is «Lack of passableness of blow-down pipeline connecting reactor gas cover with gas-type pressurizer while unit was being operated at rated power» (FIG.3). Inner diameter of blow-down pipeline connecting reactor gas cover with gas-type pressurizer is 195 mm. That blow-down pipeline is the part of the emergency core cooling and residual heat removal system. Normal operational system for temperature control of this pipeline is thermocouples. Really controlled temperature on these thermocouples was within the limits of threshold (approximately 170 degrees C). The blower of gas cladding failure detection system transfers analyzed gas from the reactor gas cover into the MCP-1 gas cover through analyzing chambers of this system. Scheme of gas cladding failure detection system consents to transfer analyzed gas from the reactor gas cover into the MCP-1 or 6 gas

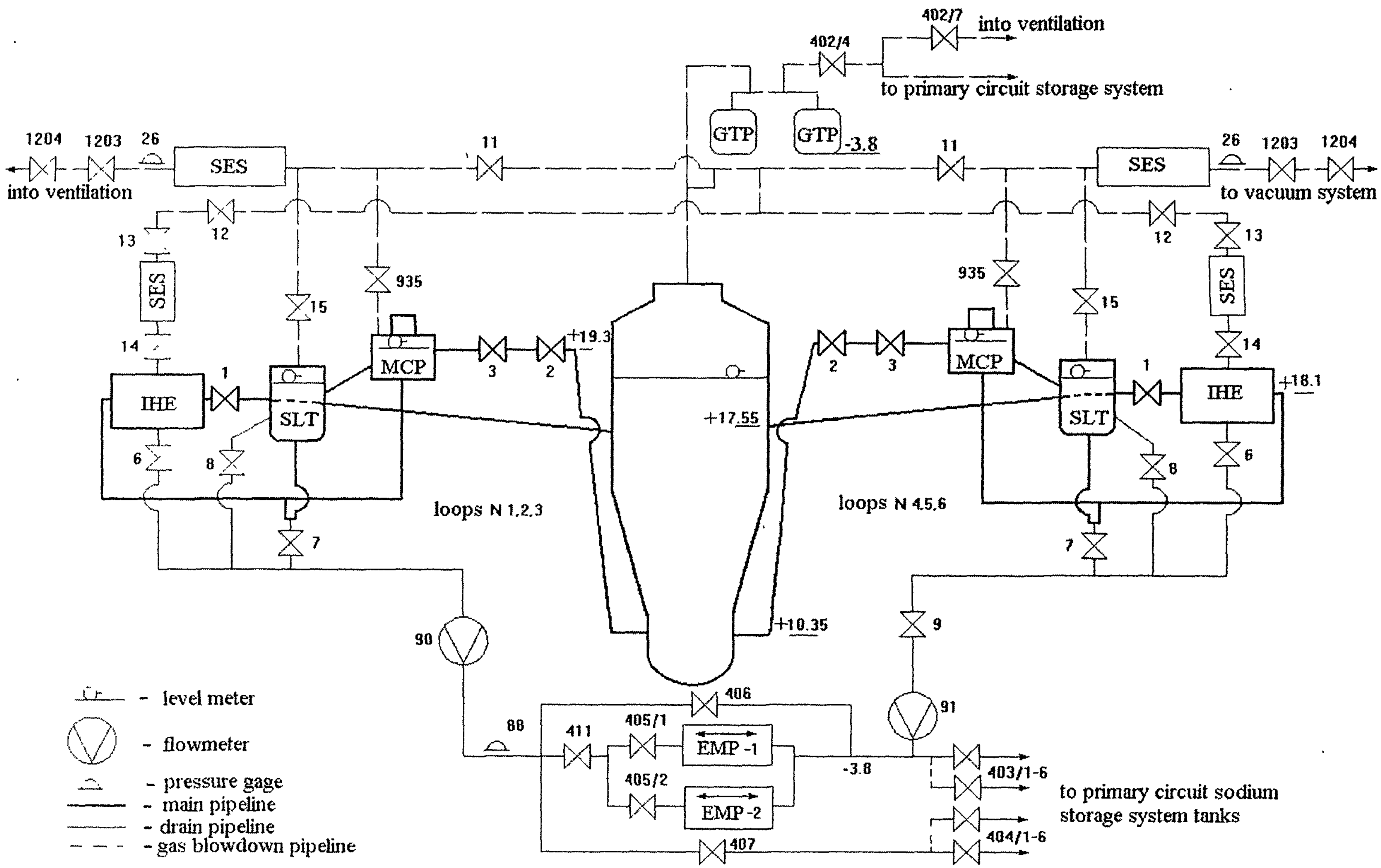

FIG. 1. Emergency cooling core and residual heat removal system.

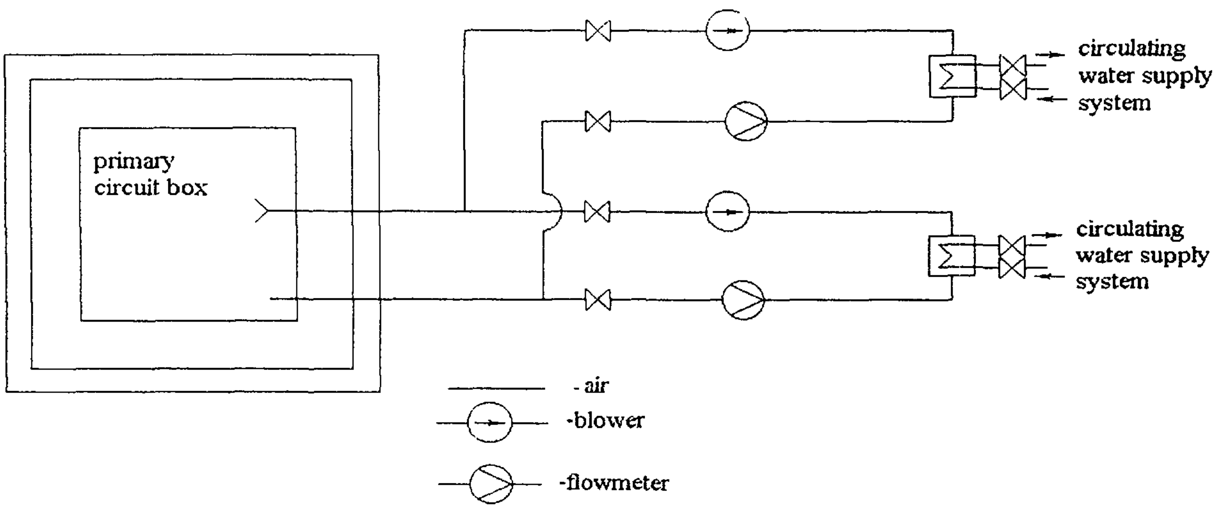

FIG.2 Primary circuit boxes recirculating air system.

cover only. Due to lack of passableness of the blow-down pipeline the pressure of reactor gas cover was down. Decrease of the reactor gas cover pressure was detected on decrease of pressure in low-pressure chamber of reactor pressure header. The direct control of reactor gas cover pressure is absented, because the sample pipeline from the reactor gas cover was plugged with sodium evaporation before, because of this sample pipeline has not sodium evaporation strainer (sodium vapor trap). After changing the electroheating threshold of electroheating zones of the blow-down pipeline connecting reactor gas cover with gas-type pressurizer from 170 degrees C to 300 degrees C the passableness of blow-down pipeline is restored. At present, the electroheating threshold of electroheating zones of the blow-down pipeline connecting reactor gas cover with gas-type pressurizer is 300 degrees C.

The second group of occurrences includes two occurrences, which were caused deficiencies of procedure.

«Sodium ingress in blow-down pipeline of loop-5 intermediate heat exchanger while loop-5 was being fed of sodium during scheduled preventative maintenance» (FIG.1) is the first occurrence in that group. Control of sodium levels of loop-5, while it was being fed of sodium, is existed with sodium level meters of MCP-5 tank and leak sodium tank-5 (intermediate heat exchanger has not its own sodium level meter). Due to worse passableness of blow-down pipelines of MCP-5 tank and leakage drain tank-5 than passableness of blow-down pipeline of intermediate heat exchanger-5, it chanced sodium ingress in blow-down pipeline of loop-5 intermediate heat exchanger. After cutting the pipeline of intermediate heat exchanger-5 and its gas blowing-down, the passableness was restored. After this occurrence procedure of feeding of primary circuit loops of sodium was changed.

The second occurrence in this group is «Sodium ingress in gas-type pressurizer through pipeline of primary sodium cleanup system and blow-down pipeline of failed MCP-2 while primary sodium cleanup system was being connected to the primary circuit» (FIG.4). The primary sodium cleanup system, connected with pipelines with pipelines of loops-2 and 3, is general for all sodium coolant of primary circuit. Loop-2 was on maintenance and separated from primary circuit with isolation gate valves N 1,2. When personal began to put into operational primary sodium cleanup system, it was fed from pressure pipeline of MCP-3

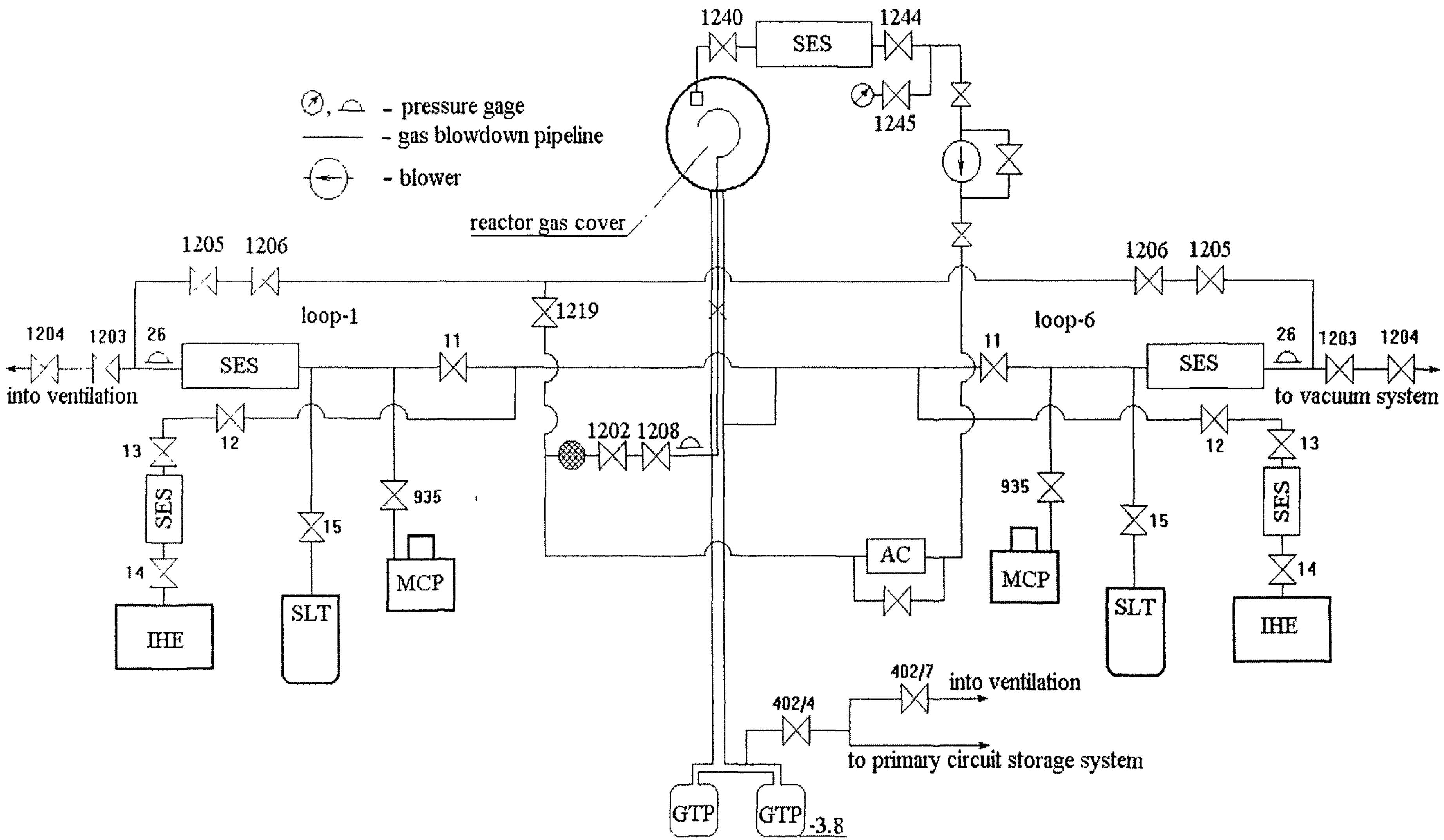

FIG.3 Gas cladding failure detection system.

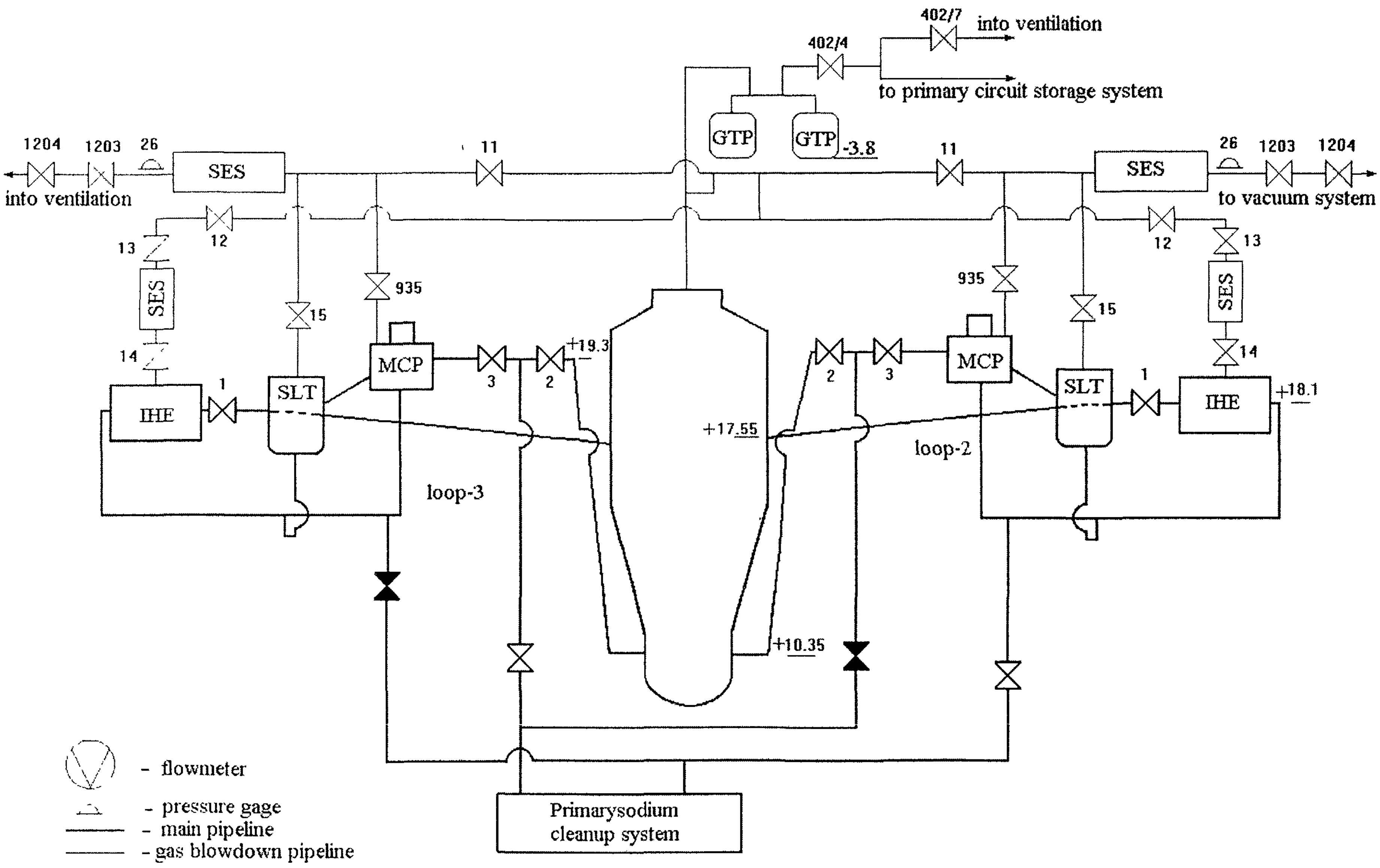

FIG.4 Scheme of sodium ingress in gas–type pressurizer.

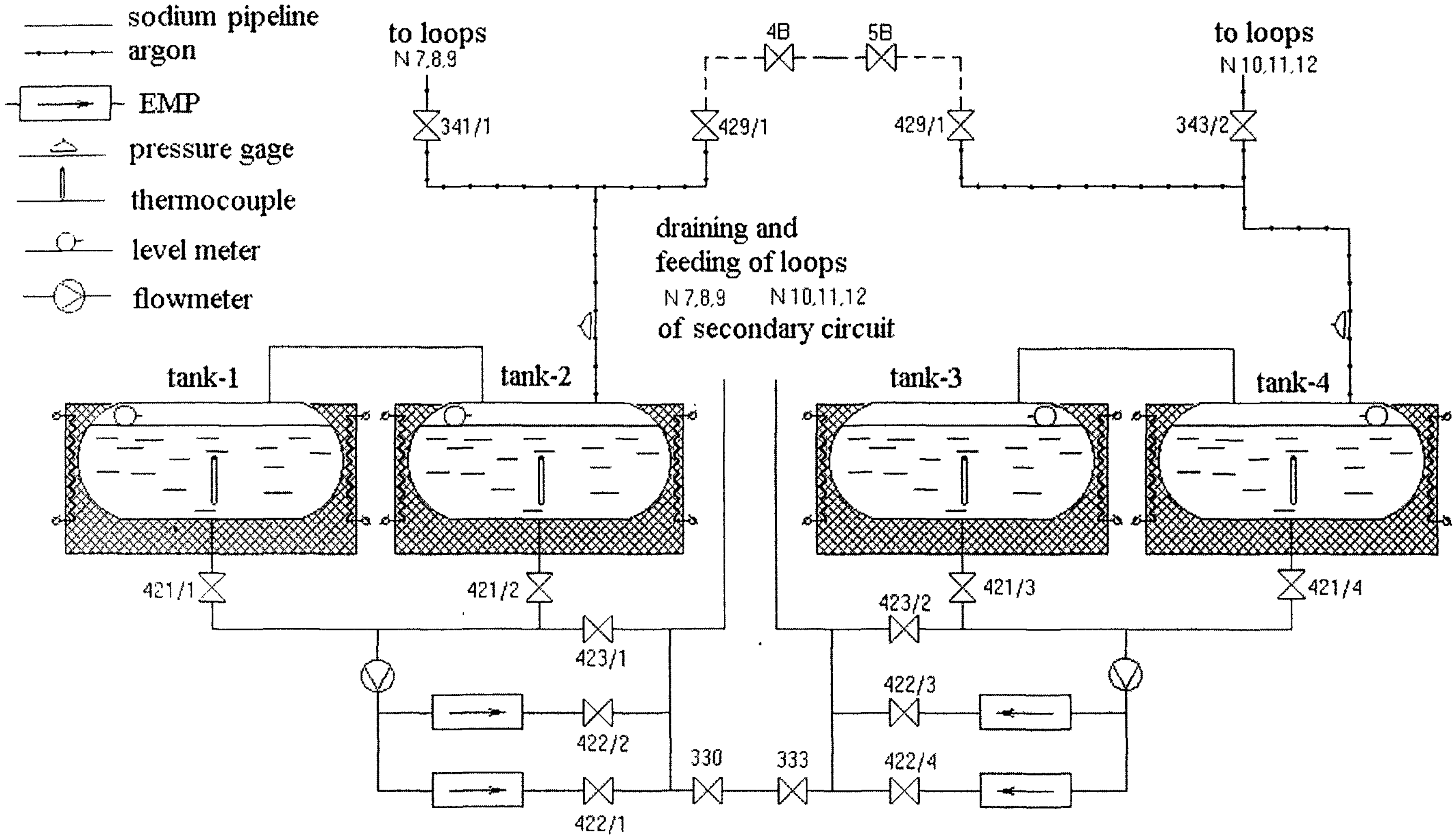

FIG.5 Drain system and Storage system of secondary sodium circuit.

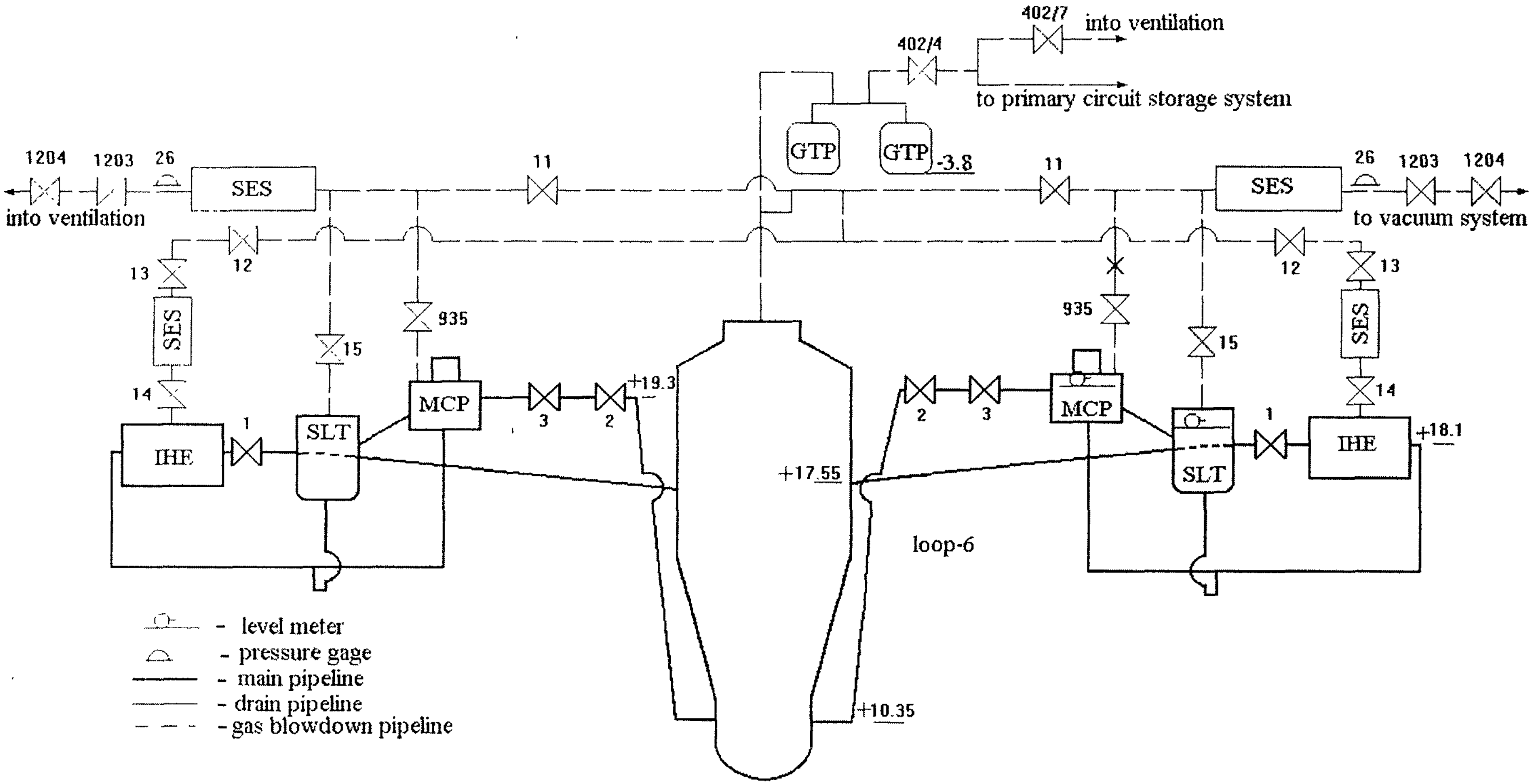

FIG.6 Scheme of blow-down pipelines of primary circuit.

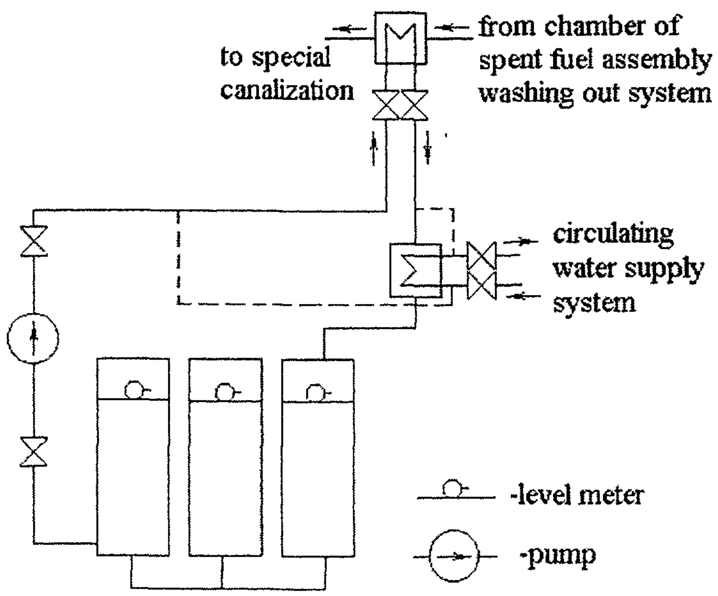

FIG. 7 Closed cooling water system.

and, by mistake, discharge pipeline of primary sodium cleanup system was connected with pipeline of failed loop-2. Immediately after supplying of sodium flow rate, sodium level in MCP-2 tank began to grow. After filling MCP-2 tank, it chanced sodium ingress in gas-type pressurizer through blow-down pipeline of MCP-2 and blow-down pipeline connecting reactor gas cover with gas-type pressurizer. Sodium level in reactor vessel began to fall and the pressure in primary circuit gas cover began to grow, because of electroheating of gas-type pressurizer was failed. Reactor was emergency shut down. After reactor emergency shutdown, sodium ingress in gas-type pressurizer was ceased. After as this occurrence was happened, procedure of connection of primary sodium cleanup system to the primary circuit was changed. At present, according to procedure, the primary sodium cleanup system is connected only to pipelines of one loop.

The third group of occurrences includes three occurrences, which were caused auxiliary reactor system fails.

The first occurrence in this group is «Resistance deterioration of electroheating zones of loop-4 due to heat exchanger leak and water ingress in air-pipeline of primary circuit boxes recirculating air system» (FIG.2). Resistance deterioration of electroheating zones of primary circuit loops is one from safety emergency alarms for sodium leak detection of primary sodium pipelines and equipment. After passing this alarm, it was found out, that auxiliary electroheating zones have resistance deterioration too. One from safety emergency alarms for sodium leak detection of primary sodium pipelines and equipment is the increasing of aerosol radioactivity in equipment primary circuit boxes. But there is no increasing of aerosol radioactivity in equipment primary circuit box of loop-4. Electroheating zones of loop-4 with resistance deterioration were turned off with keys of control. During watching air-pipelines and equipment of primary circuit boxes recirculating air system, it was found out, that due to heat exchanger leak of this system it chanced water ingress in air-pipeline of primary circuit boxes recirculating air system of loop-4. After the change-over on auxiliary air-pipeline of primary circuit boxes recirculating air system and separation of failed heat exchanger with isolation gate valves, the resistance of electroheating zones of loop-4 was restored.

