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

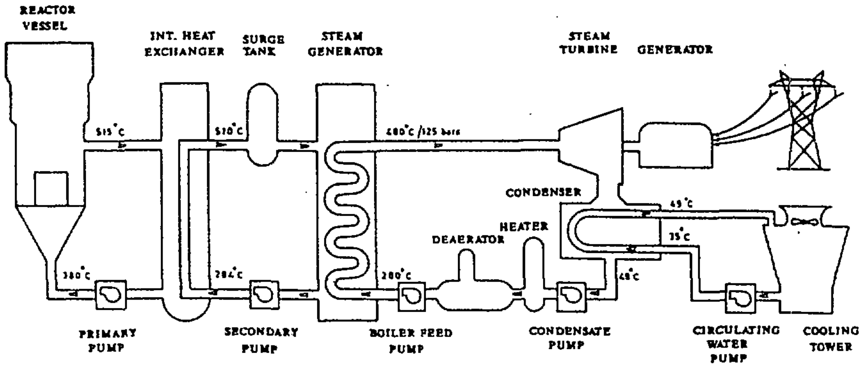


Fig.1

(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

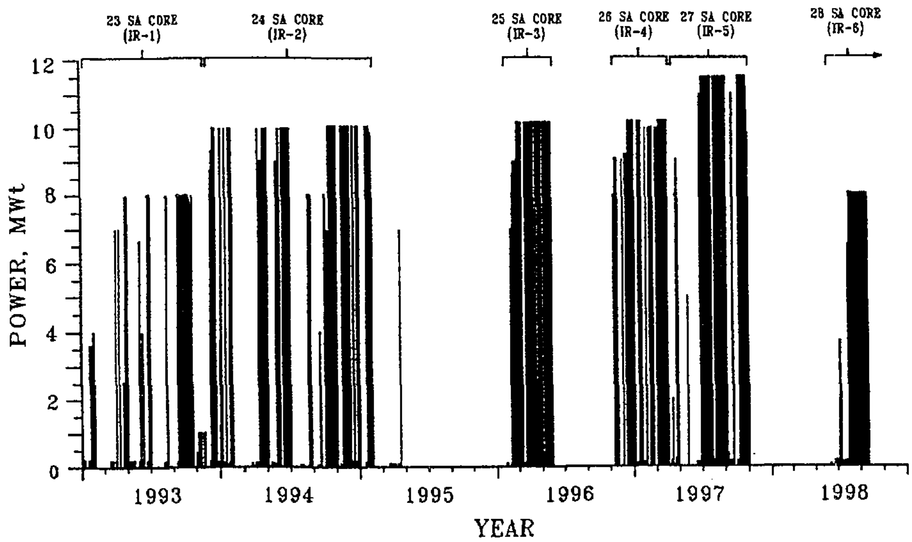
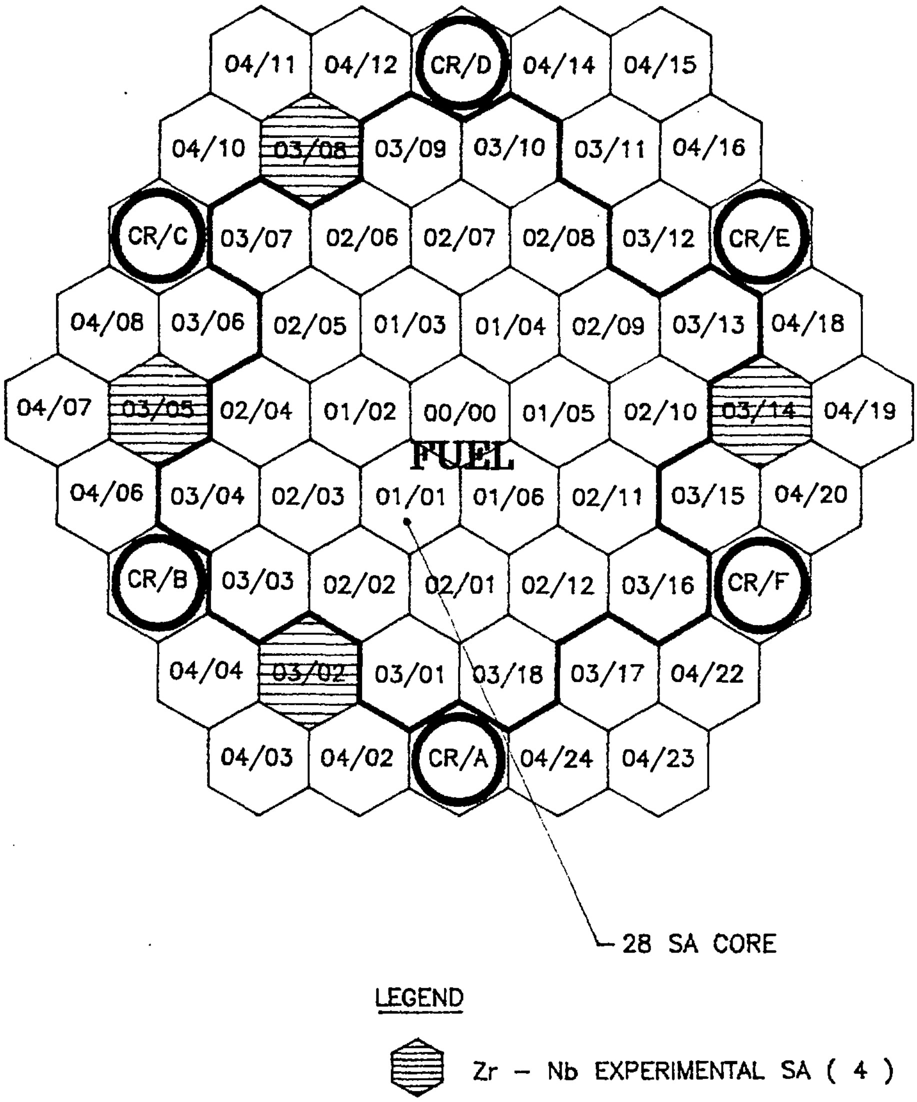