The second occurrence in this group is «Increase of water radioactivity of circulating water supply system due to heat exchanger leak of spent fuel assembly washing out system» (FIG.7). Unit was under planned preventative maintenance. There was an ejection of radioactive water in normal unradioactive circuit of circulating water supply system due to heat exchanger leakage of spent fuel assembly washing out system during this system operation. This ejection of radioactivity water was found out with leak detection system on increasing of radioactivity of normal unradioactive water. The closed cooling water system was assembled for eliminating of recurrence this event. In addition to control of radioactivity of normal unradioactive water during operational that closed cooling water system, water levels in closed cooling water system tanks are being controlled too.

The third occurrence in this group is «Resistance deterioration of electroheating zones of sodium drain header of secondary circuit was sopped in the water for the extinguishing the fire of blowing ventilation oil-strainer» (FIG.5). Unit was being operated at rated power. There was an ignition of the blowing ventilation oil-strainer due to short-circuit in electric power supply lines. By forces of shift personnel and fire-fighting brigade the fire was located and extinguished. But during the extinguishing the fire of blowing ventilation oil-strainer, the sodium drain header of secondary circuit was sopped in the water. Electroheating zones of the sodium drain header of secondary circuit with resistance deterioration were turned off with keys of control because of danger of short-circuit between electroheating zones, that electric power supplied from different phase of the same transformer. The pump sodium circulation was organized for drying electroheating zones of sodium drain header of secondary circuit. The heated secondary circuit sodium transference was organized from the sodium storage system tanks of secondary circuit into working loops of secondary circuit and return. After drying and restoring of resistance of electroheating zones of sodium drain header of secondary circuit, the pump sodium circulation was ceased. Electroheating zones of sodium drain header of secondary circuit was turned on according to order thresholds of electroheating.

The last occurrence in this paper is «Oil ingress in high pressure receiver for the not reveled reason». Protective gas (argon) is retained in three high pressure receivers for technology needs, such as gas feeding the reactor gas cover, equipment gas covers of primary and secondary circuit, tank gas covers of primary and secondary circuit sodium storage system and etc. According to test program, the protective gas in these high pressure receivers are analyzed one time per week. The protective gas is fed by high pressure compressor from argon gas bottles into high pressure receiver. Before the gas feeding, pure argon sample from these argon gas bottles is selected for the analysis. If this analysis is within the limits of norm, the protective gas is being fed into the high pressure receiver. During selection of pure argon sample for the analysis from high pressure receiver the presence of oil was revealed. The reason of oil ingress was not identified. This occurrence itself has no real or potential nuclear safety or radiological safety relevance, but if oil ingress in high pressure receiver was not revealed promptly, this occurrence maybe significant in its consequences. Possibly, such occurrence or the similar occurrences will be able to explain such occurrences as abnormal behavior reactivity and other events that took place in the reactors due to not optimal design, technology and operational practice.

CONCLUSION

For systems with pipelines filled of sodium, which are in standby mode and are emergency safety systems or part of these systems, very important, that direct method for control availability of these systems is existed. This demand is connected with enough higher temperature of sodium melting.

The main reactor system fails are caused with influence of auxiliary reactor system fails are able to be enough difficult for identifying initial origin and appropriate actions directed to eliminating and mitigating of consequences of these occurrences.

At present, mistakes of personal are caused deficiencies of procedure for auxiliary reactor systems, especially if these mistakes are not identified promptly, maybe significant at that time, when the reactor main system fail will be took place.

IMPACT OF LMFBR OPERATING EXPERIENCE ON PFBR DESIGN

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XA0056258

Abstract

PFBR is a 500 MWe, sodium cooled, pool type, fast breeder reactor currently under detailed design. It is essential to reduce the capital cost of PFBR in order to make it competitive with thermal reactors. Operating experience of LMFBRs provides a vital input towards simplification of the design, improving its reliability, enhancing safety and achieving overall cost reduction. This paper includes a summary of LMFBR operating experience and details the design features of PFBR as influenced by operating experience of LMFBRs.

1. INTRODUCTION

India has limited uranium and abundant thorium resources. For better utilisation of uranium and to use the available thorium, Fast Breeder Reactor programme is very essential. A 40 MWt / 13 MWe Fast Breeder Test Reactor (FBTR) is in operation at Kalpakkam, since 1985, and has attained its rated power level of 10.5 MWt in Dec 1993 with small Mark I core. As a logical follow-up to FBTR, a 500 MWe Prototype Fast Breeder Reactor (PFBR) is currently under design & development at Indira Gandhi Centre for Atomic Research (IGCAR). Fast reactors like BN-600, SPX-1 and Monju have capital cost significantly higher than PWRs. FBRs, for commercial deployment, after reaching maturity through large scale construction, would be required to have matching unit energy cost with PWRs. Even though incidents of minor to severe nature have occurred in LMFBRs, their operating experience provides a very important input for design simplification, improving reliability, enhancing safety and achieving cost reduction. This paper describes in brief the summary of LMFBR operating experience and details the design features of PFBR as influenced by LMFBR operating experience.

2. SUMMARY OF LMFBR OPERATING EXPERIENCE

- Considerable experience (~ 200 reactor-years) has been gained in the design and operation of sodium systems. Corrosion of structural materials in reactor grade sodium is negligible and purification of sodium for oxygen control by cold trapping is very satisfactory. It has been possible to maintain the purity of sodium in a stable manner.
- Performance of stainless steel of type 304 and 316 for sodium components is excellent. Failure of welds have occurred in the stabilised grade 321 in PFR [1] & Phenix [2] and 15 Mo3 ferritic steel in SNR-300 & SPX-1 [2]. Performance of elevated temperature components on the whole is satisfactory indicating that failure mode of creep-fatigue can be well taken care of in design.

- Very high burnup (192 GWd/t) has been achieved for mixed oxide fuel compared to the initial target value of about 62 GWd/t [3] and there has been very few fuel pin failures. This gives scope for significant decrease in fuel cycle cost. The smaller number of fuel pin failures has led to very clean sodium circuits which has also contributed to low radiological impact.
- The performance of sodium pumps has been very good. The only incident of concern has been oil leaks with a major incident occurring in PFR [4]. Other problems like excessive vibrations, seizure, malfunctioning of speed control etc have been understood.
- IHX performance except for Phenix is excellent. Failure of Phenix secondary sodium outlet header is well understood [5].
- Successful operation of steam generator (SG) holds the key to achievement of high capacity factors. SG requires the high quality during manufacture and a sensitive leak detection system for sodium-water reaction detection and mitigation. PFR incident of failure of 40 tubes of superheater has led to redefinition of design basis leak for SG from the earlier considered incident of double ended rupture of one tube [6].
- Fuel handling incidents have led to interventions & outages in some reactors including FBTR where bypassing of interlocks was done. These incidents call for under sodium scanning before every fuel handling.
- Radiation dose to operating personnel and radioactivity releases to the environment are significantly less compared to PWRs [7]. It is thus possible to define a low target person-Sivert for LMFBR and to reduce the shielding in controlled access areas to enable cost reduction.
- In-service inspection (ISI) is an important means to assess structural integrity and needs attention in design in particular, for main vessel & SG.
- Reactivity incidents have occurred in some reactors and in spite of the best efforts, fully validated explanation has not been possible. It is worthwhile to have a design with less potential for such incidents.
- Fuel meltdown incident has occurred in Fermi due to blockage of subassembly at inlet and at EBR I due to inward bowing of core subassemblies.
- Incidents of sodium leak show need for greater care in auxiliary sodium circuits to minimise failure by thermal striping. There is a need to develop sodium resistant concrete to minimise damages to structures in case of sodium leaks.
- High capacity factors are achievable in LMFBR with sound designs as experienced in EBR II, BOR-60, Phenix (initial years) and BN-600.
- The capital cost of FBR is about 1.5 to 2.5 times that of thermal reactors and significant cost reduction is essential for its successful deployment. Cost reduction measures adopted in LMFBR include elimination of ex-vessel sodium storage, decrease in number & size of components of heat transport system, compact layouts, increasing operating temperature, increasing plant life and increasing fuel burnup.

3. FBTR OPERATING EXPERIENCE

Details of operating experience and incidents that have occurred in FBTR are covered in a companion paper.

4. PFBR DESIGN FEATURES

4.1 Main options

4.1.1 Reactor Power

The successful operation of 500 MWe thermal power plants in India has enabled to fix PFBR reactor power as 500 MWe. Large sized FBR have not indicated any technological problems because of reactor size. Specific capital cost is lower for 500 MWe than for a lower power, say 250 MWe. The design and development efforts needed for 500 MWe and 250 MWe plants are comparable. Pressurised Heavy Water Reactors (PHWR) of 500 MWe are under construction in India. Constructability of 500 MWe PFBR components has been assessed and adequate industrial capability exists within the country.

4.1.2 Fuel

A proven fuel cycle is very essential for PFBR. Though mixed carbide fuel has been used for FBTR due to non availability of enriched uranium, risks associated with carbide fuel fabrication, higher cost coupled with limited burnup potential & limited experience on reprocessing of the fuel have led to adoption of mixed oxide (MOX) fuel. This fuel has shown excellent performance with respect to burnup, has well proven reprocessing technology and has also been used in most of the large sized FBR.

4.1.3 Loop vs Pool concept

Better safety features of the pool concept due to the high thermal inertia of the large mass of sodium in the pool, containment of all radioactivity in a single vessel with no nozzles leading to high integrity of the primary circuit, reliable decay heat removal by independent dedicated sodium loops and satisfactory performance of pool type power reactors abroad have led to adoption of pool type concept for PFBR. The shortcomings of the pool concept as regard to large size of components of reactor assembly, complex thermo-hydraulics of hot and cold pool, interdependence of primary circuit component's construction and maintenance are well recognized and have been looked into.

4.1.4 Operating Temperatures

In order to reduce the unit energy cost, it is essential to adopt a superheated steam cycle. Operating experience of elevated temperature components in FBR indicates that creep-fatigue damage can be well taken care of in the design. Thermo-hydraulic analysis needs to be detailed to have complete knowledge of thermal loading.

Plant temperatures have been arrived at based on structural analysis of hot leg components, in particular Control Plug, limiting clad hot spot temperature to 973 K (700 $^{\circ}$ C), steam generator material as T91 and optimisation studies on heat exchangers (IHX & SG) costs and sodium pumping cost. The temperatures of 820 K (547° C) at hot pool, 670 K (397° C) at cold pool, 628 K (355° C) at IHX inlet, 798 K (525° C) at IHX outlet and steam conditions of 16.7 MPa / 763 K (490° C) at turbine inlet have been chosen. The improved cycle efficiency with higher temperature difference across HXs result in reduction in unit energy cost.

4.1.5 Structural Materials

4.1.5.1 Clad and Wrapper

20 % CW D9 material which has shown excellent performance with oxide fuel has been selected for clad tubes and wrapper. Irradiation of indigenously produced material is planned in FBTR. Wrapper in Cr-Mo grade is also envisaged for future cores of PFBR.

4.1.5.2 Material for Hot leg and Cold leg components in sodium circuits

SS 321 has been rejected due to unsatisfactory performance in FBRs and thermal power stations. SS 347 is expected to behave similar to SS 321 at elevated temperatures. SS 316 LN, which has good high temperature characteristics and provides freedom from sensitisation in as welded state - an important aspect to avoid risk of IGSCC, in the coastal site selected, has been chosen for hot leg components such as inner vessel, control plug, IHX and hot leg of secondary sodium piping. For the cold leg components and secondary sodium piping, SS 304 LN material is found to be adequate. However, use of SS 316 LN for cold leg components and piping would be given consideration where risk of mixup of materials exists. Choice of a single grade also reduces the R&D efforts required.

4.1.5.3 Material for Steam generators

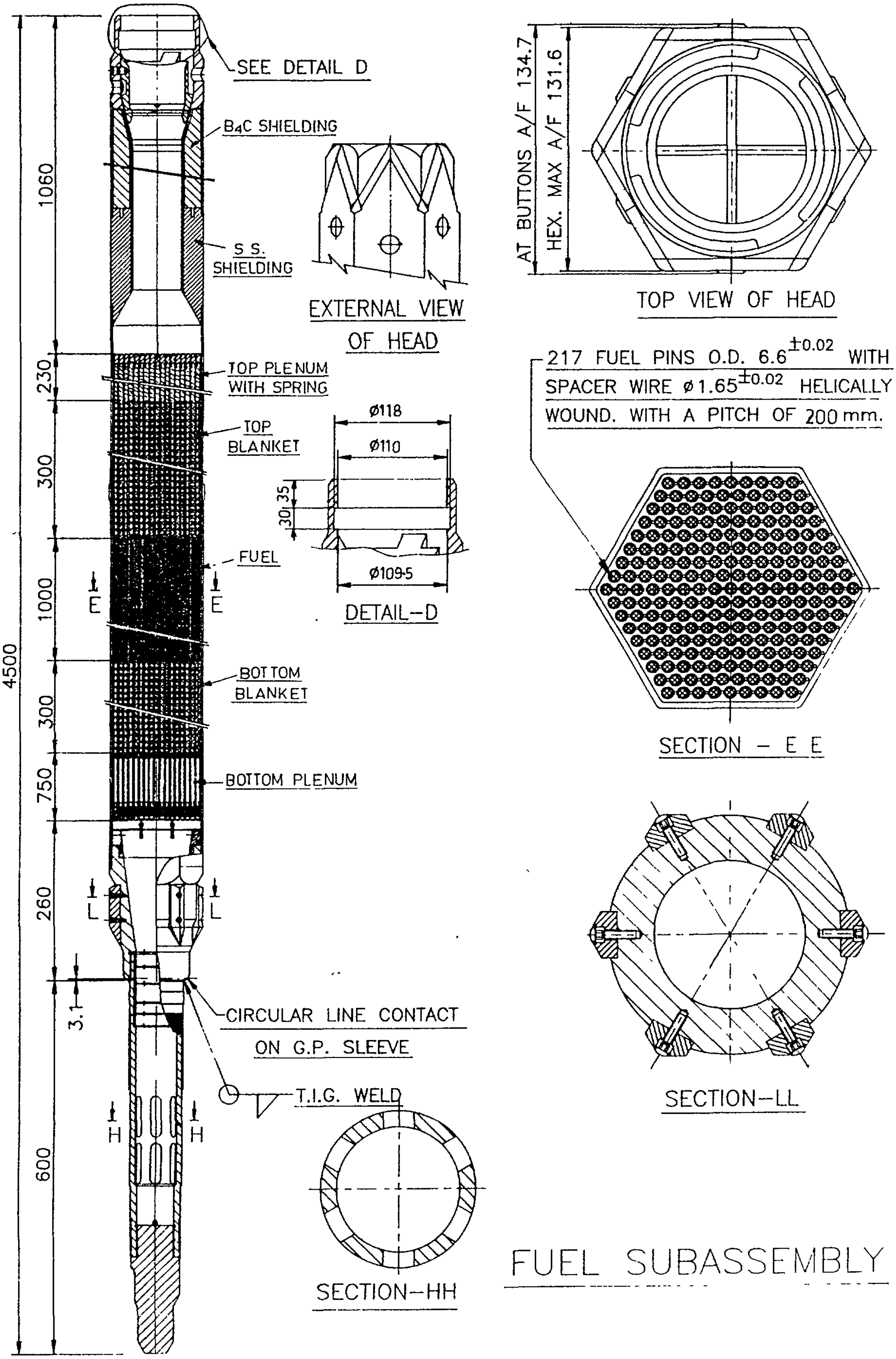
Modified 9 Cr-1Mo (T91) has been chosen for steam generators because of its satisfactory strength at high temperature, freedom from stress corrosion cracking (problem with stainless steels both for chloride and caustic environment) and risk of decarburisaiton (problem with 2.25 Cr-1Mo).

4.1.6 Number of Turbogenerator(TG) sets

Operating experience from nearly 15 TG sets of 500 MWe capacity, currently in operation in India, is excellent. A single TG set has been selected instead of two from considerations of reduction in capital cost and improved capacity factor arising from reduction in outages due to maintenance.

4.2 Core Design

The active core consists of 181 fuel subassemblies with two enrichment zones, of which 85 with ~ 21% PuO_2 content are in the inner enrichment zone and 96 with ~ 28% PuO_2 content are in the outer enrichment zone. Each fuel subassembly consists of 217 helium bonded pins of 6.6 mm outside diameter. Each pin has 1000 mm column of MOX, 300 mm each of upper and lower depleted UO_2 blanket columns and lower fission gas plenum (fig. 1).


SEE DETAIL D
B4C SHIELDING
S.S. SHIELDING
EXTERNAL VIEW OF HEAD
AT BUTTONS A/F 134.7
HEX. MAX A/F 131.6
TOP VIEW OF HEAD
1060
230
TOP PLENUM WITH SPRING
TOP BLANKET
300
Ø118
Ø110
30.35
30
Ø109.5
DETAIL–D
217 FUEL PINS O.D. 6.6 ±0.02 WITH SPACER WIRE ø1.65 ±0.02 HELICALLY WOUND. WITH A PITCH OF 200 mm.
FUEL
1000
E
4500
BOTTOM BLANKET
300
SECTION – E E
BOTTOM PLENUM
750
260
L
3.1
CIRCULAR LINE CONTACT ON G.P. SLEEVE
T.I.G. WELD
SECTION–LL
H
600
SECTION–HH
FUEL SUBASSEMBLY

Fig—1

The fuel pellets are of annular type which enables faster rise to full power. Absence of fuel redistribution & restructuring because of high density helps in attaining high burnup.

There are 3 rows of radial blanket subassemblies and 12 absorber rods arranged in two rings with 9 constituting the Control & safety rods(CSR) and 3 constituting the Diverse safety rods (DSR). Boron carbide with 63% / 50% enriched B_{10} for CSR/DSR respectively is chosen as the absorber material. The control & safety rods are of vented type and this type of design has performed well in FBTR and in other reactors.

Total blockage of SA due to external debris is a low probable event and is taken care by the arrangement of radial inlet of coolant and multiple holes inlet in grid plate sleeves / multiple slots inlet in the SA foot. Total blockage of fuel and blanket SA at outlet is ruled out by provision of an adaptor, which ensures an alternate path for coolant flow. This consists of a annular cylindrical piece with slots provided for sodium flow and this is screwed to the inside of the top portion of the SA. Two sets of 3 nos. of holes are provided on the SA hexcan outside the adaptor. In case of a total blockage of the flow path at the top of the SA, these holes along with the slots in the adaptor provides alternate flow path for sodium. The design objective is to avoid sodium boiling. During normal operation, a small flow of about 0.2 % of the flow through a SA leaks through the above holes.

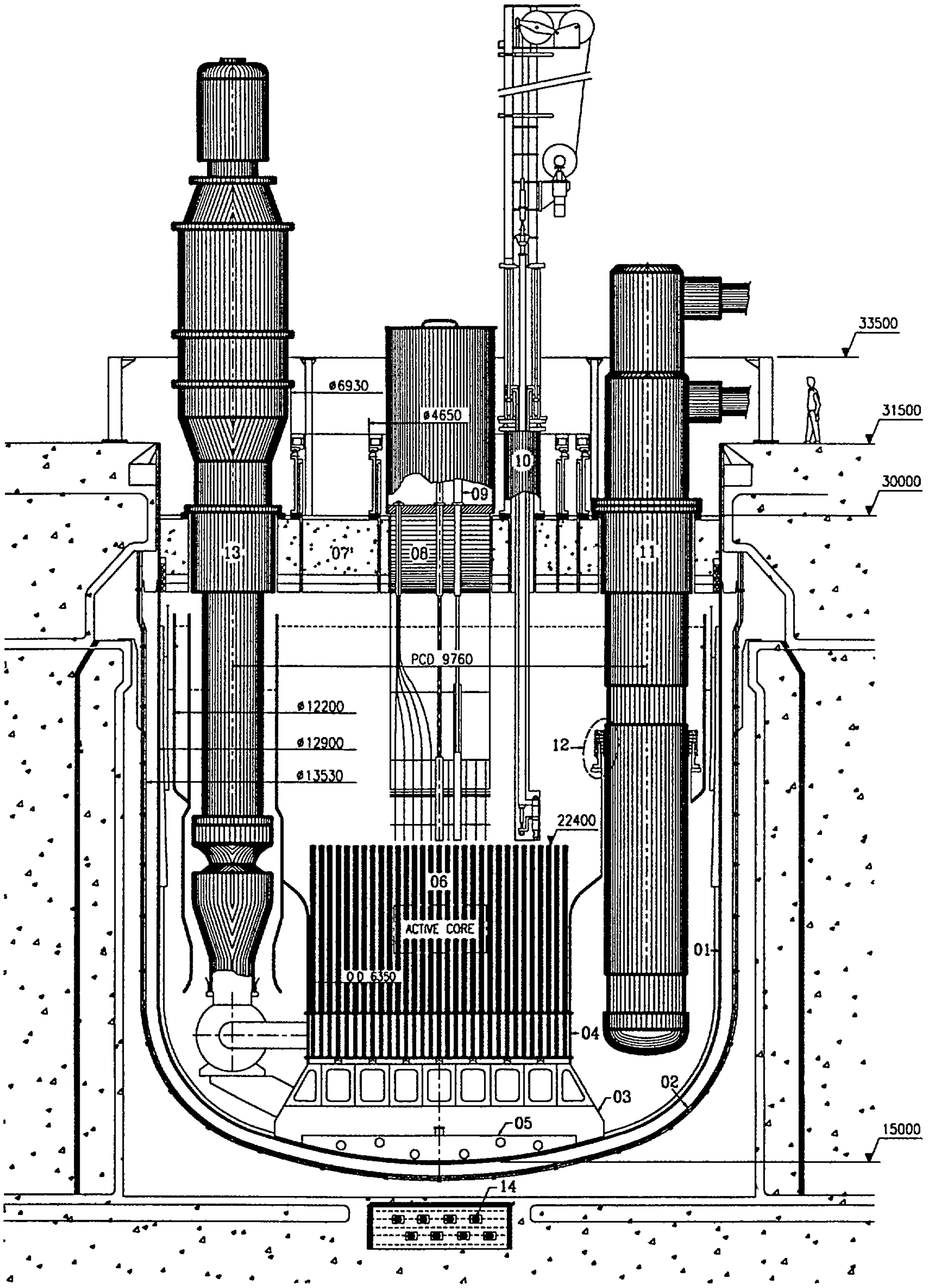
Simple naturally restrained core concept has been adopted which gives negative power coefficient and which also avoids inward radial movement of core subassemblies due to bowing. Use of a separate core barrel has been avoided.

4.3 Reactor Assembly

The reactor assembly consists of main vessel, safety vessel, core support structure, grid plate, inner vessel, roof slab, rotatable plugs and control plug (fig. 2). The main vessel (diameter 12.9 m) contains the entire primary sodium circuit including the 1100 t of primary sodium. The main vessel is cooled by cold sodium to enhance its structural reliability. The main vessel cooling arrangement has been checked for flow induced vibration behavior. The safety vessel follows the shape of the main vessel with a 300 mm nominal gap. The inner vessel separates the hot and cold pools of sodium. Argon gas seal for IHX-Inner vessel sealing has been avoided to minimise the chances of reactivity addition and a mechanical seal design with piston rings has been selected (fig. 3). The seal assembly uses two piston rings and is provided as an integral part of the IHX. It has a face to face contact with a flange integral with IHX standpipe in inner vessel. Compression springs are used to apply the required force in order to minimise leakage of sodium between the flange faces. Hydraulic experiments are planned to verify the quantity of sodium leaking across the seal assembly.

A single grid plate is used to support the core and shielding subassemblies and a fully bolted construction has been adopted. The grid plate has four inlet pipes with a pair of nozzles connected to each of the two primary pumps.

The top shield includes roof slab and two rotatable plugs. Warm roof concept is adopted for top shield to minimise sodium deposition in the annular gaps. The roof slab is a box type structure filled with concrete as the shielding material. It supports the main vessel, primary sodium pumps, IHX, and direct reactor heat exchangers (DHX) of the decay heat removal system. Use of liquid metal seals has been avoided in order to reduce the rotatable support



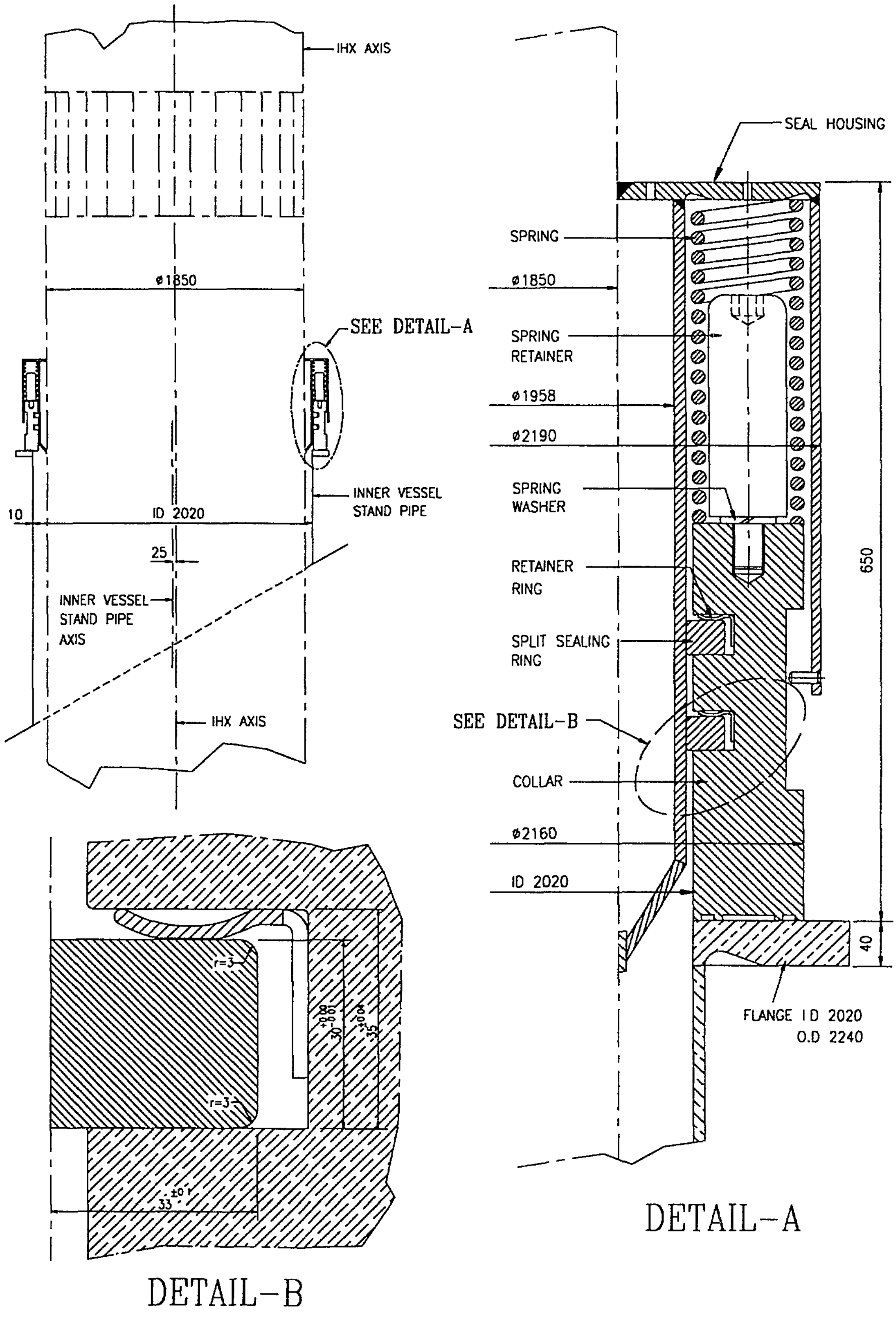
LEGEND

01. MAIN VESSEL
02. SAFETY VESSEL
03. CORE SUPPORT STRUCTURE
04. GRID PLATE
05. CORE CATCHER
06. CORE
07. TOP SHIELD
08. CONTROL PLUG
09. CONTROL & SAFETY ROD DRIVE MECHANISM
10. IN-VESSEL TRANSFER MACHINE
11. INTERMEDIATE HEAT EXCHANGER
12. IHX MECHANICAL SEAL
13. PRIMARY PUMP & DRIVE
14. NEUTRON DETECTORS

P.V.SELLAPERUMAL

PFBR REACTOR ASSEMBLY

FIG.2



IV-IHX MECHANICAL SEAL ARRANGEMENT

FIG-3

arrangement width (and hence main vessel diameter) and elastomer seals are used to seal the argon cover gas.

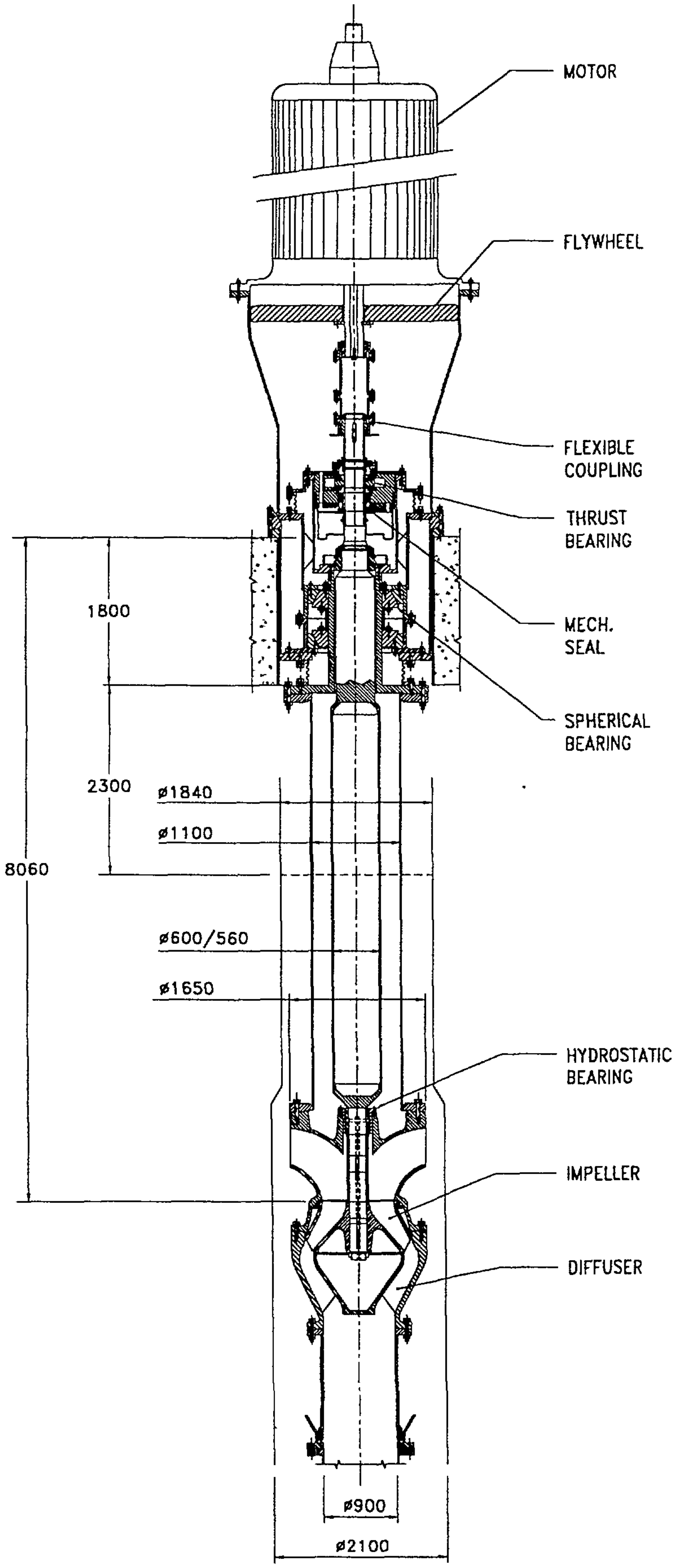
The Control plug supports the 12 absorber drive mechanisms, sleeves which house thermocouples for measurement of outlet temperature of each fuel subassembly and three selector valves with sodium sampling from each fuel SA for failed fuel location. Use of bellows has been avoided for CSRDM to extend the life of the mechanisms and to enhance reactor availability, as bellows failure has been responsible for replacement of CRDMs in reactors using this concept. V-ring seals are used between the stationary sheath and mobile assembly of CSRDM. The core thermocouples are located at a fixed distance of 90 mm from the top of the SA during reactor operation and no Core Cover Plate Mechanism (CCPM) is provided as in FBTR. Thermohydraulic analysis indicates that the thermocouples are immersed in their respective streams at all power levels thereby ensuring adequacy of temperature measurement.

Though the Total Instantaneous Blockage (TIB) of a single SA is categorised as Beyond Design Basis Event (BDBE), an internal core catcher is provided below the core support structure. This is designed for retention of core debris arising out of meltdown of 7 SA based on the SCARABEE tests which have indicated melt propogation at the most to the neighboring six SA.

4.4 Sodium circuits & Components

Detailed optimisation studies on number of loops/components led to the choice of 2-loop concept. Due to adoption of design improvements, the increase in size of the components when the number of loops is decreased is not large and is within the industrial capacity. The reduction in the number of components helps to reduce the capital cost, construction time and the outage time due to generic design failure/inspection/repair of components. Hence the capacity factor of the reactor is expected to be marginally higher for the case with lesser number of loops. Reduction in number of loops also reduces the space required for layout of secondary sodium system components. Hence, 2 loop arrangement has been chosen with two primary pumps and 2 secondary pumps. 2 IHX per loop has been selected based on the economics and is in line with other pool type reactors built so far. The number of SG/loop is based on optimisation analysis of capital cost and outage cost in case of a leak, with due consideration to construction schedule while permitting (N-1) SG modular operation and 4 SG/loop has been chosen.

The Primary pump is a top suction, single stage, centrifugal pump without non-return valve (NRV)(fig. 4), powered by a 3600 kW motor with speed variation of 20-100% of nominal speed. It delivers a flow of 4.13 m^3/s at a head of 75 mlc at an operating speed of 680 rpm. A squirrel cage induction motor fed from current source inverter is selected. A pony motor is provided to run the pump at 20% speed. It is not envisaged to operate the reactor with only one pump in operation. Further, analysis indicates that the flow through the core is adequate in case of one pipe rupture (category 4 event) even without NRV. Hence, NRV is eliminated giving the advantage of increased submergence (for a given main vessel height) thereby permitting higher pump operating speed. Elimination of NRV also increases reliability of the path for decay heat removal. A margin of 1.24 is specified on NPSH ($NPSH_A / NPSH_{3\%}$), which ensures absence of cavitation erosion and gives a pump life equal



PRIMARY SODIUM PUMP

FIG. 4

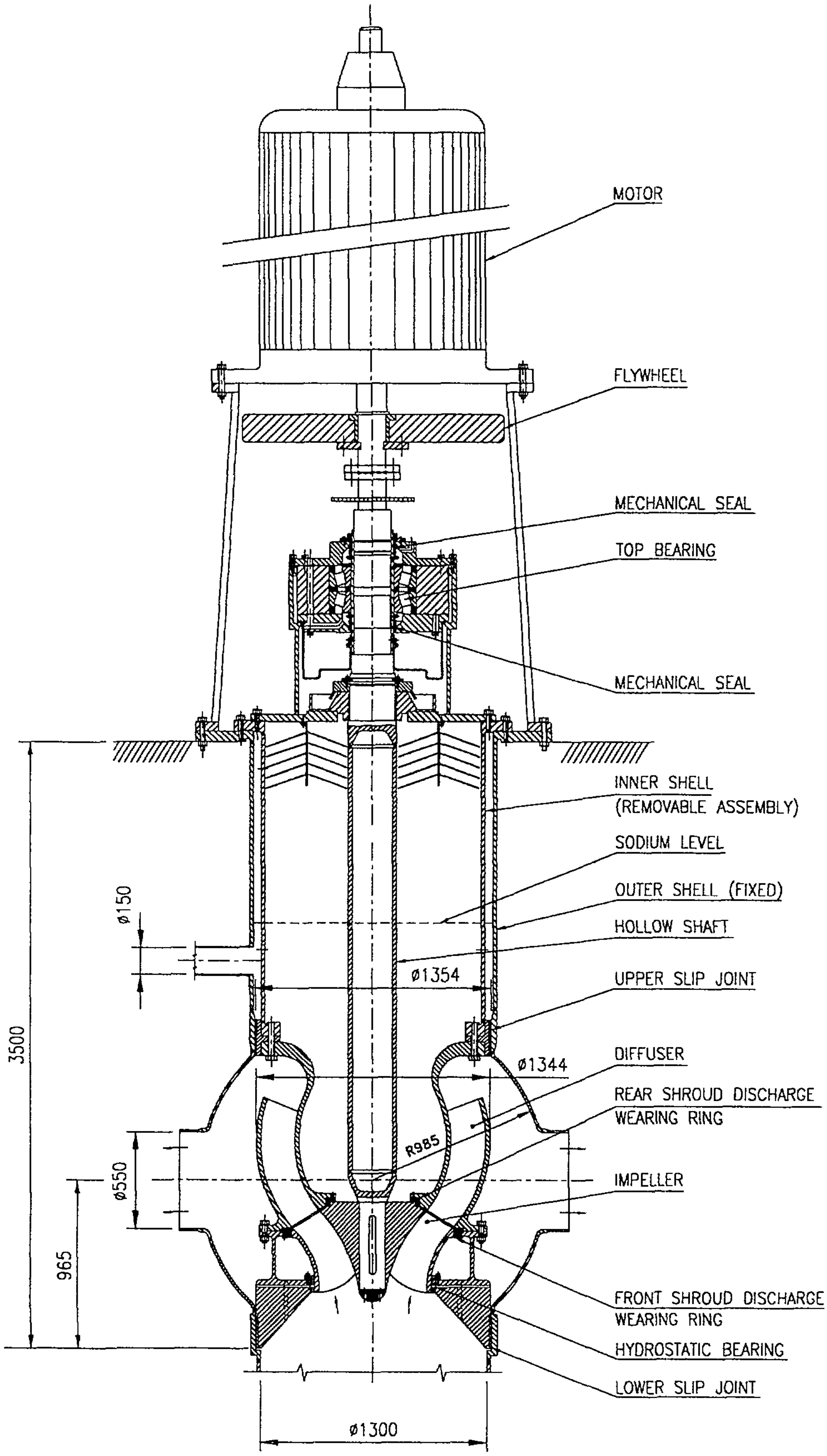
to that of the reactor. The hydrostatic bearing journal is keyed to the shaft and is also provided with a spacer preventing danger of its slippage due to thermal shocks. It has not been possible to avoid use of oil for lubrication of seals and bearings. Hence, efforts are made to avoid entry of oil into sodium. Any possibility of oil leak into the primary circuit is avoided by appropriate surveillance methods as well as by provision of an oil catch pot of sufficient size to accommodate the entire oil capacity. The pump is supported on a spherical seat arrangement to accommodate the differential thermal expansion. Full-scale hydraulic testing of the prototype pump is being done.

The secondary pump is of centrifugal type, mixed flow design delivering a flow of 3.34 m^3/s at a head of 65 mlc at an operating speed of 960 rpm (fig. 5) and is located in the cold leg at a lower elevation with respect to SG. Locating the pump in the cold leg of the secondary circuit is more economical with lower piping costs. The normal cover gas pressure in the pump tank is 0.3 MPa(g). Any danger of flooding of the secondary pump is prevented by a suitably designed piston ring seal (in the upper slip joint) separating the high pressure pump discharge from the relatively low pressure cover gas space. This seal will be experimentally tested to validate its design.

The IHX is a vertical, counter current flow, shell and tube heat exchanger (fig. 6). Each IHX has 3000 straight tubes (19 mm OD x 0.8 mm WT) with primary sodium on shell side and secondary sodium on the tube side. The tubes are arranged in circumferential pitch. A variable flow distribution is provided inside the IHX tubes with a higher flow on the outer rows to improve the thermohydraulic behaviour of the tube bundle. A mixing device is also provided at the secondary outlet to reduce the temperature differences between inner and outer shell of the secondary outlet header. Absence of flow induced vibration of tube bundle and the drain pipe in the downcomer have been verified by theoretical analysis.

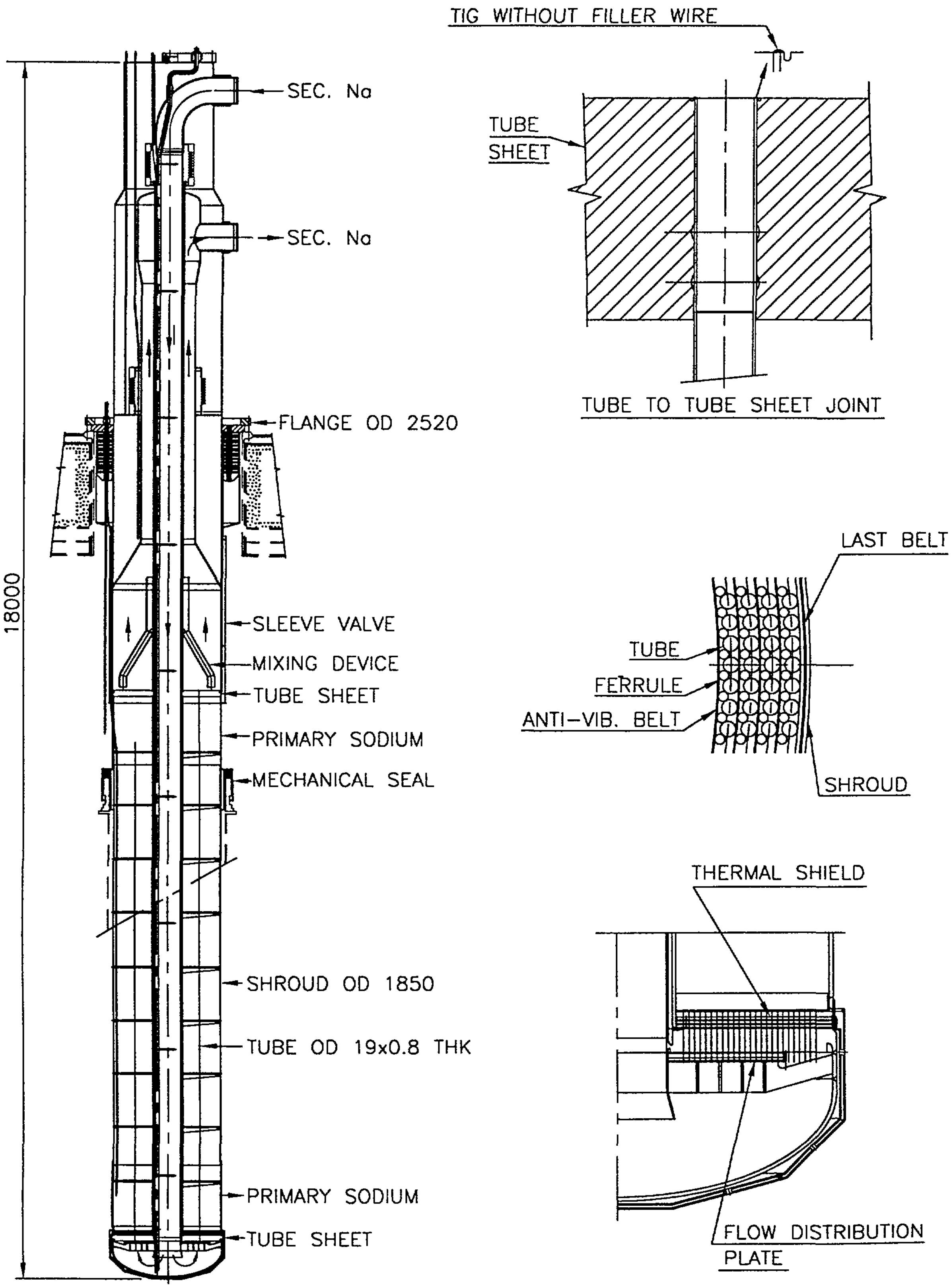
Steam reheat with integrated once-through design for the steam generators has been selected. This has been favoured over sodium reheat as the savings in SG cost, reduction in construction time and ease of design & operation outweigh the marginal advantage in efficiency associated with sodium reheat.

The SG selected is a vertical countercurrent, shell and tube heat exchanger with sodium on the shell side (fig. 7). No cover gas is provided in the SG and a surge tank is provided on the upstream side of the SG. This arrangement is less costly. Experience of multiple SG / loop without cover gas in other reactors is also good. Straight tube design with an expansion bend in each tube located in the bottom portion of the SG above the sodium outlet nozzle has been selected to take care of differential expansion between shell and tubes as well as amongst tubes. Sodium enters the SG through a single inlet nozzle, flows upwards in the annular region & top inlet plenum before entering the tube bundle. A flow distribution device is located in the annular region to bring uniformity in tube bundle flow. Sodium leaving the SG exits through the bottom outlet plenum and a single outlet nozzle. An orifice is provided at the water inlet of each tube of SG from stability consideration. The tubes are supported at various locations by formed type tube bundle support arrangements. Tube to tubesheet joint is of internal bore weld type with raised spigot to enhance reliability (crevice free and radiographable) of this critical weld joint. Long seamless tubes are used in order to reduce the number of tube to tube welds. The inspections proposed for each joint include dye penetrant testing, radiography using anode (microfocus) X-ray and helium leak testing. It is also



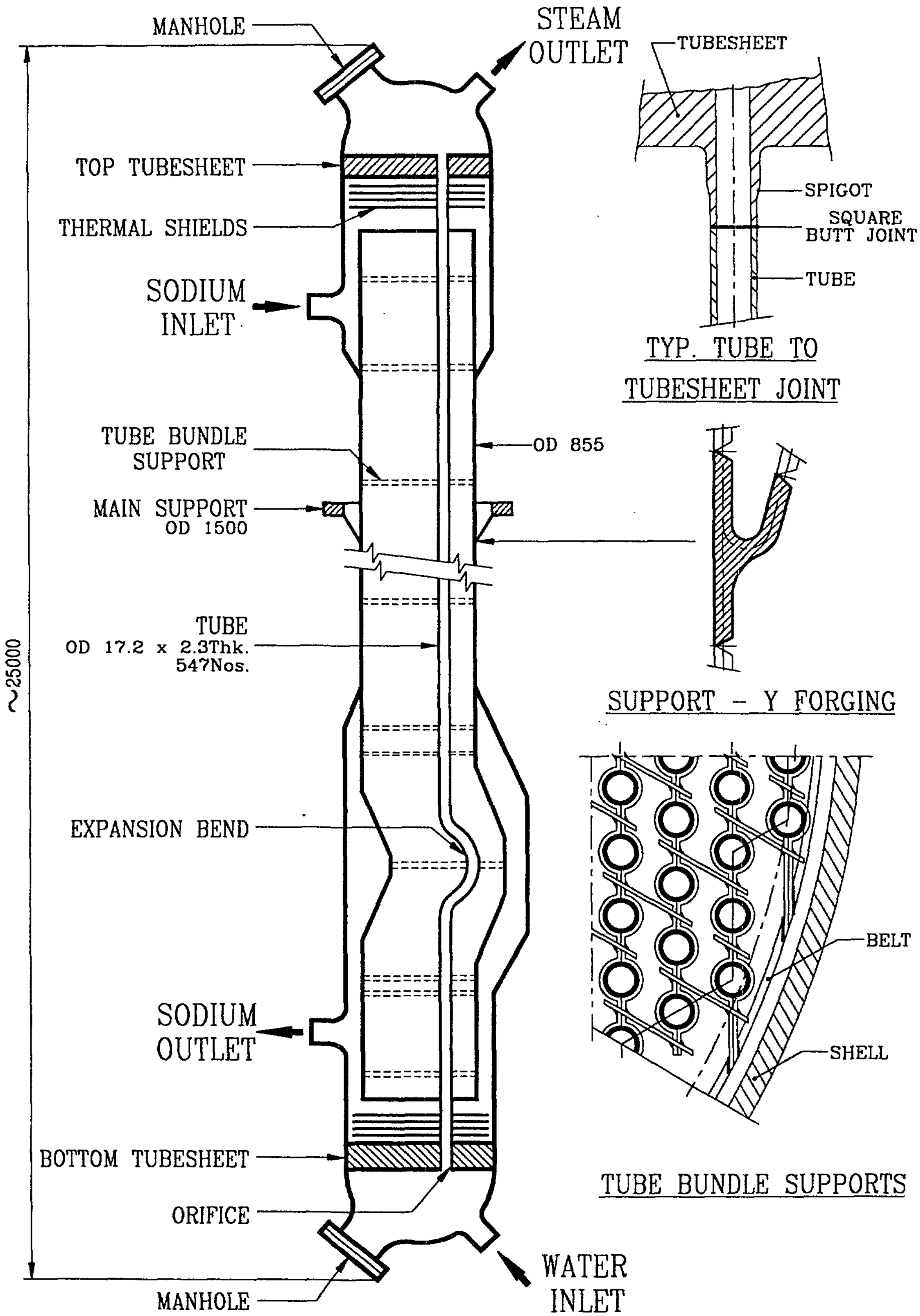
SECONDARY SODIUM PUMP

FIG. 5



INTERMEDIATE HEAT EXCHANGER

FIG. 6



STEAM GENERATOR

FIG. 7

envisaged to postweld heat treat individual joints to avoid risk of stress corrosion cracking associated with hard welds in Cr-Mo steel. Top & bottom tubesheets are protected by thermal shields. Sodium inlet and outlet shell junctions are in the form of pullouts. Manhole is provided on water-steam dished heads to permit access for in-service inspection of tubes and to carry out tube plugging, if required. The design basis accident for SG takes into account the effect of large leaks. Actuation of both the rupture discs located at the inlet and outlet of SG governs the number of failed tubes for leak analysis. Design basis leak is taken as instantaneous double ended guillotine rupture of 3 tubes at the top location of the SG. The reaction products are discharged to the secondary sodium storage tank.

Thermal striping has been the cause of some of the sodium leaks in auxiliary circuits. Mixing devices has been provided to ensure that sodium streams mix with temperature differences lower than the established safe limits.

Oil systems have been avoided wherever possible to minimise the risk of oil fires. The quick actuating valves on the water steam side and sodium side are pneumatic driven. Cooling is by nitrogen for primary cold trap and by air for secondary cold trap. For austenitic stainless steel piping, leak before break concept is used where leak monitoring provision exists. To minimise sodium fires, all the sodium pipes within the reactor containment building are double walled. The dump valves are duplicated in sodium circuits to enhance reliability of dumping, in case of a sodium leak.

It is envisaged to have one primary pump, one secondary pump and a SG as spare.

Operation with one secondary loop at maximum power of 50 % is also planned in case of non availability of one loop.

4.5 Core components handling

In-vessel handling is carried out using two rotatable plugs and an offset (fixed) arm type fuel handling machine (IVTM). An ultrasonic scanner is provided in order to check projection of any SA/absorber rods above the top of the core before starting in-vessel transfer operation. Additionally, strict administrative control on interlocks is to be provided. An Inclined fuel transfer machine (IFTM) is used to transfer the subassemblies from the main vessel to outside.

Ex-vessel sodium storage for removal of decay heat of SA has been avoided and the SA are stored in in-vessel storage locations within the main vessel. The spent fuel subassemblies are stored inside the main vessel for a period of 8 months till the decay power reduces to less than 5 kW and are then shifted to spent fuel storage bay (SFSB). SFSB is a water filled double concrete walled tank.

4.6 Decay heat removal

In case off-site power is available, the decay heat is removed through normal heat transport path of secondary sodium and water/steam circuits. Additionally, an independent safety grade passive direct reactor cooling system consisting of 4 independent circuits of 6 MWt nominal capacity each has been provided. Each of these circuits comprises of one sodium to sodium heat exchanger dipped in reactor hot pool, one sodium to air heat exchanger, associated piping and tanks. Except for the dampers provided on the air side, this system is entirely passive. A slope of 3.5 % is provided for the finned tubes of sodium-air heat exchanger in decay heat removal circuit in order to avoid gas locking.

4.7 Instrumentation and Control (I & C)

I & C, though not having significant impact on capital cost, needs detailed consideration in design as it demands considerable efforts in execution and it has a strong bearing on reactor availability. 2 loop design selected helps in reduction of sodium process instrumentation. The list of trip parameters is based on analysis. In principle, reactor should be shutdown under all design basis events using two independent trip parameters. Reactor shutdown is based on Lowering of rods (LOR) or by SCRAM. Two chromel-alumel thermocouples are provided at the outlet of each fuel SA and are used for SCRAM. Global Delayed neutron detectors (DND) and gaseous fission product detection are used for detection of failed fuel. Only global DND is used for SCRAM. 3 Failed fuel identification modules (FFIM) are provided for locating the SA with failed fuel pins. A bypass electromagnetic flowmeter is provided at the outlet of each primary pump discharge and the flow signal is used for SCRAM.

4.8 In-Service Inspection (ISI)

In-service inspection and monitoring is based on the requirements of ASME section XI, Div 3. For the main vessel, in addition to ASME requirements of continuous monitoring, ultrasonic examination is planned to be carried out through the main vessel - safety vessel interspace (300 mm nominal gap). A periscope is provided for visual examination of reactor internals. Eddy current inspection is under development for the SG tubes. SG tube size and expansion bend design takes into account this inspection requirement. Ultrasonic examination is planned for the dissimilar joints of the roof slab - main vessel and SG transition joint. The subject of ISI for other reactor components important to safety is under study. For the safety related reactor assembly components, which are non-inspectable, an additional factor of safety in design is envisaged.

4.9 Reactor Containment Building

Though the whole core accident is categorised as BDBE, a containment is provided based on the design condition of mechanical energy release of 100 MJ in case of core melt down accident. It has been checked that the main vessel and top shield can withstand this accident. The amount of sodium that is ejected into the containment building does not exceed 1000 kg and preliminary analysis indicate a pressure rise of ~ 10 kPa resulting from the sodium spray fire inside the containment. Aircraft crash is not a design basis event for the containment as the site selected meets the screening distance value of the regulatory code.

4.10 Radiation Protection

The siting, design of the plant and the operating procedures are intended to ensure that the radiation exposure to plant personnel and to the public resulting from the plant operation are controlled so as to comply with the dose limits prescribed by Atomic Energy Regulatory Board (AERB). Adequate shielding is provided wherever required to meet the pescribed dose limits. The targeted collective dose for the plant is 0.5 person-Sv/a (50 man-rem/a). For the general public, the exposure is limited to 0.1 mSv/a ($1/10^{th}$ of admissible dose is apportioned for PFBR).

5.0 SUMMARY

Systematic efforts have been made to take care of the operating experience from LMFBR into the design of 500 MWe PFBR.

FBTR operating experience has improved the confidence level in the design and operation of core, sodium systems, control rod drive mechanisms, fuel handling machines, steam water system and SG leak detection system.

Well proven mixed oxide fuel is chosen as the reference fuel. Pool type concept has been adopted. The plant operating temperatures have been arrived at based on detailed structural analysis of the hot leg components and result in reduced unit energy cost. SS 304 LN / 316 LN is used for sodium systems while modified 9 Cr - 1 Mo is used for SG. 2 loop concept with 2 Primary pumps, 4 IHX and 4 SG per loop has been selected to reduce capital costs and to improve capacity factor. Reactivity incidents have occurred in some reactors and in spite of the best efforts, fully validated explanation has not been possible. Argon gas seal for IHX-Inner vessel sealing has been eliminated and a seal design with piston rings has been selected. Improvement of thermal hydraulics of IHX and provision of a mixing device at secondary outlet have been made. Steam generator design selected takes into consideration the lessons learnt from other operating steam generators and it is expected to realise a more reliable SG.

The design features selected for PFBR are expected to yield an economic, safe and reliable design with improved capacity factor.

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SAFETY DESIGN ANALYSES OF KOREA ADVANCED LIQUID METAL REACTOR

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Abstract

The national long-term R&D program updated in 1997 requires Korea Atomic Energy Research Institute(KAERI) to complete by the year 2006 the basic design of Korea Advanced Liquid Metal Reactor(KALIMER), along with supporting R&D work, with the capability of resolving the issue of spent fuel storage as well as with significantly enhanced safety KALIMER is a 150 MWe pool-type sodium cooled prototype reactor that uses metallic fuel The conceptual design is currently under way to establish a self consistent design meeting a set of the major safety design requirements for accident prevention Some of current emphasis include those for inherent and passive means of negative reactivity insertion and decay heat removal, high shutdown reliability, prevention of and protection from sodium chemical reaction, and high seismic margin, among others All of these requirements affect the reactor design significantly and involve supporting R&D programs of substance This paper summarizes some of the results of engineering and design analyses performed for the safety of KALIMER

1. Introduction

As of the end of 1997, Korea's total nuclear capacity was more than 10 GWe, with 12 units in operation In addition, 8 units are currently under construction It is expected that the country's present nuclear capacity will be more than doubled by the year 2010, by which time nuclear generation will account for 40 % of total electric power production Nuclear generation currently stands at 35 % of the total The heavy dependence on nuclear energy raises the issue of spent nuclear fuel storage or disposal as well as that of utilization of uranium resources To date, more than 3,000 MTU of spent fuels have been stored in At-Reactor(AR) pools of the 12 operating nuclear power plants Taking only nuclear power plants currently in operation or under construction into account, the cumulative amount of spent fuels is estimated to reach up to about 26,000 MTU by 2030

From the viewpoint that liquid metal reactors(LMRs) have the potential of enhanced safety utilizing inherent safety characteristics and of resolving spent fuel storage problems through proliferation-resistant actinide recycling, LMRs appear to be the most promising nuclear power option of the future In this context, the KALIMER development program was launched as a national long-term R&D program in 1992 and has been carried out by Korea Atomic Energy Research Institute(KAERI) since then As such, the objective of the KALIMER Program was set to develop an inherently and ultimately safe environmentally friendly, proliferation-resistant and economically viable fast reactor concept

Up until July 1997 efforts had been concentrated on the development of basic sodium technologies and design methodologies unique to the LMR design and operating characteristics An initial design concept also was proposed through the feasibility study of a number of innovative design features as well as various proven design features As a result , KALIMER was defined to be a 150 Mwe pool-type sodium cooled prototype reactor that uses metallic fuels In 1997, the KALIMER program plan was updated to call for the completion of the basic design and supporting R&D work by 2006 An effort is being made to establish by early 2000, not only a self-consistent conceptual design of system configuration arrangement and key features satisfying design requirements, but more importantly computer codes and methods specific for KALIMER engineering and design analyses[1]

At early phase of the conceptual design, an emphasis has been made to come up with the self-consistent design meeting a set of the major safety design requirements to avoid "unusual occurrences", or arrest them One of the major requirements of current emphasis is that KALIMER shall be of inherent passive means of negative reactivity insertion and decay heat removal, sufficient to place the reactor system in a safe stable state for bounding ATWS events without significant damage to the core or reactor system structure Even with the inherent reactor shutdown requirement, the reactivity control and shutdown systems are required to result in extremely high shutdown reliability As for all the other sodium cooled reactors, the structures, systems, and components of KALIMER are to be designed and located to minimize the probability and consequences of sodium chemical reactions Seismic isolation is also required to achieve high seismic margins All these safety design requirements affect the design significantly and demand supporting R&D programs of substance

For the analysis of KALIMER's inherent safety, a plant-wide transient analysis code SSC-K is being developed Models for reactivity feedback effects and pool thermal-hydraulics have been developed into the code and a preliminary analysis of UTOP and ULOF/LOHS performance has been attempted Design alternatives have been investigated to improve decay heat removal capability by passive means, for which functional testings are to be done Seismic base isolation is shown to reduce seismic response of building and structures significantly and ,therefore, provides a great advantage in safety as well as economy for the structural design of nuclear power plants Substantial progress has been made in developing and validating the methodologies, and engineering analyses for the structural design of the KALIMER are under way An investment is also being made on the other key design features testing, such as electromagnetic pump , self-actuated shutdown system, and fuelling machine in reactor vessel Effort continues to be made on the development of basic sodium technologies such as measurement or detection technique as well as the investigation on thermal-hydraulic and chemical behavior Engineering and design analyses are also being made to improve IHTS configuration against sodium chemical reaction

In the following sections, the major design features of KALIMER are briefly described and some of results from the safety design analyses and supporting R&D programs are summarized

2. Major Design Features of KALIMER

Table 1 summanzes some of the major design parameters of KALIMER, which is currently under the conceptual design phase A salient feature of its key system designs is bnefly descnbed in the following[2]

Table 1 KALIMER Key Design Parameters

OVERALL		**PHTS**	
Net plant Power, Mwe	150	Reactor Core I/O Temp,°C	386 2 / 530 0
Core Power, MWt	392	Total PHTS Flow Rate, kg/s	2143 1
Gross Plant Efficiency, %	41 5	Pnmary Pump Type	electromagnetic
Net Plant Efficiency, %	38 2	Number of Pnmary Pumps	4
Reactor Pool Type			
Number of IHTS Loops	2		
Safety Shutdown Heat Removal	PSDRS		
Seismic Design	Seismic Isolation Bearing		
		IHTS	
CORE		IHX I/O temp , °C	339 7 / 511 0
Core Configuration	Radially Homogeneous	IHTS Total Flow Rate, kg/s	1803 6
Core Height, mm	1000	IHTS Pump Type	Electromagnetic
Axial Blanket Thickness, mm	0	Number of IHXs	4
Maximum Core Diameter, mm	3447	Number of SGs	2
Fuel Form	U-10% Zr Alloy		
Ennchments (IC/OC) for	14 4 / 20 0		
Equilibnum Core, %		**Steam System**	
Assembly Pitch, mm	161 2	Steam Flow Rate, kg/s	175 5
Fuel/Blanket Pins per Assembly	271 / 127	Steam Temperature , °C	483 2
Cladding Matenal	HT9	Steam Pressure, MPa	15 50
Refueling Interval, months	12		

Core and Fuel Assembly

The KALIMER core system is designed to generate 392 MWt of power The reference core utilizes a homogeneous core configuration in radial direction with two dnver fuel ennchment zones, surrounded by a layer of blanket assemblies The core layout, shown in Figure 1, consists of 96 dnver fuel assemblies, 42 radial blanket assemblies, 6 control rods, 1 ultimate shutdown system(USS) assembly self-actuated by a Cune point electromagnet,6 gas expansion modules(GEMs), 48 reflector assemblies, 54 B_4C shield assemblies, 72 shield assemblies, and 54 in-vessel storages(IVSs) in an annular configuration. The in-vessel storages(IVSs) are located between the stainless steel shielding zones There are no upper or lower axial blankets surrounding the core The reference core has an active core height of 100 cm and a radial equivalent diameter (including control rods) of 172 cm, the height-to-diameter ratio (H/D) for the active core becomes 0 581 The physically outermost core diameter of all assemblies is 344 7 cm The core structural matenal is HT9 It's low irradiation swelling charactenstics permits adequate nuclear performance in a physically small core The fuel pin is made of sealed HT-9 tubing containing metal fuel slug in columns The fuel is immersed in sodium for thermal bonding with the cladding A fission gas plenum is located above the fuel slug and sodium bond The bottom of each fuel pin is a solid rod end plug for axial shielding The dnver fuel, blanket fuel, reflector, and shield assemblies use identical structural components with only the bundle and its mounting gnd changing from one assembly type to the other The control assemblies use outer hardware (nosepiece, duct and handling socket) that is identical to that in the other assemblies. Reflector assemblies contain solid HT9 rods. The absorber assemblies use a sliding bundle and a dashpot

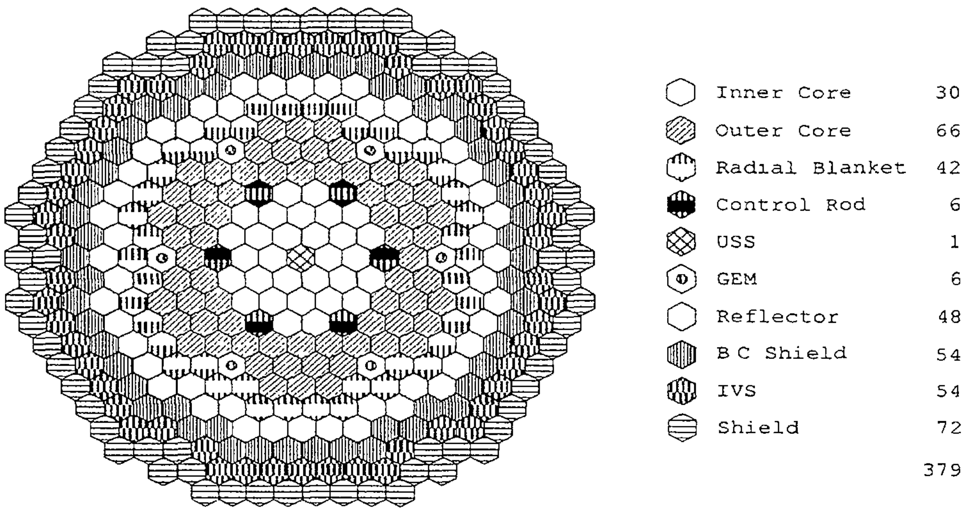


Fig.1. KALIMER Core Layout

assembly within the same outer assembly structure as the other assembly types. In all assemblies, the pins are in a triangular pitch array. The bottom end of each assembly is formed by the nosepiece which provides the lower restraint function and the coolant inlet[3].

Reactivity Control and Reactor Shutdown

Reactivity and power are controlled by means of the control rod system in the driver fuel region of the core. The control rod design satisfies both the one rod stuck condition and the unit control rod worth condition against the unprotected transient over-power(UTOP) event. The gas expansion modules(GEMs) are passive reactivity feedback assemblies that insert negative reactivity into the core during a loss of flow. The Self-Actuated Shutdown System(SASS) located at the center of the core is designed as an ultimate shutdown system by using a Curie point electromagnet which loses its magnetic force holding the shutoff rod when the temperature of the primary sodium reaches the curie point, hence a passive shutdown can be achieved.

Residual Heat Removal System

In KALIMER, the shutdown heat removal system is designed with the emphasis on system reliability to achieve a higher level of plant safety. Safety grade heat removal is achieved by the Passive Safety Decay Heat Removal System(PSDRS), which consists of the air path around the containment vessel and takes the decay heat from the reactor pool and discharges the heat to the atmosphere. Normally the decay heat is removed by steam generators and the condenser. During the maintenance of any IHTS, heat is removed by the remaining IHTS loop. Also there is the Steam Generator Auxiliary Cooling System(SGACS) to aid the decay heat removal. SGACS induces natural or forced circulation of atmospheric air past the shell side of steam generator Intensive analysis on the system performance and design parameters is under progress for system level design optimization

Reactor Structure

The reactor vessel has overall dimensions of 17 6m height, 7 02m diameter, and 5cm thickness in preliminary concept design and is composed of a cylindrical shell with an integral hemispherical shell bottom head The structural integrity and safety of the reactor vessel has been achieved by providing no penetration nozzle and no attachments other than the core support structure The shape of the core support structure is skirt-type All equipment like IHX, EM Pump IVTM, and UIS are supported by a reactor head and a rotating plug is adopted for the refueling operation The support barrel, which is a major component of reactor internal structures serves as a redan to separate the hot sodium pool and cold sodium pool and as a support of internal structures including the reactor core The containment vessel, which encloses the reactor vessel, is easy to access from the reactor vault so that the inspection and maintenance of the vessel can be easily accomplished General arrangements for NSSS and reactor building are tentatively developed as shown in Fig 2&3

The seismic base isolation for the rector building using high damping rubber bearings has been adopted to achieve sufficient structural integrity and economic design of KALIMER, when subjected to the design basis earthquake such as a horizontal Safe Shutdown Earthquake of 0 3g The development of a design concept adopting 3 dimensional seismic base isolation is under consideration to reduce both horizontal and vertical seismic responses

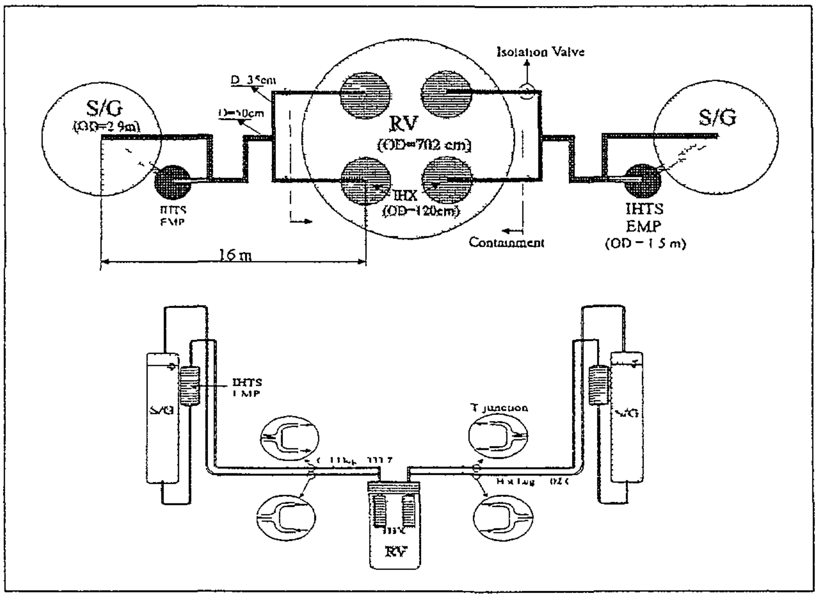


Fig 2 General Arrangements of NSSS

Heat Transport System

A superheat steam cycle is implemented to have a high plant efficiency noting that high thermal efficiency reduces the heat discharge from the plant, resulting in less impact to the environment IHTS

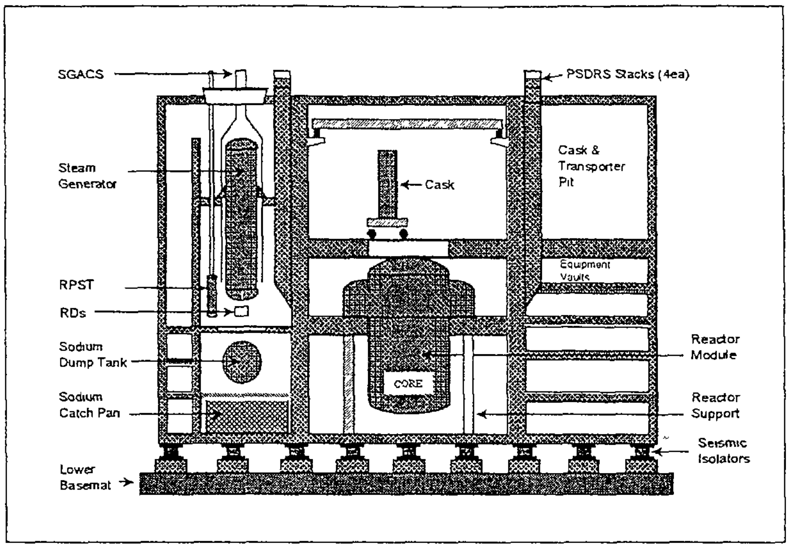


Fig. 3. General Arrangement of Reactor Building

consists of two loops and each loop is equipped with one steam generator unit to simplify the system design and increase the plant operation flexibility. For safety, large system thermal inertia is achieved by using a pool based primary system. Strong emphasis has been given to the prevention and mitigation of possible sodium-water reaction events for the IHTS piping routing. Valves for isolation of IHX from the sodium-water reaction products are installed at each IHTS piping penetrating the containment. The system reliability is improved by using electromagnetic (EM) pumps, which do not have moving parts, for both of the primary and intermediate coolant pumping. The low momentum inertia of the EM pump is compensated for by using an auxiliary device which keeps a certain amount of rotating kinetic energy when the EM pump runs normally but supplies electricity from the rotating kinetic energy to the EM pumps when the electricity supply to the pumps is interrupted. The operating temperature and component size were determined to make the net plant thermal efficiency higher than 38%. Preliminary analysis on economic effects was made in setting up the plant heat balance, as shown in Figure 4, for system design optimization.

3. Safety Design Analyses and Supporting R&D Programs

3.1 Inherent and Passive Safety Design Analyses

Inherent Safety Analysis

A plant-wide transient analysis code is being developed for the analysis of KALIMER's inherent safety and for the assistance in the development of design, where new design features will frequently demand not just new data but new models. Transient and safety analysis code SSC-K is under development based upon the SSC-L code which was developed by BNL for the analysis of loop type

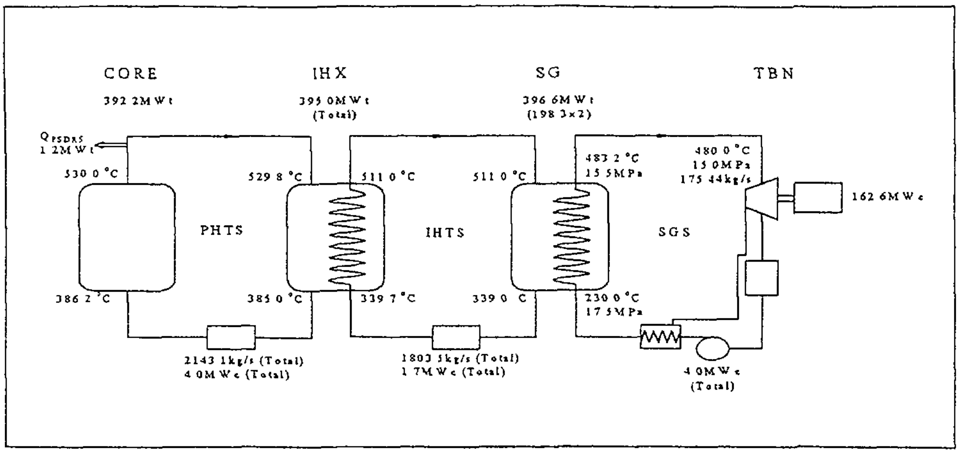


Fig 4 KALIMER Plant Heat Balance

LMRs with oxide-fueled core Models modified and newly developed into the code so far include models for reactivity feedback effects and pool thermal-hydraulics In order to verify the logic of the models developed, and to assess the effectiveness of the inherent safety features based upon the negative reactivity feedbacks in achieving the safety design objectives of passive safety, a preliminary analysis of UTOP and ULOF/LOHS performance has been attempted

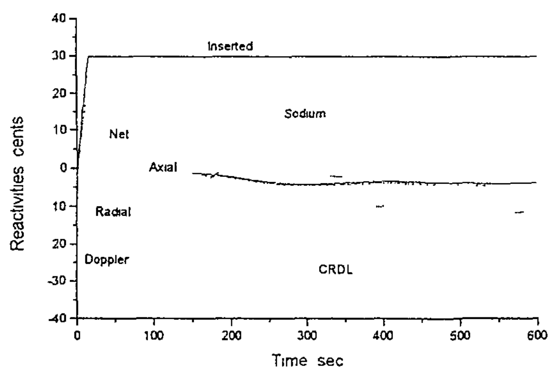


Fig 5 Reactivities during UTOP Event

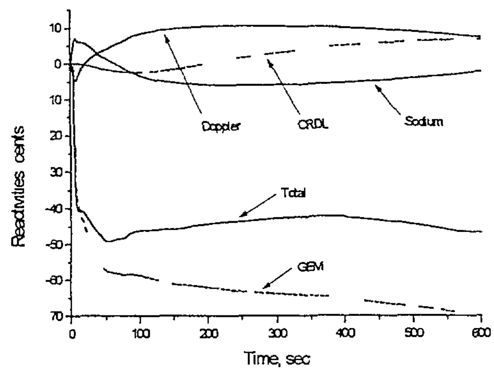


Fig 6 Reactivities during ULOF/LOHS Event

Inadvertent withdrawal of the control rod at reactivity insertion rate of 2 cents/second was assumed for the simulation of UTOP As expected, the reactor power reaches an asymptotic level higher than that of the initial steady state due to negative feedback effects As shown in Figure 5, the Doppler effect is an instantaneous and important feedback for UTOP and the net reactivity increases initially and then decreases to negative values due to feedback effects

Trip of all primary pumps with coastdown and the loss of IHX heat removal capability due to sodium water reaction in the steam generator is assumed for the ULOF combined with LOHS event Reduction of the core flow is due to the coastdown of primary electromagnetic pumps, and the reactor power decreases to about 6% of the rated power due to negative reactivities When there were no

GEMs in the core, there occurred a sodium boiling since the reactor power decreases rather slowly and power-to-flow ratio increases As shown in the Figure 6, the net reactivity is always negative during the course of the transient due mainly to the largest contribution from GEMs[4]

According to the preliminary evaluation of the inherent safety characteristics, there is a large safety margin even under severe unprotected event conditions In order to validate the SSC-K code for safety analysis, code-to-code comparison calculations and/or calculation against experimental data need to be performed Potential safety concerns of KALIMER need to be resolved as well Even though EBR-II experiments have shown the possibility of inherent safety of small metallic cores, there need to be an investigation in extending the result to larger cores Coastdown characteristics of electromagnetic pumps has a significant effect on the core safety under loss of flow events, and the performance of synchronous machine for inertia need to be evaluated Effect of the fluctuating sodium level inside GEM on reactivity and the effect of GEM reactivity insertion due to the restart of pumps at low power operation need also to be investigated

Probability of HCDA occurrence is extremely low due to inherent safety characteristics of KALIMER, and mechanistic analysis is not planned during the conceptual design stage However, depending upon the decision of the licensing authority, there may be a developmental effort for the mechanistic approach in the future A simple model, based upon Modified Bethe-Tait model, is being developed for the estimation of energy release and available work under HCDA for the analysis of ultimate safety

Passive Decay Heat Improvement Analysis

To increase the capacity of decay heat removal of a LMR system that uses a natural air circulation cooling, feasibility of heat transfer enhancement has been studied for a planar air channel by introducing a new channel configuration using radiation-convection structures of compact heat transfer surface For the new channel configuration, the heat transfer mechanism has been investigated and design guides for the radiation-convection structure have been developed based on the investigation results Following the developed design guides, a new radiation-convection structure has been also devised Analysis of the air channel cooling with the new radiation structures revealed substantial heat transfer enhancement and the feasibility of the heat transfer enhancement with the new channel configuration design has been confirmed[5]

In the operation of PSDRS, core decay heat is transferred to the containment vessel, as shown in Fig 7 and the heat from the containment vessel is dissipated to the air flow which is generated by the natural circulation from the density difference between the air channel and the environment The heat dissipation to the air flow is made of two paths One is the direct convection heat transfer from the containment wall surface and the other is an indirect path to the air In the indirect path, heat is first transported from the containment vessel surface by radiation to the air separator which separates the hot air from the incoming cold air Then the heat is dissipated to the air flow by convection

The main resistance in the heat transfer from the core to the air in a system like PSDRS is at the path from the containment vessel to the air [5]. The improvement of the heat removal capacity of the system comes to heavily depend on the improvement of the heat transfer in the air channel. Two types of works have been made to improve the heat removal capacity. One is the modification of the wall surface to enhance the convection heat transfer coefficient and the other is modification of the air channel configuration itself. In this study, a new channel configuration is introduced and the feasibility of the heat transfer enhancement of the new channel configuration is examined for black body surfaces. The new channel configuration is shown in Fig. 8 and is provided with lateral compact heat transfer surface structures. The new channel configuration is different from the conventional configurations in that it uses compact heat transfer surface and the surface is located across the channel. Since the configuration is different, the heat transfer mechanism becomes different from that of previous studies.

By introducing the lateral structures to the air channel, the radiation heat transfer from the containment vessel is redistributed. The high heat transfer performance of the radiation-convection structure (hereafter called as radiation structure) effectively dissipate the heat received by radiation to the air and the overall heat transfer capacity can be increased depending on the channel system design. For the geometry of the air channel of black surface and fixed wall temperature condition, the overall heat transfer is predicted to increase up to 6 times than the heat transfer rate of the same gap size , and up to about two times than the rate at the optimum gap size without radiation structures

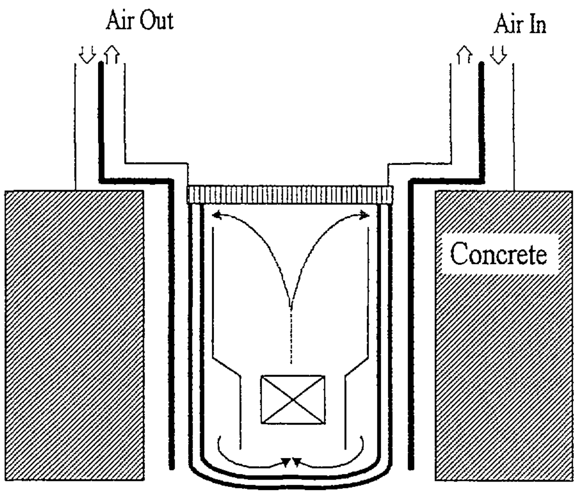


Fig. 7. Analysis domain & Heat Transfer Path Network

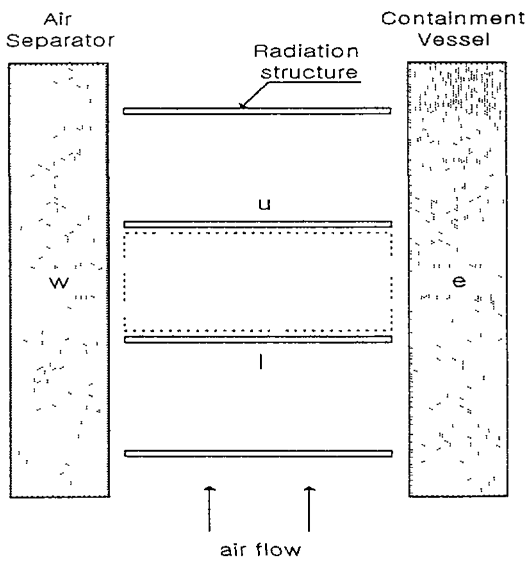


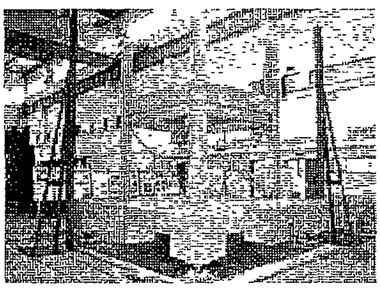
Fig.8. Radiation Structure

3.2 Seismic Isolation Study

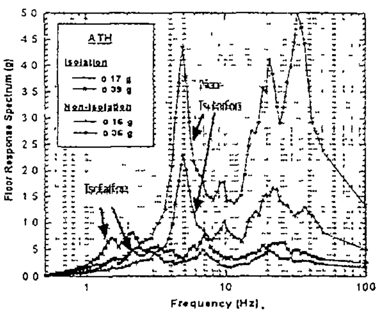
Some essential results of the seismic base isolation studies for KALIMER are summarized in this section.

LRB and Shake Table Test

For the rubber specimen and laminated rubber bearing(LRB) tests, various effects such as the shear strain, the loading rate, the cyclic loading, and so on are investigated. In these tests, the LRB being developed in KAERI shows good mechanical characteristics applicable to KALIMER. In the shaking table tests for the seismically isolated structure, it is confirmed that the seismically isolated structure produces significant reductions of the seismic responses compared with the case of non-isolated structure. Structural dynamic test for base isolated structure equipped with 2 dimensional 4-1/8 scale high damping rubber bearings were performed using 30 ton-6 dof shaker. Fig. 9 shows test model structure and seismic response results at upper slab of the test model to artificial time history input of SSE 0.3g[6].



Test Model



Test Results of Isolated and Non-isolated Test Model to Artificial Time History 0.3g & 0.15g

Fig.9. KALIMER Seismic Isolation Test Program

Reactor Building Analyses

To obtain the time history of the seismic responses of reactor building, a lumped-mass beam model is developed. The model is composed of two sticks; the one is for the reactor building and the other is for the reactor support structure. The time history responses for the non-isolated and isolated reactor buildings are calculated for an artificial time history earthquake generated by using the seismic design spectrum curve of US NRC RG1 60. Design basis earthquakes for KALIMER are SSE 0.3g for horizontal and 0.2g for vertical direction, and OBE 0.15g for horizontal and 0.1g for vertical direction respectively. The isolation frequency of reactor building is 0.5 Hz and the equivalent damping of LRB is 12%. The lumped-mass model of the reactor building is presented in Fig. 10

The total weight is about 68,000 tons. The 9 beam elements for the reactor building and the 3 beam elements for the reactor support structure are used. The maximum acceleration responses of the non-isolated and isolated reactor buildings for the horizontal and vertical earthquake data are shown in Table 2

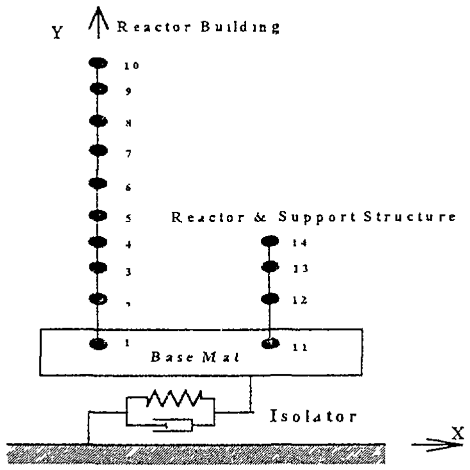


Fig 10 Lumped mass-beam models of KALIMER building

Table 2 Accerations and Displacements of Reactor Building Under ATH Earthquake

Location	X-Direction(g)		Y-Direction(g)		Z-Vertical (g)	
	Non-isolated	Isolated	Non-isolated	Isolated	Non-isolated	2D isolator
Base	0 30	0 175	0 30	0 177	0 205	0 321
Top	**1.461**	**0.177**	**1.609**	**0.179**	**0.577**	**0.848**
RV support	**0.583**	**0.173**	**0.676**	**0.175**	**0.362**	**0.558**

The time history responses for x-direction displacement are presented in Fig 11 and the response spectra at major locations represented in Fig 12

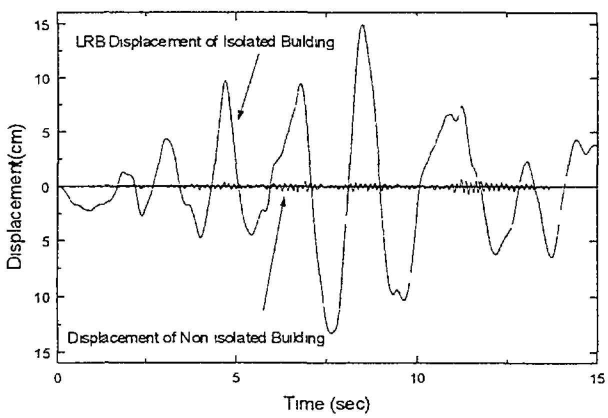


Fig 11 Displacement Responses of Reactor Building

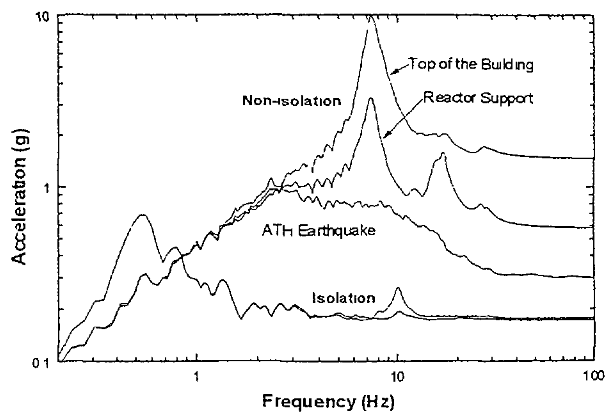


Fig 12 Comparison of Acceration Reponse Spectra of Reactor Building(ATH,X-dir 0 3g)

The maximum peak acceleration is reduced to 0 177g for isolated condition, while it is 1 46g for non-isolated condition The maximum displacement becomes larger to 15 0cm for the isolated condition The maximum acceleration for the vertical earthquake of 0 208g ZPA is amplified to 0 848g for isolated condition, while the maximum acceleration is amplified to 0 577g for non-isolated condition This agrees with the general trend that the horizontal isolation of structure can amplify the vertical responses[7]

Reactor internal structures and components

To produce the seismic analysis model for the reactor internal structures, the lumped-mass modeling technique is used From the 3-dimensional finite element model of KALIMER reactor internal structures, the detail local stiffness analyses are performed to construct the lumped-mass seismic analysis model The seismic analysis and evaluation of KALIMER are presented through the modal analysis, the seismic time history analysis, and the equivalent seismic stress analysis Table 3 shows the natural frequencies of the reactor structures resulted from the modal analysis for the seismic analysis model shown in Fig 13

Table 3 Results of Modal Analyses of KALIMER

Mode	Horizontal (Hz)		Vertical (Hz)	
	Isolation	Non-isolation	Isolation	Non-isolation
1	0 70	8 11	1 87	1 87
2	11 51	11 88	8 09	8 25
3	13 69	18 81	17 77	17 94
4	21 04	27 85	23 08	34 26
5	27 90	27 97	34 85	36 59
6	31 29	33 13	36 60	36 71
7	35 54	36 95	37 01	37 15
8	38 19	39 77	62 10	78 16
9	39 78	53 00	86 14	91 53
10	53 29	58 07	95 24	98 22

The seismic responses of reactor structures of seismically isolated KALIMER are significantly reduced for accelerations and relative displacements in horizontal direction For the isolation case, the maximum peak acceleration in horizontal direction is same in all structures and components, i e , 0 11g for OBE and 0 22g for SSE The responses are reduced about 14 times in IHX, 9 times in EMP and 8 times in reactor vessel liner, support barrel, and core compared with those in non-isolated case However, for the vertical direction, significant response amplifications occur in whole structures This is due to the vertical structural frequency of 8 1Hz located in dominant excitation frequency band of input motion

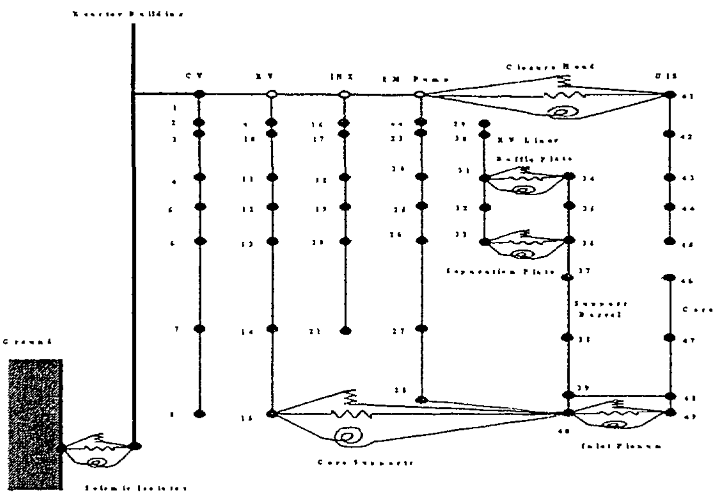


Fig.13. Seismic Analysis Model of Reactor Structures

Table 4 shows the results of the seismic margin evaluations and the seismic capacity of KALIMER reactor internal structures including the reactor vessel and containment vessel. From the results, the containment vessel, reactor vessel, inlet plenum, and core support have large seismic stress margins but the reactor vessel liner, support barrel, separation plate, and baffle plate have small margins. The maximum stress occurs in reactor vessel liner parts connected with the separation plate due to the vertical seismic loads.

Table 4. Accerations & Displacements of Reactor Building Under ATH Earthquake

Items	σ_{SSE} * (MPa)	P_{L+b}* (MPa)	Margins*	Minimum Seismic Capacity*
Containment Vessel	21.4	401.9	17.78	
Reactor Vessel	39.6	401.9	9.15	
RV Liner	340.0	401.9	*0.18*	0.354g
Support Barrel	113.0	382.4	*2.38*	
Inlet Plenum	20.8	401.9	18.32	
Separation Plate	188.0	401.9	*1.14*	
Baffle Plate	193.0	382.4	*0.98*	
Core Supports	72.1	401.9	4.57	

□□σ_{SSE} = Total stress intensity for horizontal and vertical SSE loads

* P_{L+b} = 1.5 x Min [2.4 S_m , 0.7 S_u], ASME Code Sec.III App.F.

* Margin = (P_{L+b} / σ_{SSE}) - 1

* Seismic Capacity = Min[Seismic Margin +1] x SSE

To evaluate the maximum seismic resistance in preliminary designed KALIMER reactor internal structures, the index of seismic capability(SC) is defined in this paper as follows:

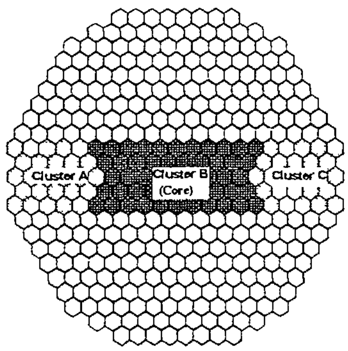
SC = Minimum [seismic stress margins +1] x SSE

Using above equation, the seismic capability of KALIMER is preliminary calculated as 0.354g. When the vertical stiffness of the support barrel/separation plate/reactor vessel liner region increases by the design change, this index value is expected to be significantly increased[8].

Core Seismic Response

The seismic analysis of LMR core structures is a complex problem involving the dynamic interaction of many hundreds of individual fuel, blanket, and shield assemblies in a sodium environment. To simplify the core seismic problem, the cluster modeling technique shown in Fig.14 for a diametral row of the core is used. The clusters of assemblies are assumed to have no relative motion between the assemblies within a cluster. The diametral row modeling approach gives conservative results and it is easier to evaluate the core seismic behavior compared to a full core model using the cluster technique.

In the present analyses, 3-clusters row model, in which cluster B represents fuel assemblies and clusters A and C represent the shield, blanket, reflector, and etc. as shown in Fig.14, is used to simplify the core seismic problem. The clusters A and C have 26 assemblies in each and cluster B has 51 assemblies. Fig.15 shows the core seismic model used in analysis.



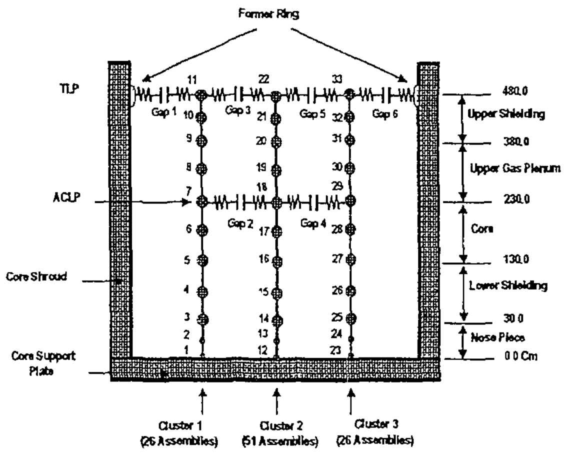


Fig.14. Clustering of LMR Core Assemblies *Fig.15. Simplified Core Seismic Analysis Model*

To investigate the dynamic characteristics of LMR core seismic analysis model shown in Fig.15, the modal analysis is carried out. To generate the linear model used in modal analysis, all the gap stiffness shown in Fig.4 are eliminated. The results of modal analysis show that the fundamental frequency of LMR core is 4.3 Hz and the second natural frequency is 24.3Hz. These natural frequencies of core will show non-linear behavior during impacts at load pads.

For the general investigation of core seismic responses, the harmonic excitations subjected to rigid core shroud and core support plate are used in the analyses considering conservative excitation conditions. Table 1 shows the input loading conditions.

Table 5 Results of Modal Analyses of KALIMER

	Core Support Excitation for SSE Conditions (0.3g)		
Load Case	Acc.	Freq.	Remarks
1	1.28g	8.1 Hz	Non-Iso, RI Freq.
2	1 28g	4.3 Hz	Non-Iso.Core Freq
3	0 22g	4.3 Hz	Iso., Core Freq.
4	0.22g	0.7 Hz	Isolation Freq.

The results of the core seismic response analyses show that . the load case 4, which is the case of a seismically isolated LMR gives significantly reduced seismic responses compared with those of the load cases 1 and 2, which are for the cases of non-isolated LMR. The seismic responses for the load case 3 , which may give the limit design case of the seismic isolation frequency for core, show little reduction in seismic responses(Fig.16). When the seismic isolation frequency(0.7Hz) is much lower than the core fundamental frequency(4.3Hz), a good isolation performance is observed in terms of core seismic responses. Fig.17 illustrates the impact load at the gap 3(TLP).

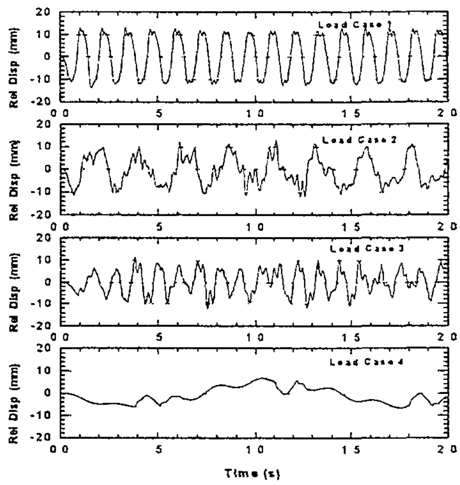


Fig.16 .Relative displacement at node 22(TLP)

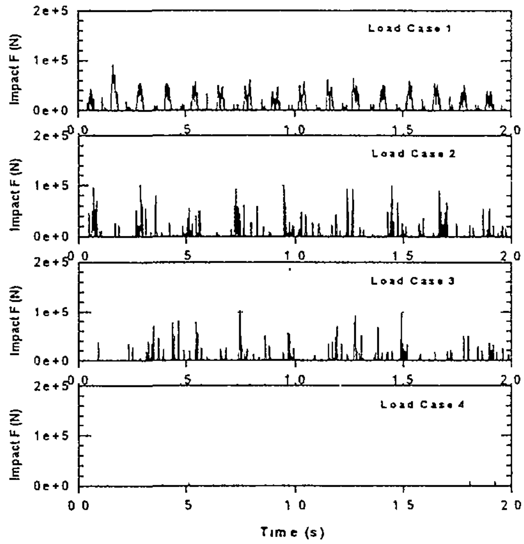


Fig.17. Impact Loads at Gap3 (TLP)

From these results, we can conclude that the seismic isolation provides great reductions of the impact loads as well as the number of contacts at the load pads of core assemblies at the former ring. This can allow the simple design of a core control system. And it is expected that the requirements of the core compaction and reactivity insertion problem can be easily satisfied when an efficient seismic isolation is adapted for the LMR design[9].

3.3 Sodium Technology Development

Sodium Water Reaction Analysis

Large scale water leakage into the sodium side due to the failure of tubes in LMR steam generators leads to an increase in the pressure and temperature by hydrogen and the heat of reaction, and may give significant effects on the structural integrity of the intermediate heat transport system(IHTS). Prior to designing IHTS and steam generator, a pre-estimate of the pressure effects for this system should be conducted As a general trend of pressure change, when water leakage occurs, a relatively high pressure is formed within milliseconds and is called the initial spike pressure After this peak pressure, a lower secondary pressure follows and decreases slowly because pressure change is not sensitive to time. This step is called the quasi-steady state. The intensity of the initial spike pressure depends on the internal structure of the steam generator and the transient characteristics of the sonic waves. The intensity of the pressure depends on the inertia constraints of the IHTS

A computer code, SPIKE, has been developed for analyzing the various characteristics of IHTS resulting from initial spike pressure. Briefly, the sodium flow in the IHTS is assumed to be a compressible, one dimensional, unsteady viscous flow. From these assumptions and equations of continuity, momentum and energy, the governing equations were developed.

A comparison of the calculated results using the SPIKE code with the experimental value is shown in Fig. 18. for an experimental IHX model of a 1/12.5 scale. The figure shows that the calculated results are consistent with the experimental values in the IHX inlet[10]. The code will be further verified by simulation experiments at KAERI's test facility, which is scaled down from KALIMER in the ratio of 1/256 (the heat load scale-down ratio, about 1/6 of the linear scale-down ratio). The SG-model has a diameter of 420mm (O.D.) and length of 2750mm with 5 layer helical coil tubes of which the total length is about 280m and the material is stainless steel 304 without welding. To assure the safety from accidents caused by large water leakages in KALIMER steam generators, studies on leak propagation, their simulation, and a pressure change estimation by computer codes have been carried out The computer codes, HOPRE and DIPRE , are being developed to analyze the quasi-steady-state pressure.

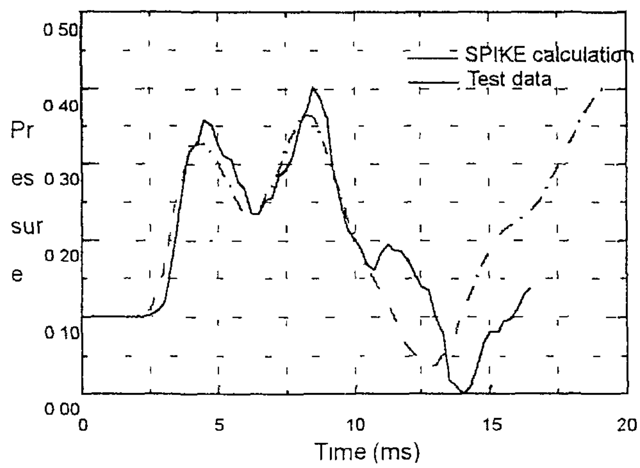


Fig.18. Comparison of pressure changes

The SPIKE code has been applied to investigate the pressure transients at various points of the IHTS of KALIMER. As shown in Figure 2, KALIMER is of two IHTS loops, each loop consisting of a steam generator, two intermediate heat exchangers, a pump and pipes which are connected with several fittings. The IHTS of KALIMER was modeled as a network having 40 branches and 39 junctions for the analyses. The results show that pressure transients or peak pressures are rather sensitive to such design parameters as leak rate, distance between the lower plenum of steam generator and rupture disc, and distance between steam generator and IHX. Figure 19 shows the pressure transients at various points of the IHTS with the rupture disk intact. It is noted that pressures tend to be monotonically increasing but heavily oscillating at some points. It was also observed that pressure transient behavior was quite sensitive to the size of sodium expansion tank[11].

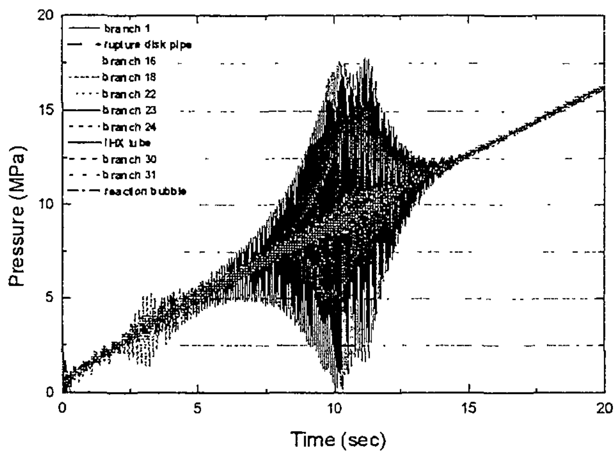


Fig.19.IHTS pressure transients

Sodium Thermal hydraulics and Component Development Testings

Small-scale sodium experiments have been performed to investigate the coolant thermal-hydraulic behavior, such as turbulent mixing in compact reactor space, flow reversal by natural circulation with

an electromagnetic pump operation, and decay heat removal by wall cooling, among others. Sodium experiments continue to be performed to develop the technologies to measure such parameters as differential pressure, local flow rate, and void fraction.

In addition to the safety evaluation analyses of the IHTS for large leaks, R&D work on sodium water reaction carried out to date includes small water leak experiments for the determination of a design base leak rate, development of reliable and real time detection system of water leaks using the acoustic signal as well as hydrogen detection, among others. Sodium fire characteristics and phenomena are also being investigated with $48m^3$ of rectangular type fire cell Analyses of various types of sodium fire phenomena, development of sodium leak detection system, and fire extinguishment, prevention and mitigation, aerosol filter and scrubbing devices will be carried out

In the area of key component development, the submersible-in-pool type electromagnetic pump of operating temperature of 600° C and 200 l/min maximum flow rate were developed using the theory of magneto-hydraulics and the equivalent circuit analysis and its prototype was manufactured and its operation tests were performed The SASS and IVTM were developed and their mockups are manufactured and their theoretical validation tests were performed

6. Conclusion

An effort has been made to establish by early 2000 the conceptual design of KALIMER with system configuration, arrangement and key features satisfying design requirements Emphasis is currently placed upon coming up with the design features meeting a set of the major safety design requirements for accident prevention, which include those for inherent and passive characteristics of negative reactivity insertion and decay heat removal, high shutdown reliability, high seismic margin, and prevention of sodium chemical reaction, among others

For the analysis of the KALIMER's inherent safety, a plant-wide transient analysis code SSC-K is being developed Models for reactivity feedback effects and pool thermal-hydraulics have been developed into the code and a preliminary analysis of UTOP and ULOF/LOHS performance has been attempted The results show that net reactivity stays negative during the transients analyzed Design alternatives have been investigated to improve decay heat removal capability by passive means, for which functional testings are to be done Seismic base isolation is shown to reduce seismic response of building and structures significantly and, therefore, provides a great advantage in safety for the structural design of nuclear power plants Engineering and design analyses are also being made to improve the IHTS configuration against sodium chemical reaction An investment is also being made on the other key design features testing, such as electromagnetic pump, self-actuated shutdown system, and fuelling machine in reactor vessel

Substantial progress has been made in developing and validating the methodologies, and engineering analyses for the conceptual design of KALIMER However, we still have a long way to go down the road to accomplish the mandate, that is, to complete the basic design of KALIMER as well as supporting R&D work

Acknowledgements

This work was performed under the 'Long-term Nuclear R&D Program' sponsored by the Ministry of Science and Technology of the Republic of Korea

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XA0056260

SEVERAL ACCIDENTS ABOUT ERHRS OF CEFR

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Abstract

An analysis about several unusual accidents about Emergency Residual Heat Removal System (ERHRS) of China Experiment Fast Reactor (CEFR) is present. CEFR is a pool-type sodium-cooled fast reactor. The ERHRS of this reactor is designed in passive principle, which enhance the interior reliability of CEFR. It is consist of two sets of independent channels. Each channel is comprised of decay heat exchanger (DHX), intermediate circuit, sodium-air heat exchanger (AHX) and related auxiliary system. Both DHX are located in the hot pool of the main vessel directly, which is used to cool the hot sodium. The whole set of ERHRS is completely passive except the ventilation valves of AHX. But, as a very important set of engineered safety features which is the final way to remove the heat from the reactor core, it is necessary to pay attention to all of possibilities that may reduce this ability. Several accidents are analyzed include the ventilation valves couldn't be opened, only one set of ERHRS could work and so on. The calculation results show that the ERHRS can keep the reactor in a safety status. Even though it is, experiments are necessary in the view of engineering yet.

1. Introduction

ERHRS of CEFR is consisted of two independent sets of channels and each channel can remove 0.525MW heat from the reactor core. The whole system is a passive system except the ventilation valves of sodium-air heat exchanger. Each channel of the ERHRS is mainly include

- An sodium-sodium decay heat exchanger (DHX),
- An sodium-air heat exchanger (AHX),
- Intermediate circuit and chimney,
- Auxiliary systems include impurity inspecting system, heating system, temperature detecting system, sodium fire prevention system, etc.

The design of ERHRS must be ensure that the temperature of main vessel and metal structure in the vessel couldn't exceed 560 centigrade degree in order to keep their strength. At the same time, these means have been adopt to enhance the stability and reliability.

- The ventilation valve of AHX consist of three inlet air doors and one outlet air door that could be driven either by electric motor or by manual,
- Emergency power supply,
- Two valves locate in the line to auxiliary system,
- All intermediate circuit pipe is covered with guard tube,
- Physical separation is designed for two channels.

Figure-1 gives us the profile of ERHRS. In the following cases, it is possible for main heat transfer system losing its function:

- Loss of off-site power supply;
- Loss of main feed water;
- Earthquake.

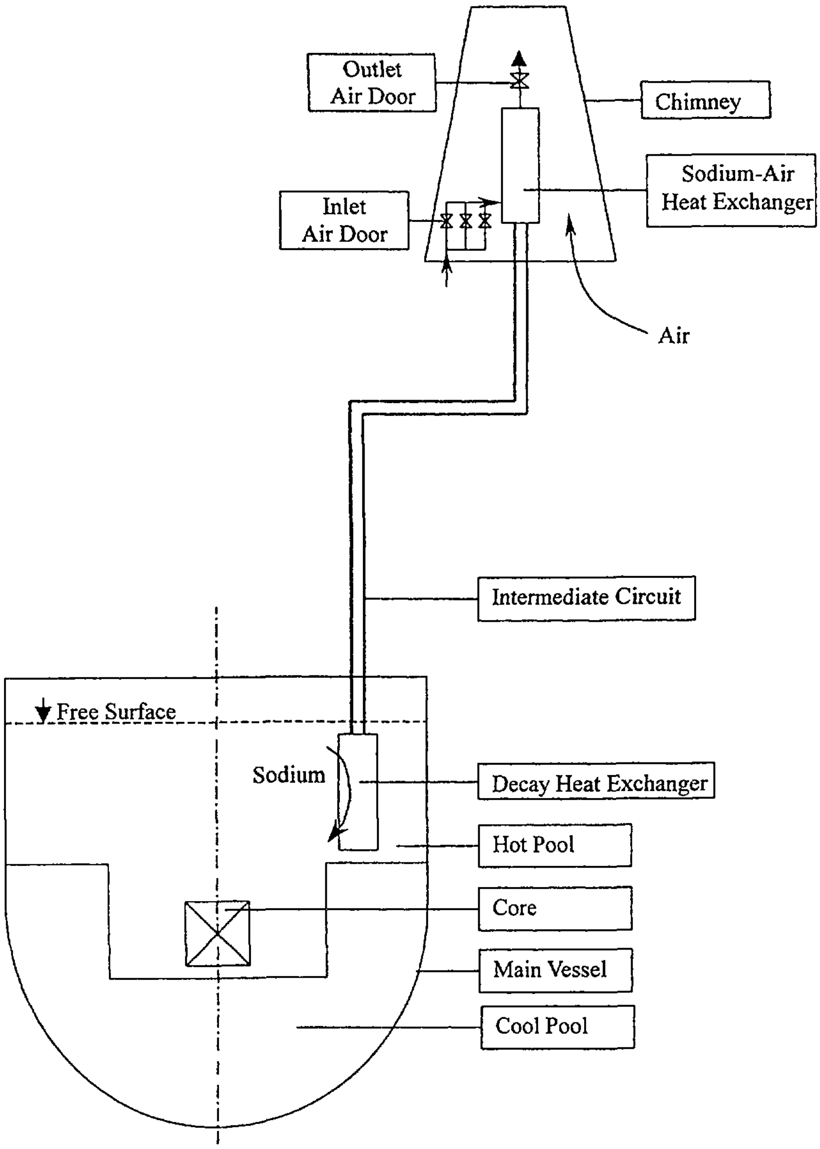

FIG. 1 Principle Diagram of ERHRS

When the main heat transfer system can not work normally, the ERHRS will automatically put into operation in ten minutes to remove reactor residual heat to ultimate heat sink (atmosphere).

2. Several Typical Accidents

For ERHRS of CEFR, there are some conditions that may cause the system loss partial or whole ability to removal residual heat. These occurrences are, 1) the natural convection is broken because pipe losing its integrality or losing partial or all coolant; 2) the pipes is blocked because of sodium's frozen; 3) the ventilation valves couldn't be opened. For the first case, the design has considered earthquake and all of the pipes are equipped with guard-tube that makes this type accident meet a very low probability. For the second case, there are few sodium-temperature detecting points and electric-heaters that prevent those accidents occurring, and, it is possible to adjust the ventilation-valve to decrease heat losing which could reduce the probability at the same time. For the last case, the failure probability is lager than the forth two cases because the ventilation valves need power to drive though it is divided into several sections and every section is consist of multi-vanes. Although it is, this condition must be analyzed in the base of safety considering and safety-guide requiring. We are either interesting in the condition that only one loop could work and other unusual occurrences except those mentioned above.

2.1 Both ventilation valves could not be opened

CEFR is pool-type reactor and the main vessel is divided into two parts. The upper part is called hot pool because it contains hot sodium whose temperature is 516 centigrade degree. The lower part is called cool pool because it contains cool sodium whose temperature is 360 centigrade degree. This accident is a BDBA (Beyond Design Basis Accident) in the Primary Safe Analysis Report of CEFR. Here are initial conditions,

- The reactor operated under full power (65MW),
- ERHRS is standing by.

In order to calculate within a long period, this accident must be analyzed in detail in the first hundreds second. 3D method is used for taking into consideration the complex rapid flow changing. Figure 4 shows us the sodium temperature field and flow scheme in the main vessel at 1 second, 81 second and 618 second respectively. These figures show that the cool sodium from could cool reactor core and partial residual heat is stored in the sodium.

The curves of sodium temperature in the core and mass flow of primary loop are given in Figure-2 and Figure-3 respectively. At the beginning of this accident, the coolant-temperature falls first because of the primary pumps' coast-down more slowly than the reactor power's decreasing. After that, the coolant temperature increase in the reason of flow lower than residual-heat and natural convection begins to establish. Figure 3 shows that the coolant flow changes very slowly after 100

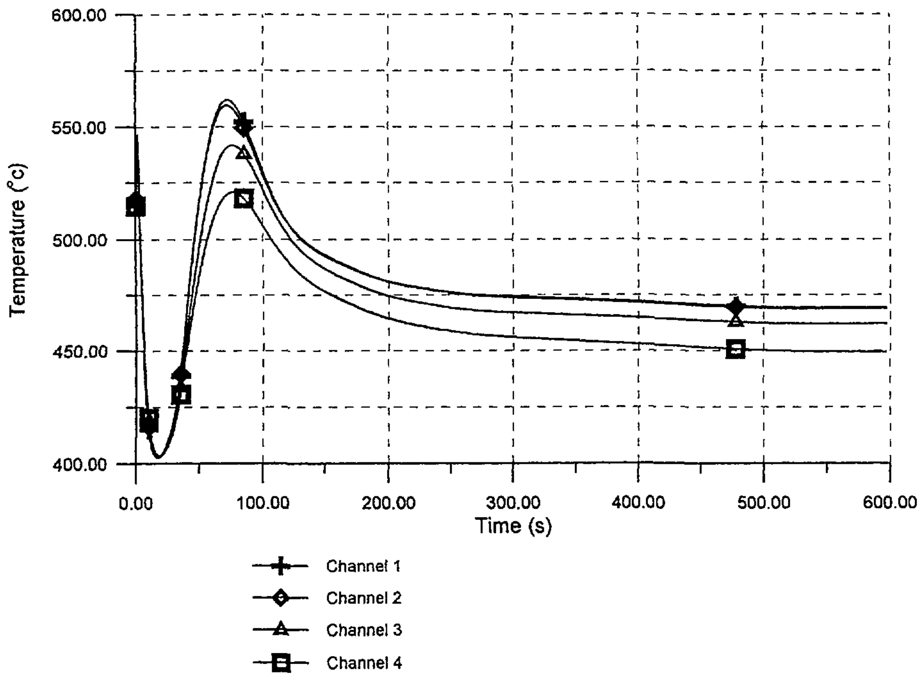

FIG. 2 Core Outlet Temperature

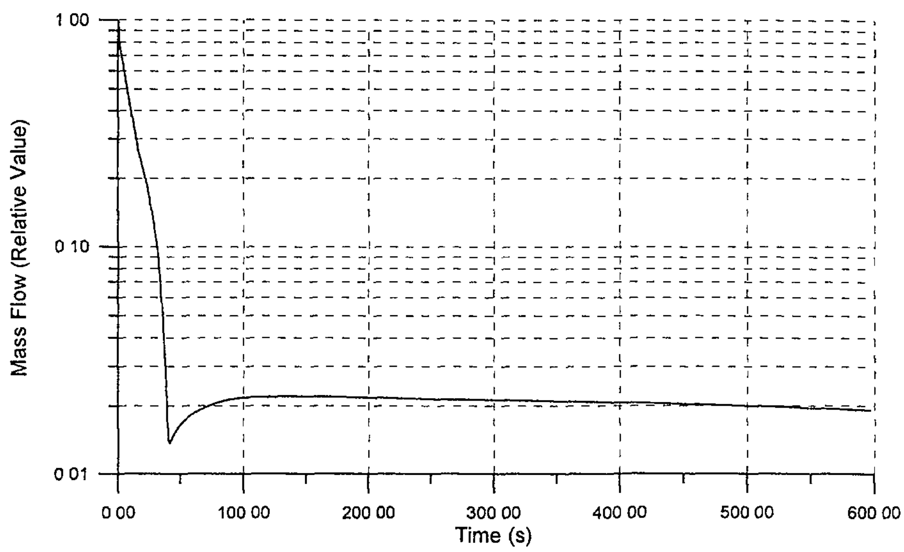

FIG. 3 Primary Sodium Flow

second, which means the natural convection has established basically. During this period, we should pay attention to the clad temperature because the maximum value will appear in this period. Fortunately, the maximum temperature is 563 centigrade degree, which is lower than the permitting line (850 centigrade degree).

1D method is adept to analyze the quasi-steady state. Figures 5 to 8 have given the calculation results on a long period. It can be seen that the sodium temperature outlet from the reactor core reach to 547 centigrade degree at 19.13 hour. On the other hand, the sodium temperature, bounded with main vessel, continue decrease for the reason of heat loss through steel vessel wall. And the temperature of wall approximately equal to this temperature. It can be summarized that the reactor structure temperature is lower than the permit value of 560 centigrade degree. The heat removed by ERHRS changes slightly after the accident occurring. Otherwise, it is better to pay attention to the heat losing through secondary loop, which makes a great contribution to the mass flow rate of primary loop. In another way, it can decrease the maximum temperature at the exit of core.

2.2 One loop failure absolutely

It's necessary to analyze this accident in the view of studying the reliability of ERHRS except the accident mentioned above. Figure 9 to 12 give us the calculation results. It can be seen that the sodium temperature in the hot pool is lower than the first accident. The reason is that AHX power jumps from 10% to 100% very soon after the ventilation valves being opened. Another more severe case is calculated too which the air door could not be opened at the same time. The results are good either if the vanes could be destroyed after 40 minutes.

2.3 Ventilation valves opened abnormally

The nominal power of CEFR is 65.MW and the sodium mass flow rate through the reactor core is 300kg/s. If the ERHRS is put into using unexpected, heat losing through ERHRS is only 1.62

percent of full power. The difference of sodium temperature at the entrance of IHX (Intermediate Heat Exchanger) is less than 3 centigrade degrees. On the other hand, it will spend 3 minutes at least to decrease 1 centigrade degree of the hot pool sodium temperature. Time is enough to terminate this accident with various means.

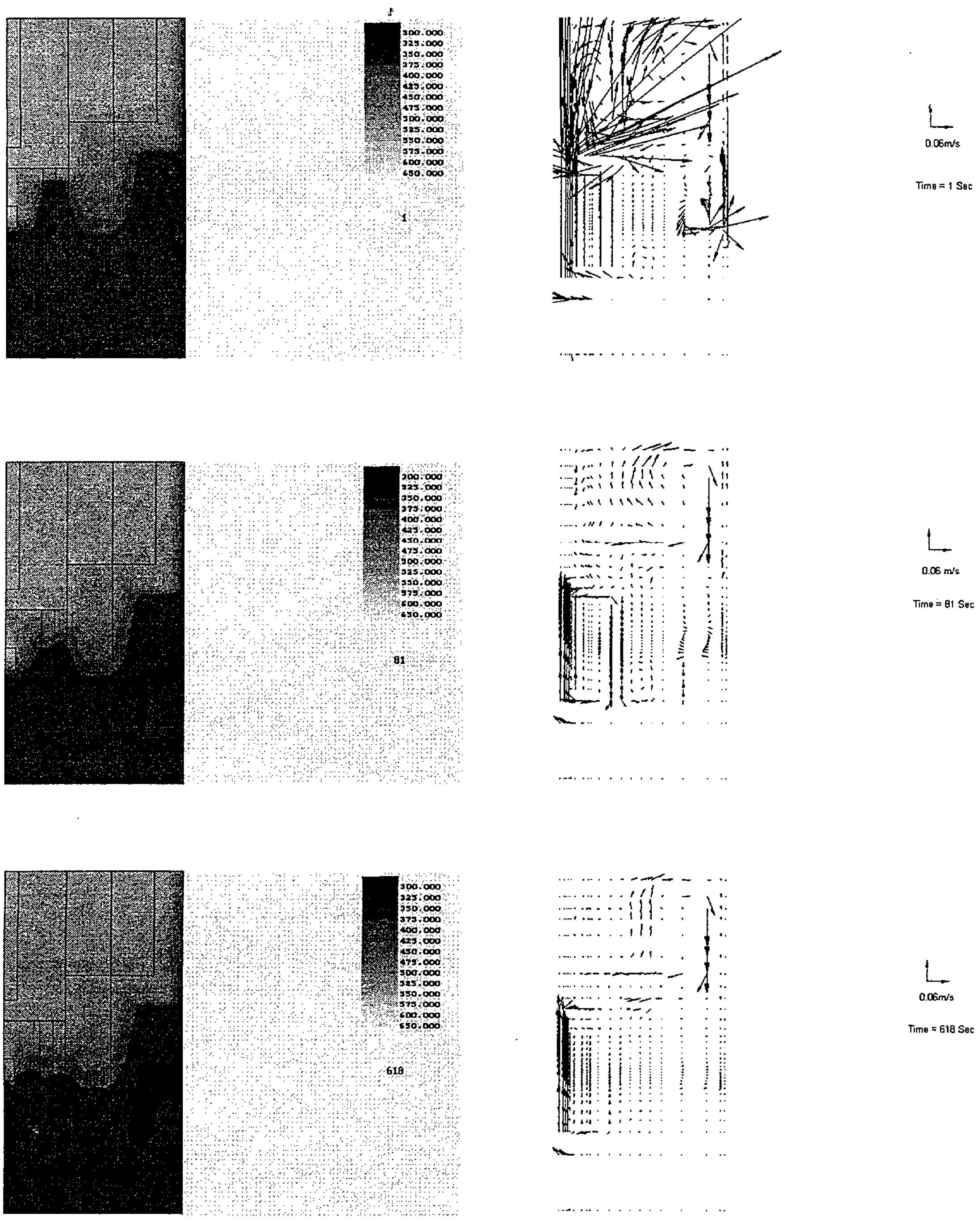

FIG. 4 Temperature and Flow Field of Primary Sodium in Reactor Vessel

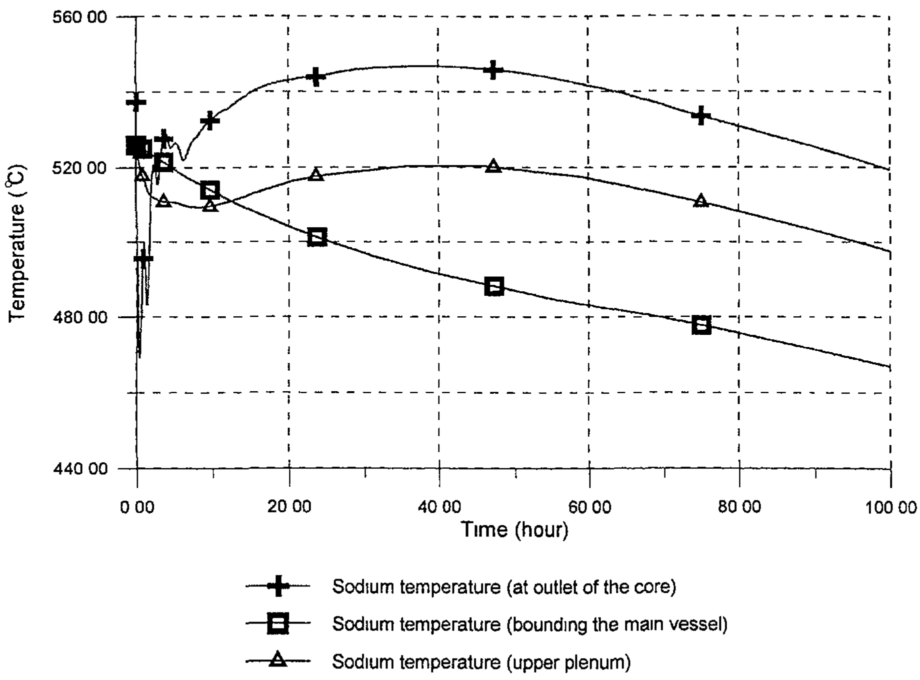

FIG 5 Sodium Temperature in Hot-Pool

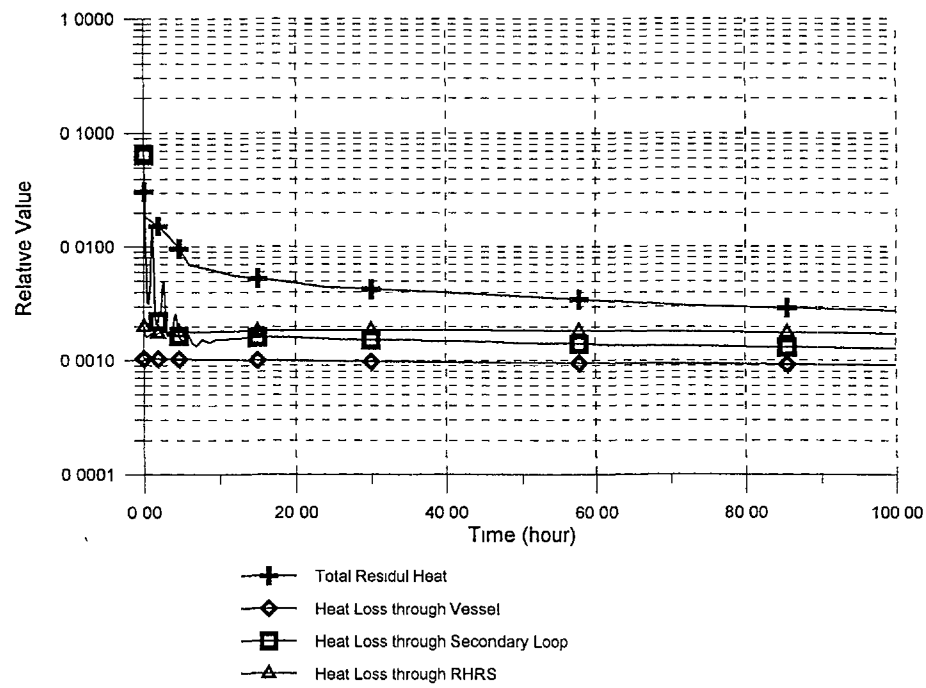

FIG. 6 Heat Removal Portions

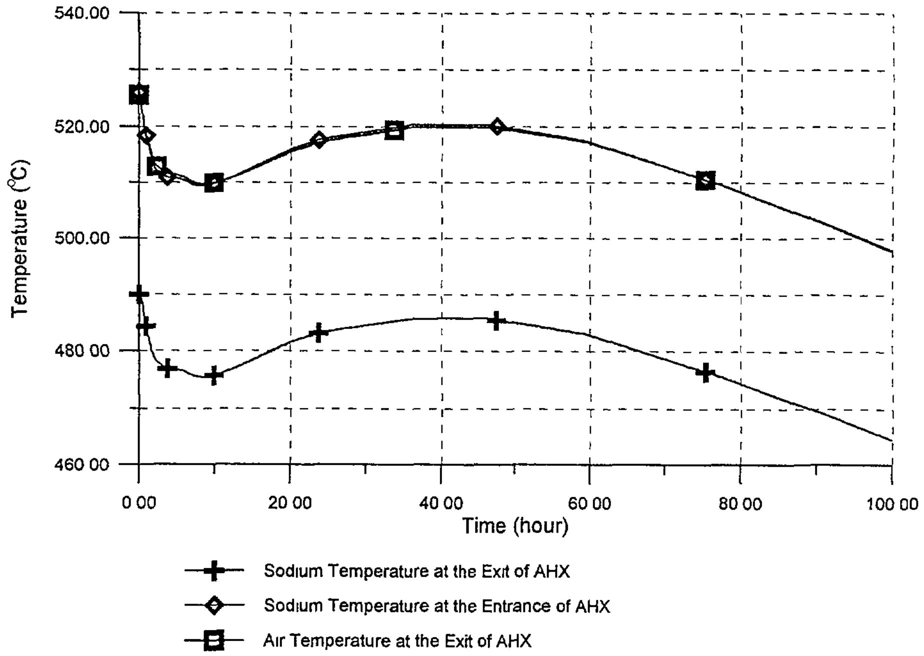

FIG. 7 Temperature Variation of AHX

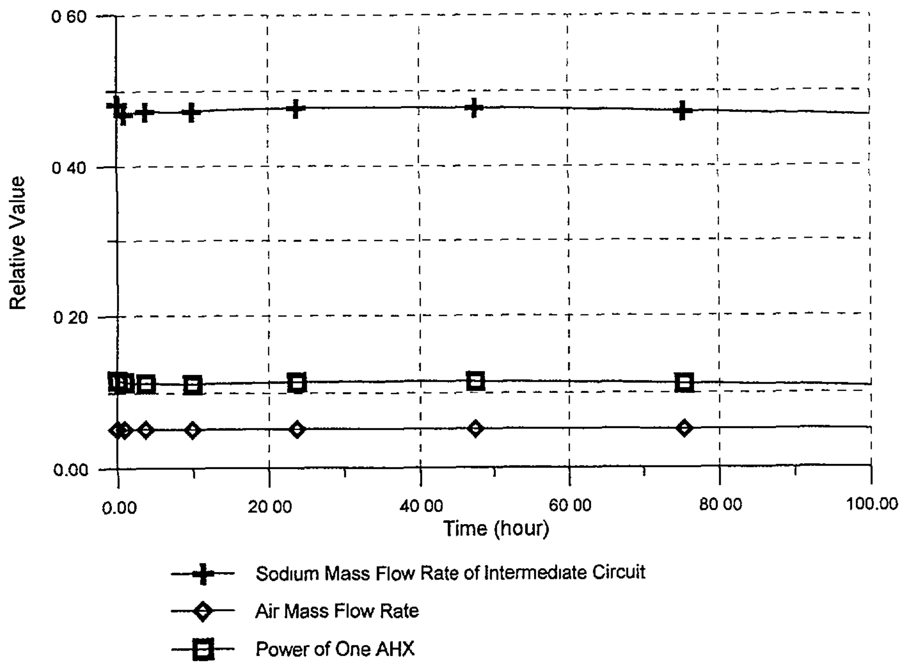

FIG. 8 Flow and Power Variation of AHX

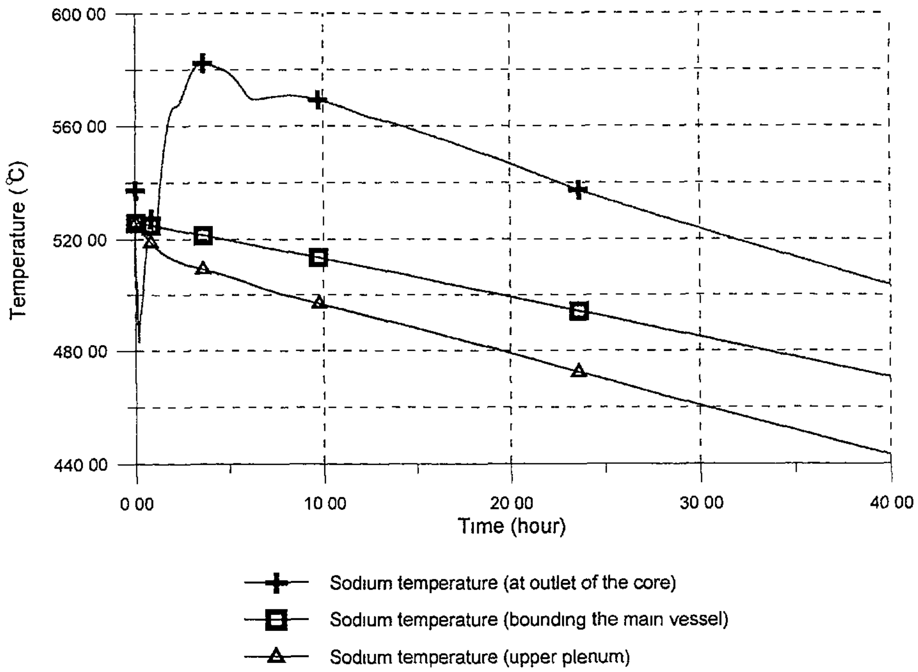

FIG. 9 Sodium Temperature in Reactor Vessel

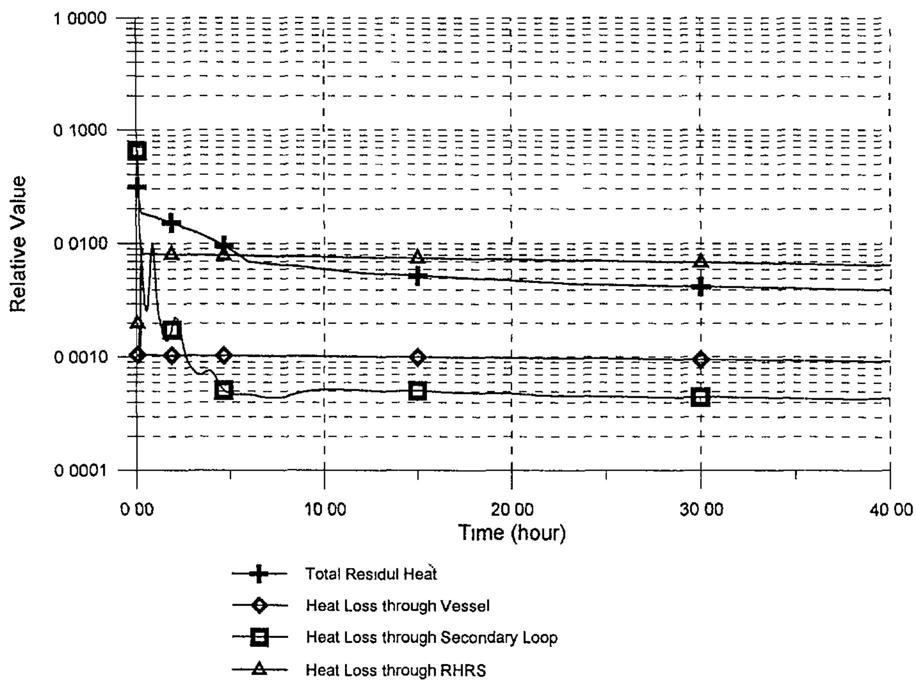

FIG. 10 Heat Removal Portions

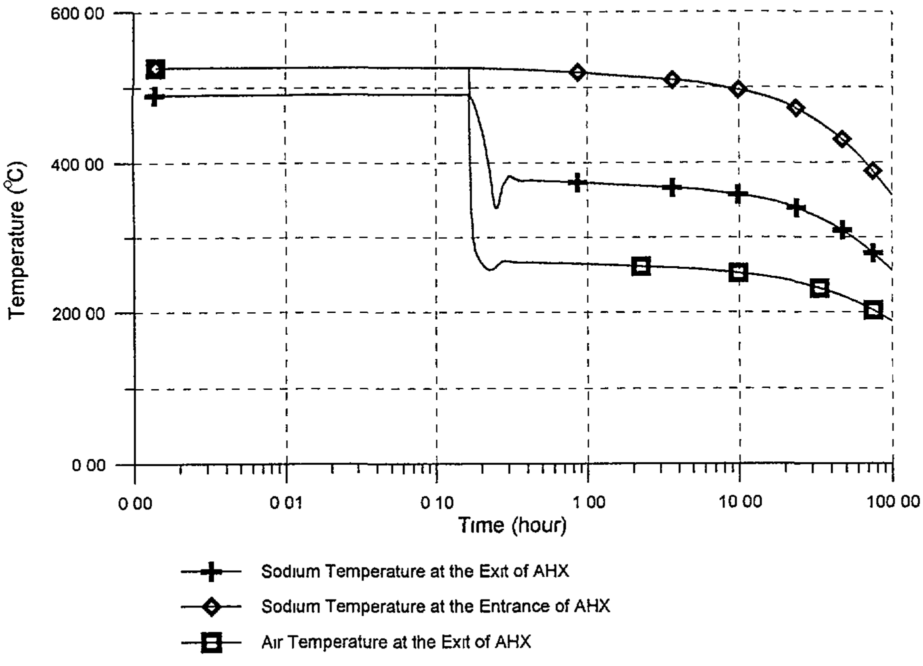

FIG. 11 Temperature Variation of AHX

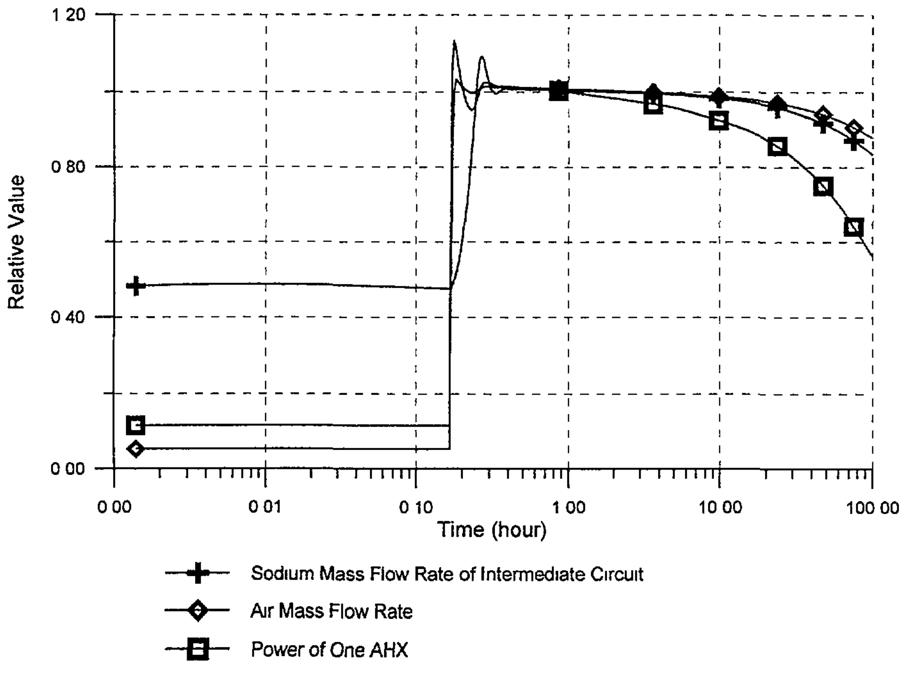

FIG. 12 Flow and Power Variation of AHX

3. Conclusion

As mentioned above, the ERHRS is designed to ensure safety and reliability, therefore, few measures are considered in design. They are:

(1) Inlet air door of AHX is divided into three sections,

(2) Ventilation valves could be driven by normal power supply, emergency power supply or manual operation,

(3) All intermediate circuit pipes are covered by guard tube,

(4) Physical separation is designed for two subsystems.

Calculations show that in all considered ERHRS accidents, including BDBA, primary sodium temperature is always within safety range.

LIQUID METAL FAST REACTOR TRANSIENT DESIGN

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Abstract

An examination has been made of how the currently available computing capabilities could be used to reduce Liquid Metal Fast Reactor design, manufacturing, and construction cost. While the examination focused on computer analyses some other promising means to reduce costs were also examined.

Introduction

A major problem with commercialization of Liquid Metal Cooled Fast Reactors has been high design and manufacturing costs. Operation with coolant and metal structures temperatures in the creep range, coupled with sodium's high thermal conductivity necessitate use of codes, standards, requirements, and design approaches that have been very costly.

While it has been accurately stated that the cost of high temperature design on FFTF and Clinch River was small compared to the cost of the components, this design cost was still very high by any measure. Liquid metal reactor components and systems designers must perform extensive strain limited structural analyses, address considerations associated with phenomena such as creep fatigue interactions, and provide for the design to accommodate relatively large movements and flexibility. An additional expense was associated with the effort related to code interpretations and code cases. Considerable effort was involved in the use of simplified methods (screening procedures) to pinpoint problem areas and minimize the more elaborate analyses of critical structural configurations required to show code compliance. This made it possible to resolve design and analysis problems, but restricted choices system and component design.

Much larger costs than the obvious costs directly associated with high temperature design were associated with the impact of this required methodology on the resulting designs and design requirements for systems and components.

Design transients and operating transients

The duty cycle (design) transients are events selected, by the designer, as representative of operating conditions that have been determined may occur during plant operation and that are sufficiently severe or frequent to be of possible significance to component behavior. These events are classified as normal events, upset events, emergency events, and faulted events. Attachment 1 provides a definition of these categories of events and a short discussion of the severity of damage and significance of the event categories.

These transients are not normally expected to represent actual plant operations, but are meant to be used for predictions of systems response to the events and for component stress analyses. The systems analyses include thermal/hydraulic analyses of the coolant, neutronic analyses of core behavior, coupled thermal/hydraulic-neutronic analyses to allow consideration of the effects of interactions, and analyses using coolant temperatures to predict metal temperatures, temperature rates of change and the like. In addition, it was necessary to build scale models on do testing (e.g., hydraulic, mixing, vibration) to obtain information for both direct use and for use in dimemsionless analyses. Simplified analyses were used to idealize complex geometries, loading histories, and material models. These screening analyses were used mainly to identify highly stressed and critical areas to guide design choices.

Attachment 2 provides a listing of the design transients for the Clinch River Breeder Reactor Plant. These events and frequencies provided are from Revision 118, the last Revision before the project was terminated. These design transient events and frequencies were used to order essentially all of the components for Clinch River. Most of the design work was completed. All major components were either delivered or well into manufacturing. The plant had essentially completed the licensing process, with all major issues having been resolved. Revision 118 is a mature listing of transients that had been used in the real world to design a loop type LMFR.

These demonstrably conservative transients were then grouped into umbrella transients and applied to various components and locations in components. This grouping of transients resulted in the frequencies of some relatively minor transients being summed with the most severe transient in the group. Thus the selected umbrella design transients are normally very conservative both as to severity and number.

The actual operational transients experienced should be considerably less severe than the design transients and have a significantly lower frequency of occurrence. The use of the actual operating transient history, the loading conditions accompanying these events, and the frequency can be used to reevaluate to design, for purposes such as life extension. These analyses should demonstrate that the achievable operating life of the plant is much greater than the originally specified design life.

Past practices

Experience showed that minor differences of opinion on analysis procedures concerning the ASME Code and code cases exaggerated difficulties and had considerable impact on increasing cost and restricting design options. The restrictions resulted from the inability to perform the large number of complex analyses required for many options due to costs, lack of adequate computing power, and related factors. The result was the need to select approaches that avoided, or minimized, uncertainties and to rely on the simplified, screening analyses, with the attendant restrictions in design options.

Differences in design practices have in the past handicapped developments that could have reduced costs. For example, some difficulties with past design practices are evident in examining application of the ASME Code and Code Case N-47 to pool type LMFRs, such as Super Phenix. The design of Super Phenix resulted in considerably thinner components. This required different buckling rules. Super Phenix design creep effects were negligible during normal operation, requiring different treatment from loop type reactors during transients into the creep range, such as resulted during emergency and faulted conditions. Because of these and other such problems the French developed a set of LMFR rules to address shortcomings of the ASME Code and apply available design and construction experience.

Needs to address problem areas in high temperature structural analyses were studied by the Working Group, Codes and Standards (WGCS) of the Commission of the European Communities in 1979. The WGCS examined and promoted efforts on benchmark calculations, constitutive equations, fracture mechanics, and seismic analyses.

Work to resolve these and other questions has been substantially reduced in recent years. Very few of these problems that rely on obtaining long term materials data have progressed. Some work on alternate materials to austenitic stainless steels has produced results.

Approach and limitations

An examination was made of the work that was done for Clinch River design using these transients. FFTF design work on reactor components was also considered. This effort focused on consideration of what would be done different at the present using currently available computer technology. The evaluation was qualitative, since there was no reasonable way to determine exact savings in design, analysis, and manufacturing costs.

There was one key technology, other than computer technology considered. This was the possible use of 9 Cr - 1 Mo, a ferritic/martensitic alloy. 9 Cr - 1 Mo promises to be an excellent substitute for austenitic stainless steels and 2-1/4 Cr 1 Mo steel. This material could be used in the entire system, thus eliminating dissimilar metal transition joints. It offers resistance to irradiation induced metal swelling and creep, helium embrittlement, and would allow higher design margins for ratchetting and creep-fatigue in steam generator applications.

In the past considerable effort has been devoted to finding and using "short cuts" to avoid the necessity of what were considered to be impracticably complex analyses.

The steps in the design process were examined. Areas where currently available computing power and analytic capabilities offered promise for cost reductions were identified. The potential reductions in manufacturing costs by increases in design flexibility were identified. Areas where no potential improvements could be identified at this time where identified.

Design areas and potential cost reductions

Systems design requires the interaction of a number of different disciplines with different interests. Examples are instrumentation and control, component design, licensing, and operation people. Design transient events must be realistic from the stand point of the instrumentation and control systems. They must be achievable by the component designers. They must be acceptable to the nuclear regulatory authority. And last, but not least they must be acceptable to the operations people. For this reason the operations people were involved from the start. Preparation of operating procedures was done in parallel with design analyses.

Due to the cost of the extensive analyses required, it has been the practice of designers to "lump" transients together as "umbrella" transients. Thus, instead of having a duty cycle requiring thousands of analyses of events that may occur only once or at most a few times, there is a duty cycle with a small number of events. The price paid is that the resulting "demonstratively conservative duty cycle results in the need for extremely expensive design solutions.

This duty cycle creates challenges that have been met by designers of liquid metal reactors that operate at high temperatures, such as the Fast Flux Test Facility and the Clinch River Breeder Reactor Plant. However, this was done at great cost for the analyses, extremely high manufacturing costs, and has created a disconnect between design transients and the real world.

Examination of the traditional design process, screening analyses, selection of the duty cycle, umbrella transients, unresolved differences in national codes and standards, inability to quantify the safety margins involved in use of codes and standards, and actions that overlay conservative actions on conservative actions shows that this process, while demonstrated to be highly effective, is also extremely expensive. The extreme conservatism involved in this design process is not visible, with the result that the public cannot see or understand the level of design conservatism and safety that results. Worst yet, many designers also do not understand. This results in apparently conservative actions that may do nothing to improve the design.

A new design approach should be possible based upon current computing and information technology. Such an approach could result in significant design simplifications and reduce the cost of manufacturing and construction.

It is now appears possible for designers of liquid metal fast reactors to achieve the same results at considerably lower cost. The historical practice of "umbrella" transients and the associated simplified duty cycle is no longer necessary. The current computing capability of easily affordable PCs and work stations now makes it possible to analyze the many transients that must be considered in the design and develop more economical structural and systems design solutions. Future Liquid Metal Reactors can take advantage of computer technology to significantly lower costs.

To determine the systems response to transients it is necessary to prepare a model of the entire reactor system. This can be done at several levels of complexity, with areas such as the core or the steam generators being represented in considerable detail, or relatively simple. The transient model can readily be a simulator. The use of "masks" to allow ease of modeling has been demonstrated in Nuclear Plant Analyzers developed for various nuclear power plants including VVERs in various countries. The use of the available spectrum of modeling complexity allows the systems designer to work with the various specialized disciplines to obtain a detailed understanding of the response of various systems to various design options. Areas of special concern can then be modeled in greater detail, or more detailed simulations run on the more promising design options.

It should not be necessary to develop, early on, a restrictive duty cycle to be applied as a design specification. The potential for ease of modeling can allow optimization of the transients that would be characteristic of various design approaches. Specific structural responses to these transients can be determined and changes made to mitigate problems.

This would entail a much more elaborate effort in the conceptual design phase that would reduce design uncertainties in the later phases.

Rather than work to achieve a simplified duty cycle and simplified umbrella transients, more exacting and realistic analyses could be performed. The currently available computing power allows these multiple analyses to be performed, facilitating optimization and removing the necessity for very conservative approaches to be taken to minimize and simplify calculations and analyses. At the same time the actual safety margins can be much better understood and more transparent, thus resolving one of the past difficulties between US and European design requirements.

By building on past operating and modeling experience it should be possible to use computer modeling and thus reduce the requirement for, or extent of, model or component testing.

The use of advanced computer modeling should allow reduced costs of fabrication, inspection and surveillance. In the longer term, it should be possible to develop uniform international code approaches for LMFR design that are less restrictive and allow improvements in plant availability and reliability.

There are some areas where it may not be currently practical to realize improvements using computer technology. There remains a need for improved materials data. However, it is possible that considerations of current computer analysis capabilities might result in changes in requirements for material properties information. For example, in some locations thermal striping causes a very large number of repetitive thermal transients on the metal surface being washed by sodium of fluctuating temperatures. It has been necessary to design for an infinite number of cycles. This requires the use of materials such as inconel, with associated high material and manufacturing cost.

An area where savings may be possible with improved materials data is on evaluation of welded joints. The material properties of the complete weld joint, as opposed to the weld metal alone, are needed. Ductility under multiaxial loading plays a key role in the structural adequacy of weldments subject to cyclic loading.

The US Liquid Metal Fast Reactor program compiled Nuclear Systems Materials Handbook. There is a vast amount of published information concerning structural design and evaluation of nuclear power plant components and high temperature structural design. Of late the quantity of such publication has greatly diminished. There is and will be, however, significant information coming from activities such as the Phenix Life Extension Project, from ongoing development activities, from applicable non nuclear work, and potentially from other activities, such as at Dounreay.

A related point is the need for an internationally available, and maintained, nuclear materials data handbook. The existence of the vast amount of potentially usable information, coupled with the extreme difficulty in searching for and locating this information is totally incompatible with current information technology. For example, while it is easy to find much information on many subjects on the Internet, it is virtually impossible to find any information on topics related to high temperature structural design and materials data. This data is published in had hoc reports, in a number of technical journals, and in various reports that are not generally available. As people who know where the information is leave the field these documents become increasingly hard to find and use. The result is that much valuable information, obtained at great expense, is being lost. Action to correct this situation would be of significant benefit.

There has been, and continues to be, significant High Temperature Structural Design technology developments for complex, critical non-nuclear structures that are subjected to elevated temperatures during normal operation. For example, such activities have been conducted in many nations to support aerospace development programs, such as those related to engines.

Because these developments are focused on intended applications in other fields and not problems of LMFR design there are significant differences in design lives, service conditions, materials, manufacturing practices, etc. The types of structures differ. The impact of these differences on such design information as constitutive models, material failure modes and models, and structural failure modes and consequences are sometimes difficult to assess. However, computer modeling, structural analyses methods, and analytic methods to understand materials behavior have advanced greatly in some of these non nuclear areas.

It is obvious that application of these developments in non nuclear areas is not a trivial undertaking. In spite of the obvious difficulties adapting these developments to LMFRs offers considerable promise and should be aggressively pursued. The reductions in the overall level of LMFR development activities, with the attendant reduction in work specifically directed at LMFRs makes such effort doubly attractive.

Attachment 1 **Definitions of categories of events**

Normal: Normal operation includes steady power operations and those departures from steady operation which are expected frequently or regularly in the course of plant operations, refueling, maintenance, or maneuvering of the plant. These events are to cause no damage. No damage is defined as those that:

1) result in no significant loss of effective fuel life;
2) are accommodated within the fuel and plant operating margins without requiring manual or automatic protective actions; and
3) result in no planned release of radioactivity.

Upset Any abnormal incident not causing a forced outage or causing a forced outage for which corrective action does not include any repair or mechanical damage. These off-normal conditions can cause anticipated conditions which individually may be expected to occur once or more during the plant lifetime. These operational incidents are occurrences that:

1) result in no reduction in effective fuel lifetime below the design values;
2) can be accommodated with, at most, a reactor trip that assures the plant will be capable of returning to operation after corrective action to clear the trip cause; and/or
3) Result in plant radioactivity releases that may approach the 10CFR20- guidelines.

Emergency Infrequent incidents requiring shutdown for correction of the condition or repair of damage in the system. There is no loss of structural integrity. These include unlikely off-normal conditions which individually are not expected to occur during the plant lifetime. However, when integrated over all plant components, events in this category may be expected to occur a number of times. These may result in minor incidents, that is an occurrence which results in:

1) a general reduction in the fuel burnup capability, and at most, a small fraction of fuel rod cladding failures:
2) sufficient plant of fuel rod damage that could preclude resumption of operation for a considerable period of time; and/or
3) plant radioactive releases that may exceed 10CFR20 guidelines, but does not result in interruption or restrictions of public use of areas beyond the exclusion boundary.

Faulted Postulated event and consequences where integrity and operability may be impaired to the extent that considerations of public health and safety are involved. These are off-normal conditions of such extremely low probability that no events in this category are expected to occur during the plant lifetime, but which represent extreme or limiting cases of failures which are identified as design bases. These are major incidents which can result in:

1) substantial fuel and/or cladding melting or distortion in individual fuel rods, but the configuration remains coolable;

2) plant damage that may preclude resumption of plant operations, but that will not cause loss of safety function necessary to cope with the occurrence; and/or

3) radioactivity release that may exceed the 10CFR 20 guidelines, but are well within the 10CFR100 guidelines.

Attachment 2

Clinch River Breeder Reactor Plant
Design Transient Events and Frequencies
(From Overall Plant Design Description, OPPD 10, Revision 118)

The **Event** is a short description of events to be considered in the plant structural design (see Attachment 1).

The **Frequencies** are the maximum number of occurrences of each event expected during the life of the plant. *These frequencies are used as the basis for plant structural design.*

A. Normal Events and *Frequencies*

N-1 Dry system heat up and cool down, sodium fill and drain loop for an entire system
5 total system + 8 per loop + an additional 17 per intermediate loop, exclusive of the Intermediate Heat Exchanger.

N-2a	Startup from refueling	*140*
N-2b	Startup from hot standby	*700*
N-3a	Shutdown to refueling	*60*
N-3b	Shutdown to hot standby	*210*
N-4a	Loading and unloading	*9300 (each)*
N-4b	Load fluctuations	*46~500 (each, up and down)*
N-5	Step load changes of +10% of full load	*750 (each)*
N-6	Steady state temperature fluctuations	*30×10^6*
N-7	Steady state flow induced vibrations	*10^{10} (Sodium)*

B. Upset Events and *Frequencies*

Note: The total frequency for U-1 events is associated with normal decay heat so as to balance the trips associated with partial decay heat for events U-2 through U-23.

U-1a	Reactor trip from full power with normal decay heat	*180*
U- lb	Reactor trip from full power with minimum decay heat	*0*
U- lc	Reactor trip from partial power with minimum decay heat	*0*
U- 2a	Uncontrolled rod insertion	*10*
U-2b	Uncontrolled rod withdrawal from 100% power	*10*
U-2c	Uncontrolled rod withdrawal from startup with automatic trip	*17*
U-2d	Uncontrolled rod withdrawal from startup to trip point with delayed manual trip	*3*
U-2e	Plant loading at max. rod withdrawal rate	*10*
U-2f	Reactor startup with excessive step power change	*50*

Note: These events are part of the startups specified for event N-2b and should not be added as separate startups.

U-3a	Partial loss of primary pump	*2 per loop*
U-3b	Loss of power to one primary pump	*5 per loop*
U-4a	Partial loss of one intermediate pump	*2 per loop*
U-4b	Loss of power to one intermediate pump	*5 per loop*
U-5a	Loss of AC power to one feedwater pump motor	*10*
U-5b	Loss of feedwater flow to all steam generators	*5*
U-6	(Deleted)	
U-7a	Primary pump speed increase	*5*
U-7b	Intermediate pump speed increase	*5*
U-8	Primary pump pony motor failure	*5 per pump*
U-9	Intermediate pump pony motor failure	*5 per pump*
U-l0a	Evaporator module inlet isolation valve closure	*4 per loop*
U-10b	Superheater module inlet isolation valve closure	*2 per loop*
U-l0c	(Deleted)	
U-l0d	Superheater module outlet isolation valve closure	*2 per loop*
U-11a	Water side isolation and dump of both evaporators and the superheater	*6 per loop*
U-11b	Water side isolation and dump of evaporator module	*6 per loop*
U-11c	Steam side isolation and dump of superheater	*3 per loop*
U-12	Loss of feedwater flow to one steam generator loop	*3 per loop*
U-13	Feedwater throttle valve failed open	*6 per loop*
U-14	Loss of one recirculation pump	*8 per loop*
U-15a	Turbine trip (without reactor trip)	*50*
U-15b	Turbine trip with reactor trip (loss of main condenser or similar problem)	*10*
U-16	(Deleted)	
U-17	(Deleted)	
U-18	Loss of all offsite power	*16*
U-19	Plant shutdown in response to small sodium-steam/water leak indications	*3 per loop*
U-19a	(Deleted)	
U-19b	(Deleted)	
U-19c	(Deleted)	
U-20a	Inadvertent opening of one turbine bypass valve	*5*
U-20b	Turbine bypass valve fails open following reactor trip	*5*
U-21a	Inadvertent opening of evaporator outlet safety power relief valves	*5 per loop*
U-21b	Inadvertent opening superheater outlet safety/ power relief valves	*3 per loop*

U-22 Inadvertent opening of SGAHRS steam drum vent valve *3 per loop*

U-23 Inadvertent opening of evaporator inlet dump valve *3 per loop*

U-24 Reactor trip with failure of one PACC to perform *10*
Note: These events are part of the reactor trips for event U-la and should not be added as separate trips.

C. **Emergency Events**

The frequencies for these events are that each component must accommodate 5 occurrences of the most severe emergency transient for that component (one every 6 years) plus two consecutive occurrences of the most severe event (or consecutive occurrences of 2 unlike events if the unlike events provide a more severe effect than consecutive occurrences of the most severe event). However, if event E-15 is the most severe condition for a component, it shall be evaluated for a frequency of 2 for that component in addition to the 7 occurrences of the next most severe transient.

E-l Primary pump mechanical failure

E-2 Intermediate pump mechanical failure

E-3a (Deleted)
E-3b (Deleted)
E-4a (Deleted)
E-4b (Deleted)
E-4c (Deleted)
E-4d (Deleted)

E-5 Loss of primary pump pony motor with failure of the check valve to shut

E-6 Design basis steam generator sodium/water reaction

E-7 One loop natural circulation heat rejection from initial two loop operation

E-8 Rupture disk failure in SGS sodium/water protection system

E-9a Water/steam side isolation and dump of an evaporator/ superheater module with failure of a module outlet isolation valve to close
E-9b Water/steam side isolation and dump of an evaporator/ superheater module with failure of an evaporator inlet isolation valve to close
E-9c Water/steam side isolation and dump of an evaporator/ superheater module with failure of a superheater inlet isolation valve to close

E-l0 Water side isolation of an evaporator module with failure of the water pump valve to open

E-1l Steam side isolation of a superheater with failure of one relief valve to open

E-12 (Deleted)

E-13a (Deleted)
E-13b (Deleted)

E-14 Inadvertent dump of intermediate loop sodium

E-15 DHRS activation 24 hours after scram

E-16 Three loop natural circulation

E-17 Two loop natural circulation heat rejection from initial three loop operation

E-18 Two loop natural circulation

E-19 Loss of flow in two sodium loops

D. **Faulted Events**

F-1 (Deleted)

F-2 DHRS Activation without SGS cooldown

F-3 a Feedwater line rupture between steam drum and inlet isolation valve
F-3b Feedwater line rupture in main incoming header

F-4a Saturated steam line rupture
F-4b Main steam line rupture

F-4c Rupture between superheater module outlet and superheater outlet isolation valve
F-4d Rupture between superheater outlet isolation valve and main steam line

F-5a Recirculation line break between drum and recirculation pump inlet
F-5b Recirculation line break between evaporator outlet and drum inlet

F-6 Intermediate Loop Sodium-Air Leak

A LIFETIME EXTENSION PROJECT FOR THE PHENIX REACTOR: ADDITIONAL KNOWLEDGE REGARDING THE IN-SERVICE BEHAVIOUR OF ITS MATERIALS AND STRUCTURES

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Abstract

The PHENIX Life Extension Project groups together all the actions required to pursue operation of the reactor, particularly with a view to performing irradiation experiments in the framework of the back-end of the fuel cycle programs. As such, it comprises a series of investigations whose objective is to assess the state of the reactor after about one hundred thousand hours of operation.

The following points have been particularly investigated :

- The materials behavior (austenitic, austeno-ferritic, and ferritic steels - base metal, welds, heat affected zones) in terms of thermal aging and its effect on mechanical properties, embrittlement, sensitiveness to corrosion (in normal and incidental environment), and radiation effect on the potentially exposed structures.
- Furthermore, specific programs have been devoted to the assessment of thermo-mechanical response of some particular components. This concerns some types of welds with regard to fatigue or creep fatigue, some parts of large shells with regard to ratchetting and buckling, and main secondary piping.
- An extensive program was dedicated to the recovery of the thermo-mechanical damage undergone by the structures and its extrapolation to the future. This has led to consider in details thermo-hydraulical effects such as fluctuations in streams and bedding zones.
- Some intergranular cracking of welded joints had to be closely examined ; this was achieved by a research work that has produced important advancements in that field.
- With the aim of evaluating potential defects, real progresses have been made in the knowledge of large defect's behavior in thin shells.

The feedback of the examination and studies was also derived in terms of relevance of manufacturing, exploitation and monitoring conditions. It is believed that this experience will be useful for future design rules.

1. INTRODUCTION

Built at the beginning of the 1970s and linked up to the network on December 13,1973, the fast reactor (FR), PHENIX, has now accumulated almost one hundred thousand hours of operation. Its regular operation until 1990 enabled researchers to validate the concept, acquire considerable experience in the field of operation and did much to improve fuel performance. Incidents occurring during operation concerned the intermediate heat exchangers, and later the steam generators. They were understood and quickly brought under control.

At the end of the 1980s, automatic shutdown problems involving negative reactivity surges began to appear along with the first signs of major component aging on the secondary circuits. The latter, however, brought about much longer investigations given the related questions of safety issues involved. Therefore from this time onward, the reactor has only been in operation for short campaigns of an experimental nature.

At the same time, serious thought devoted to the back end of the nuclear cycle, has emphasized the significance of FRs and the role they might play in the management of plutonium and the transmutation of long-term nuclear waste. The associated research and development programs, CAPRA and SPIN have been designed to provide answers that will fit into the framework of the Law of 1991, which has set the final date of 2006 for convincing solutions to the question of nuclear waste to be put forward.

Viewed within this context, PHENIX, alongside SUPERPHENIX, appeared to be an important tool. This importance has, of course, become even greater since the decision to shut down SUPERPHENIX was made.

These elements, establishing the probability of operation until about the year 2004 in order to take the operation deadlines for irradiations into account, led to the launching in 1993 of a «reactor rejuvenation» project. It is founded on a detailed assessment of the reactor's actual state and takes the requirements set down by the French safety authorities into account.

This whole operation is entitled «The PHENIX Reactor Lifetime Extension Project».

The aspects which interest us here are those which, for the purpose of assessing the condition of the materials to justify prolonged operation, have yielded knowledge. We shall comment first and foremost on aspects concerning the nuclear steam supply system understood as the reactor block and the secondary circuits, including the steam generator.

What we are dealing with is a whole entity built, for the most part of stainless steels and which operates within a temperature range of 350 to 550°C (600°C locally), under low pressure, in a sodium environment, neutral gas or air (water-steam under high pressure for steam generator tubes).

Two essential elements contribute to the importance and pertinence of the data acquisition :

- First, the effort expended in reconstituting the original state of the reactor at the time of its commissioning (a re-examination of its design and construction, the thermal-mechanical loads and chemical environments...) and the consequences,
- And second, the accessibility of the secondary circuits which has allowed a great number of detailed examinations to be carried out.

2. BEHAVIOR OF THE MATERIALS, THEIR AGING UNDER TEMPERATURE

PHENIX uses a whole range of important steels :

- **Austenitic stainless steels of the following type** :

18% chrome-10% nickel, low or medium carbon content, some stabilized with titanium, others not ;
17% chrome-12% nickel plus molybdenum low or medium carbon, either stabilized or not with titanium ;

- **Chrome, nickel and molybdenum** (in a composition 19-12-2, 16-8-2...), in the form of welding products ;

- **Ferritics** (non-alloyed or slightly alloyed with chrome and molybdenum) ; austenoferritics (in the form of molded products).

Such diversity created a need for data in the assessment of the materials through calculation and the opportunity of completing the basis. Thus, knowledge of these materials sometimes lacking, sometimes requiring confirmation, especially in the aging state, has been completed. Such is the case of the Z6 CNT 18-10 and the Z6 CN 18-10 steels with regard to low cycle fatigue characterization (consolidation, symmetrization, fatigue strength) and of course in an aged state (one hundred thousand hours) at service temperatures for the instrumentation concerned, (i.e. the major secondary circuit piping and the steam generators).

In the field of creep rupture and creep strain, the study has been expanded to include the Z2 CND 17-12 steel, for the specialized field dealing with short time periods and high temperatures (pertinent for certain accident sequences).

The most notable effects of aging deal with the secondary creep speeds in the Z6 CNT 18-10 steel which are several times higher (by a factor of ten) than at the non-aged state (see figure 1).

In the field of welding filler metals, research on the embrittlement of the «cold» parts of the circuits has been carried out (temperature range : 350-400°C). Thus, a program verifying toughness in an aged state was conducted, based on the grade of steel present in the secondary circuits(19-12-2). The embrittlement of molded austeno-ferritic steels was assessed in comparison to its initial state after 75000 to 90000 hours at 350 to 400°C. This first involves tension and impact strength measurements performed at ambient temperature for samples on the primary pump and both ambient and service temperatures for samples taken from the secondary circuits. Concerning this last case, an ongoing program is performing toughness measurements and also examining the effect of overaging.

Simultaneous knowledge of toughnesses and impact strengths at both cold and hot temperatures will enable us to test the correlations (commonly used for PWRs in particular) while being aware of the fact that the ferrite contents, a bit lower in the PHENIX reactor, are a potential source of deviation.

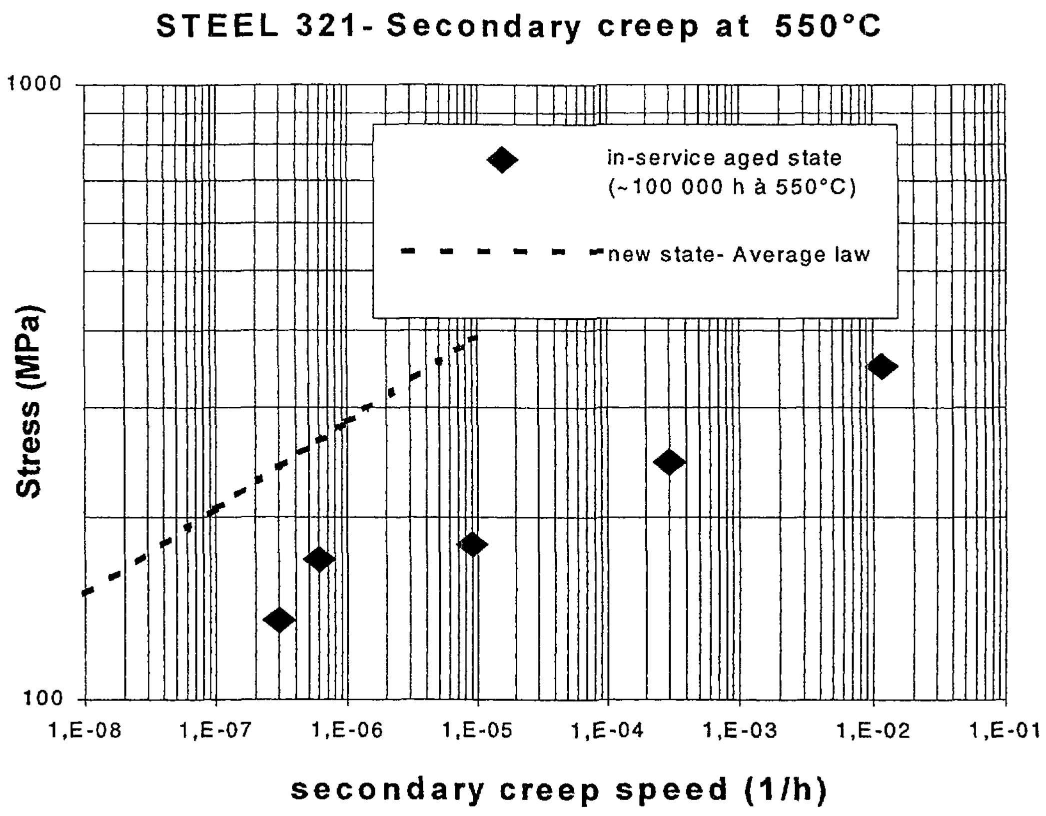

FIG. 1 Secondary creep speed for steel 321

Verification regarding the state of the sensitivity of various materials to intercrystalline corrosion (basic materials, filler materials, welding heat affected zone) has also been accomplished in the present aging state. Overall confirmation of the diagrams on this subject was thus obtained concerning «low» temperatures and long time periods. This remains a field in which very little data is available. A few deviations or rather, peculiarities in behavior were also observed. For the austenitic steel with molybdenum and low carbon from the reactor block (and in its welded zones), the «nose» of the sensitivity diagram (short time period and high temperature), was defined by specific tests performed on non-aged material. The purpose of this was to demonstrate the harmlessness of the initial stripping treatments performed on the welds of the materials.

3. BEHAVIOR OF THE STRUCTURE MATERIALS UNDER IRRADIATION

Since we are dealing here with structures operating at 400°C, (the supporting elements of the reactor core), it has been shown that the upper diagrid plate in its central zone (which will have integrated 1.75 ± 36% dpa NRT for iron at the end of its lifetime) will sustain a slight embrittlement. Verification of this effect is to be carried out on the structural base of an assembly (an extractable component) which has remained within the reactor for the quasi-totality of its present lifetime. On the lower structures, the anticipated effect is negligible. The structures exposed to high temperatures (above the core), where the potential effect of the irradiation is different and becomes evident by an embrittlement of the grain boundaries caused by helium, (which itself results from the transmutation of the boron and nickel), are, in fact, only slightly affected in the PHENIX reactor owing to the efficiency of its top neutron shielding.

4. KNOWLEDGE OF THE THERMAL-MECHANICAL LOADS

Studies conducted on the PHENIX reactor in this field provided the first practical industrial application of recent developments in thermomechanics, particularly those concerning the Large-Scale Simulations of turbulence in the calculation of flow instabilities. Thus, the mixing of fluid sprays at different temperatures having led to the cracking of a secondary piping zone at a «T» level were analyzed in their fluctuating aspect. This study, incidentally, has become presently the object of international intercomparisons (see figure 2).

The core outlet zone has also been the object of extremely minute modeling in order to evaluate the effect of heat fluctuations on the lower structures of the above core structure.

On an even greater scale, an overall modeling of the bottom of the hot plenum in all its geometric complexity, including the inter-assembly, aims at assessing the fluctuating aspect of the inner containment vessel's stratification and at establishing both its extent and its temporal and spatial characteristics (either local or extensive). This program includes the design and construction of a new thermometric rod on the reactor which will be sensitive enough to cross-check calculations.

Such advances must quite logically lead to the evaluation of the mechanical consequences on the structures in terms of random fatigue or crackling. This will take place in the near future by coupling mechanical codes, by carrying out complementary studies focusing on the response of the materials in terms of damage and, *in fine*, a complete approach methodology for these problems.

5. MECHANICAL DESIGN

A few examples will be cited here illustrating the significance of the PHENIX feedback in the field of design.

In the field of angle weld fatigue, an experimental program has been set up. It has allowed various parameters to be tested such as the effect of back welding, the effect of medium stress and the effect of a crossing with a perpendicular weld. These results, apart from the immediate application made of them for reactor files, can be used for design methodology, be it on the basis of linearized stresses and coefficients or a local analysis obtained with a detailed modeling. In several cases, it has been necessary to carry out an experimental study involving the strength of special welds against primary stresses.This study revealed the extreme conservatism of current rules concerning ductile materials.

In the field of cyclical loads of the same special welds, the inadapted nature of simplified, present day rules (through the weld coefficients) has become evident whether it be in the progressive strains or in the fatigue-creep. This has brought about the extensive use of

LIGNE SECONDAIRE PHENIX
Simulation des grandes echelles du melange
en aval d'un piquage
Evolution dans le temps des isothermes
dans le plan de symetric longitudinal

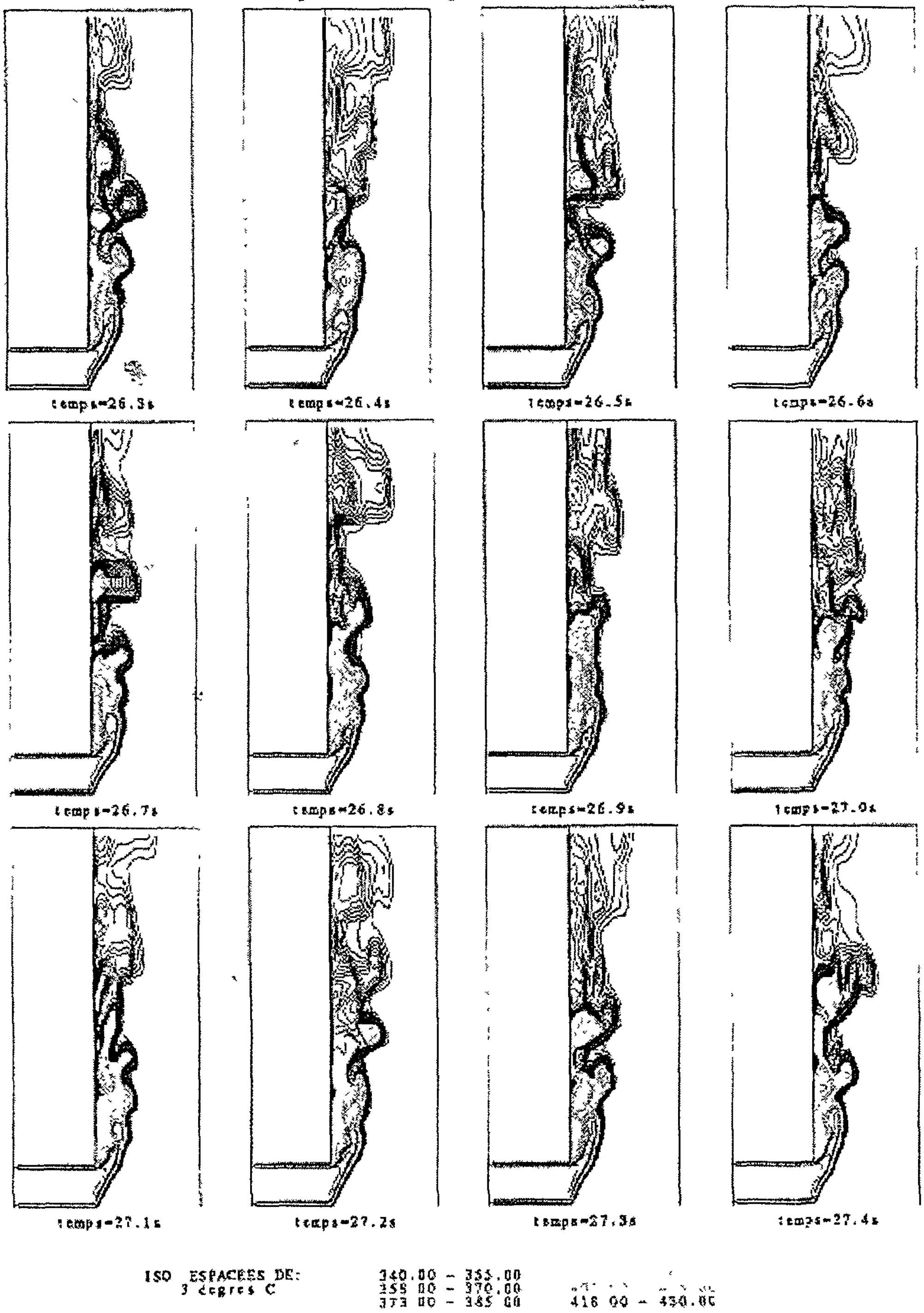

FIG. 2 : PHENIX Secondary Line
Large–Scale Simulation of the mix downstream from a sampling/connection.
Evolution in time of isotherms, in a lengthwise symmetry view

the recently developed A16 [1] project guide which is dedicated to the assessment of defects. Thanks to this, important feedback is now available and can be used :

- in comparison with experimental points involving very long lengths of time since the welds after functioning for 50.000 hours in the reactor have been removed and examined by experts for appraisal,

- in comparison with experimental programs which have been carried out or are still ongoing.

This feedback puts forward two essential questions concerning the following :
- the need to reduce conservatism ;

- the need to take *mixed creep modes* into account.

The free surface zone of the PHENIX reactor's primary containment has been the object of a carefully drafted file especially with respect to the question of buckling and thermal stress ratcheting, mechanisms that can cause plastic strains. Geometric measurements made in this relatively accessible zone of the reactor will prove to be of great significance in evaluating the precision of the methods and must be planned at least at the end of life. The major secondary circuit pipes have been the object of damage reconstitution. At the same time, they have been and will once again be inspected by a non-destructive control. Furthermore, dismantling and removal operations are now underway on certain sites appearing to have accumulated the most amount of damage. Comparison of these results is of the greatest value when one knows that the complexity of the damage suffered by a component, particularly during creep conditions, can prove to be significant for :

- The analysis of the stresses and strains ;

- The estimation of the stress acting under heat conditions ;

- A consideration of the local effects (welds, geometric discontinuities).

Finally, in a different field, the seismic re-evaluation of all buildings has revealed the necessity of devoting deep thought to the methodology and criteria that must be applied to the study of existing facilities.

6. HIGH TEMPERATURE INTERGRANULAR CRACKING OF WELDED JOINTS

Feedback from PHENIX includes observations regarding intergranular cracking under heat conditions in the area around the welded joints that have stimulated R&D work given the fact that conventional methods of design do not deal with these questions or do not enable us to be sufficiently aware of them.

First and foremost, this concerns the Z6 CNT 18-10 steel which is known to have a certain sensitivity to the phenomenon termed relaxation cracking. Observed on certain parts of the major secondary pipes, in the buffer tanks and on the headers on the top floors of steam generators, the problem was to approach it quantitatively so as to establish definite conclusions about other components. A procedure was designed, designating as its point of departure, the strained state and the precipitation of the zones in question as principally determining the cracking. This state was simulated on a mass material, and the intergranular cracking under only one imposed movement was reproduced in a laboratory. (see figure 3)

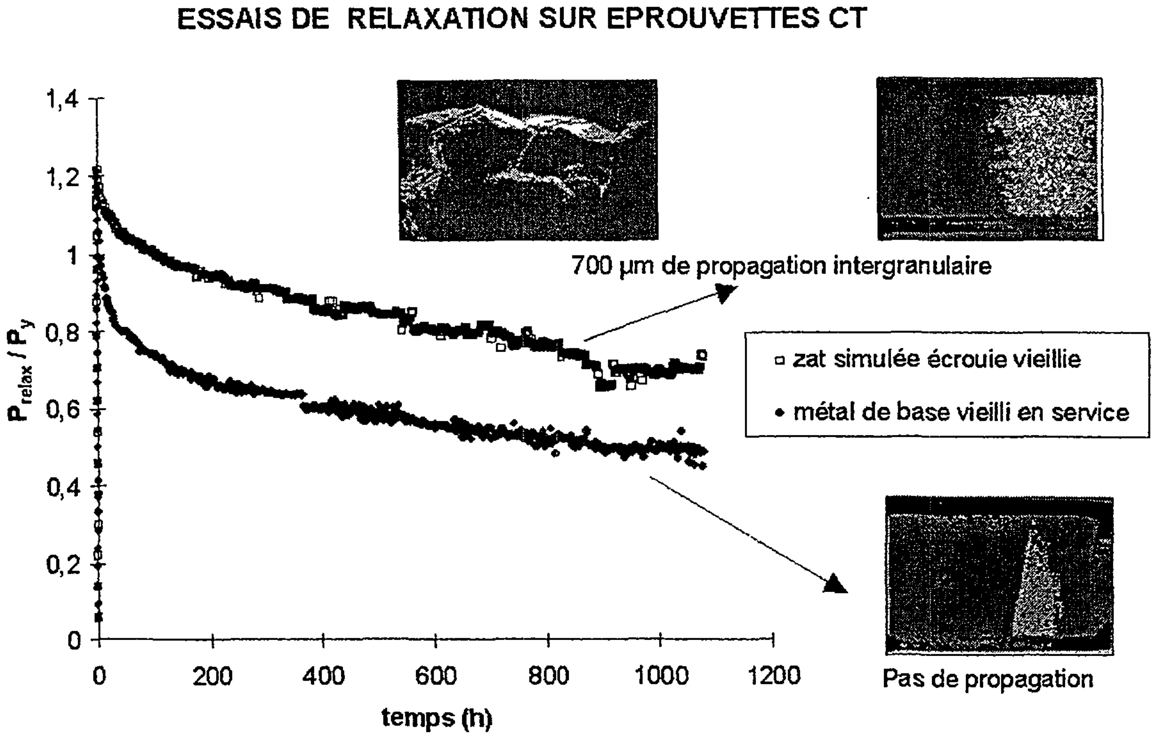

FIG. 3 : Experiments of relaxation with CT specimen
(squares) : simulated heat effected zone (hardened and aged)
(diamonds) : aged basis metal (in service)

This result, unusual in light of the present day acquired experimental data (almost exclusively built on new material), at least proves that important ingredients contributing to cracking have been identified in simulations of the thermally affected zones of the welds.
A damage model has been elaborated and adjusted on the basis of tests causing the multi-axialty of the strains to vary. The overall procedure will thus be available very soon in order to deal with industrial situations.

Creep damage is also thought to have led the appearance of intergranular cracks in certain welded assemblies particularly large and flanged ones in Z6 CN 18-10 steel. Apparently, the data and conventional methods which in particular neglect the residual stresses of welding do not properly reveal these findings. Accordingly, another program was launched for the purpose of measuring and modeling them, and to grasp the behavior of the products in question (creep ductility, sensitivity to the multi-axialty of the stresses...). Modeling, moving on to the simulation of welding and the superposition of the operating conditions will accompany a complete structural test which is to be performed on the representative component.

These problems are important for they have appeared after 100 000 hours of operation and whether they call into question present damage evaluation methods or design rules governing welded assemblies, they are at the very heart of lifetime extension concerns. They have been observed on steels that are similar to those developed after PHENIX, and their study provides us with an opportunity to confirm these choices by taking advantage of the compositional differences existing among the three different steel grades, which appear as so many parameters for serious thought about the metallurgy.

The project has provided an opportunity to test local mechanical approaches on an intermediate scale (quite typically the one used in metallography and the one for the grain) in order to attempt to link up metallurgical evolution due to in-service aging and intergranular damage to the macroscopic mechanical behavior. Within the framework of the project, these questions have brought about an international exchange and a thesis has been written on the subject under the direction of the Ecole des Mines in Paris.

7. EVALUATION OF THE HARMFULNESS OF DEFECTS

The demands related to the prolonged operation of the PHENIX reactor have led us to focus on hypothetical defects which, in the various operating conditions envisioned, might question the principle safety functions, and among these, the control of the reactivity.
Therefore the extension of such defects, penetrating the great shells which make up the core support structure, had to be evaluated. In order to facilitate their early detection, the geometric disorders associated with much shorter defects constituted another point of interest in the specification of surveillance methods.

At first the analyses were carried out in the case involving the elastic linear behavior of the material by using formulas of existing stress intensity, often coming from studies on plates and arranged by a curb correction.

These formulas turned out to be exaggeratedly conservative, the result being that new formulas, built on a calculation basis of specific finite elements were put together. These formulas, constructed for cones and cylinders whose radius ratios over thickness are on the order of 150 to 250, deal with length defects extending to the radius of the shell or indeed even beyond that. They have been the object of cross-checks with other formulas in their common validity field and have results in documentation. They are henceforward available for other applications. Elastoplastic calculations allowing direct access to the «J» cracking force have also been carried out.

These calculations, bringing into play crossing length defects which may go beyond the radius of the shells, have been dealt with using a mixed modeling of shells and mass solids. Major feedback is to be found as much on the level of the cracking force (whose value conditions the critical size) as it is in the field of the associated movements, but the essential finding to be derived from both elements amounts to this : the care taken in the modeling of the boundary conditions, as is also the case in buckling, is of prime importance. Therefore, in the case of a fast reactor block, practically all the structures must be represented. This type of calculation has also provided the opportunity to confirm the pre-eminence of loads of a mechanical origin before those of thermal origin owing to their «J» contribution.

8. THE EFFECTS OF MANUFACTURING CONDITIONS

It is interesting to list the number of times an expert appraisal of observed damage has been traced to manufacturing conditions, which, for any new construction must at least involve the responsibility of being able to trace even the slightest details.

Regarding this subject, we could cite as an example the use of chemical cleaning treatments which are too corrosive given the material in question (high carbon austenitic forged material and in a zone where welding is concerned, for instance). It might also be stated that certain methods applied (forming) have proven to be too severe. The inadapted character of certain thermal treatments has also been confirmed (the re-quenching of austenitics after shaping and welding). In welds, local repairs have sometimes been cited as having caused damage by creating a rather special type of stress, i.e. that of a membrane.

In certain cases, the welding procedure parameters have been called into question (excessive energy input).

Feedback on cracking in hot conditions in deposit metals with an austenitic base can also be drawn from the PHENIX experiment. It generally confirms what is known about the influence of elements such as boron in the basic metal (which was the example in the case of the first PHENIX reheaters, removed for other reasons after about approximately 50 000 hours of operation). The same effect applies for welds in a removed metal in which the tendency of ferritic insufficiency has been discovered and defined as a contributing deterioration factor. However, it should be pointed out that cases of cracking under heat conditions are relatively rare and are even more rarely suspected in the development of an in-service defect.

9. THE EFFECTS OF OPERATING CONDITIONS

Concerning the mastery of chemical mediums, and quite apart from certain incidents in the past with the development of aqueous soda in the field of temperature in which austenitic steels are sensitive, or perhaps even a radical decontamination, no significant damage has been reported. The greatest fears concerned the secondary circuits where numerous interventions took place after total draining of the sodium, and where non violent sodium/water reactions occurred at the beginning of the 1980s. The reconstitution of such mediums developed in these conditions, in as much as it remains reliable, especially locally, has not provoked any specific worries. This is in coherence with examinations that do not pose any questions apart from those concerning the presence of a few intergranular markings, therefore not very deep and which are still not clearly understood. An ongoing R&D program dealing with mediums developed in intervention conditions on sodium-soiled components should also confirm present procedures, particularly those concerning drying with control of the dew point at the time of renewed startup.

Several corrosive effects on the external surfaces of the secondary circuits have also been observed (in air) due to pollutants that have proven difficult to identify and which emphasize the importance of mastering chemical environments on all cases.

Last, it should be stated that the return to operating conditions illustrates the great importance of having valves that function properly and reliably and the need to monitor them along with the advantage of simplification in the auxiliary circuits.

10. METHODS OF CONTROL, MONITORING AND INTERVENTION

The PHENIX Lifetime Extension Project has also contributed to an improved characterization and enhanced performances of the sodium leak detection systems, to the determination of sizing of accessible defects through Ultrasonic controls especially on thin austenitic products.
It has provided us with the opportunity to devote some very serious, in-depth thinking about the monitoring of fast reactor structures with a primary integrated circuit.
Extensive studies have been carried out within the framework of this project which, given the conditions of environment, access, precision, quality, deadlines and required cost constitute in themselves precious feedback.

11. CONCLUSION

We should like to insist on the importance of the three aspects of acquired knowledge derived from the PHENIX Lifetime Extension Project :

- The interest in re-examining our Rules of Design and Construction (RCC) [2] and consolidating their criteria, be it on the level of the documented materials, their tabulated properties, their design and manufacturing, the calculation methods, (and this at all levels including the coupling with thermal-hydraulics) and finally, adequate evaluation of the lifetime. We shall add that the treatment of defects encountered whether they be in the field of calculations, non-destructive controls or monitoring should form the basis of a future publication that might be entitled «Monitoring Rules Governing the Use of Materials during Operating Conditions»(RSEM) ;

- The affirmation of the importance of the welded joints in questions regarding reactor lifetime especially during creep conditions which must lead to an increased R&D effort based on the methods adapted to their local aspect and multi-materials.

- The interest in having the project followed up by the necessary actions that are fundamental in profiting from the results. Incidentally, this results could in turn be usefully completed by clearly-targeted expert appraisals (without ignoring an evaluation of the cost versus interest, an item which must be foreseen and taken into account when a reactor reaches its end of life).

ACKNOWLEDGEMENTS

The authors would like to express their thanks to all the agents working at the DRN, DTA and NOVATOME for their contribution to the studies presented herein.

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