Fig.2



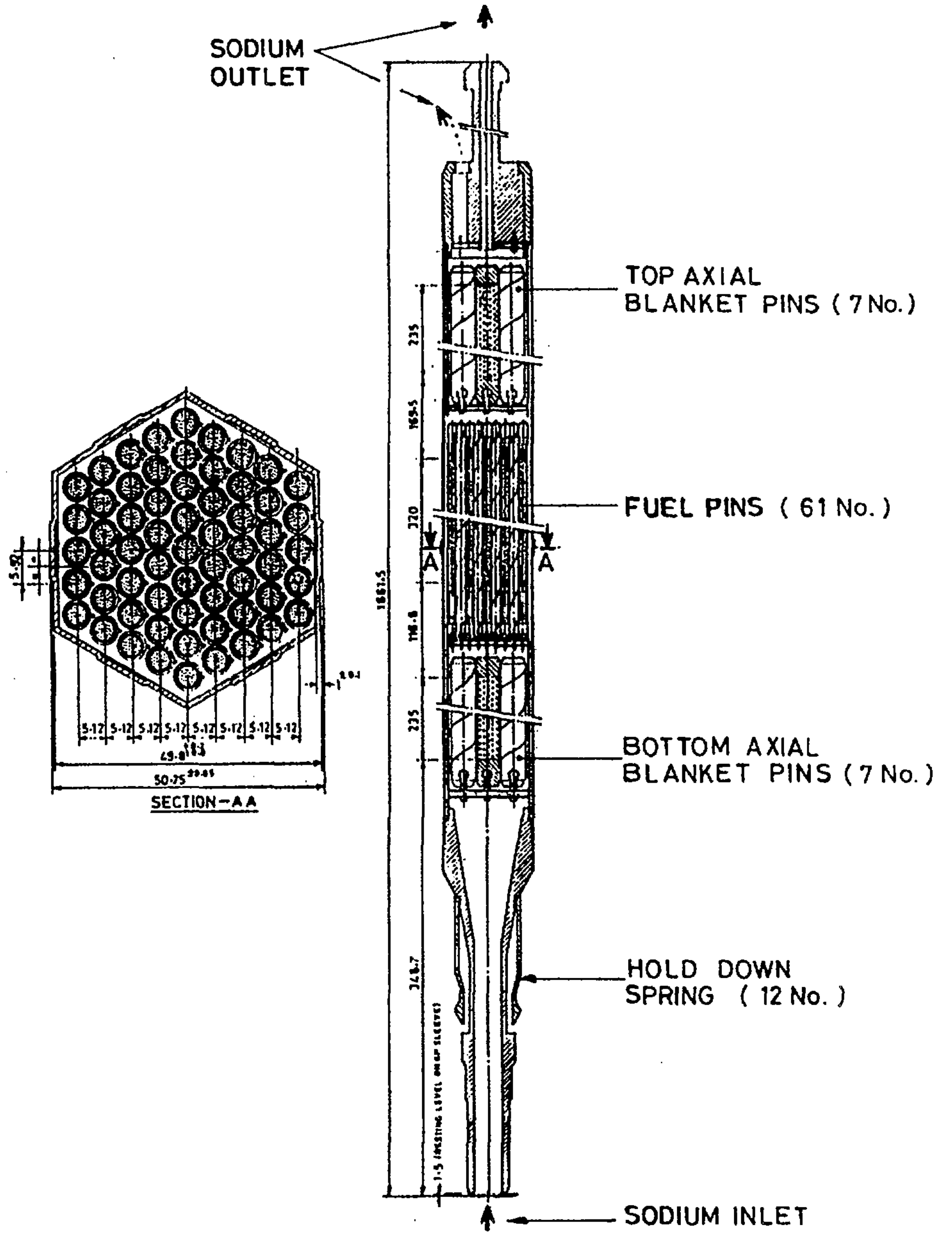
CORE CONFIGURATION

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

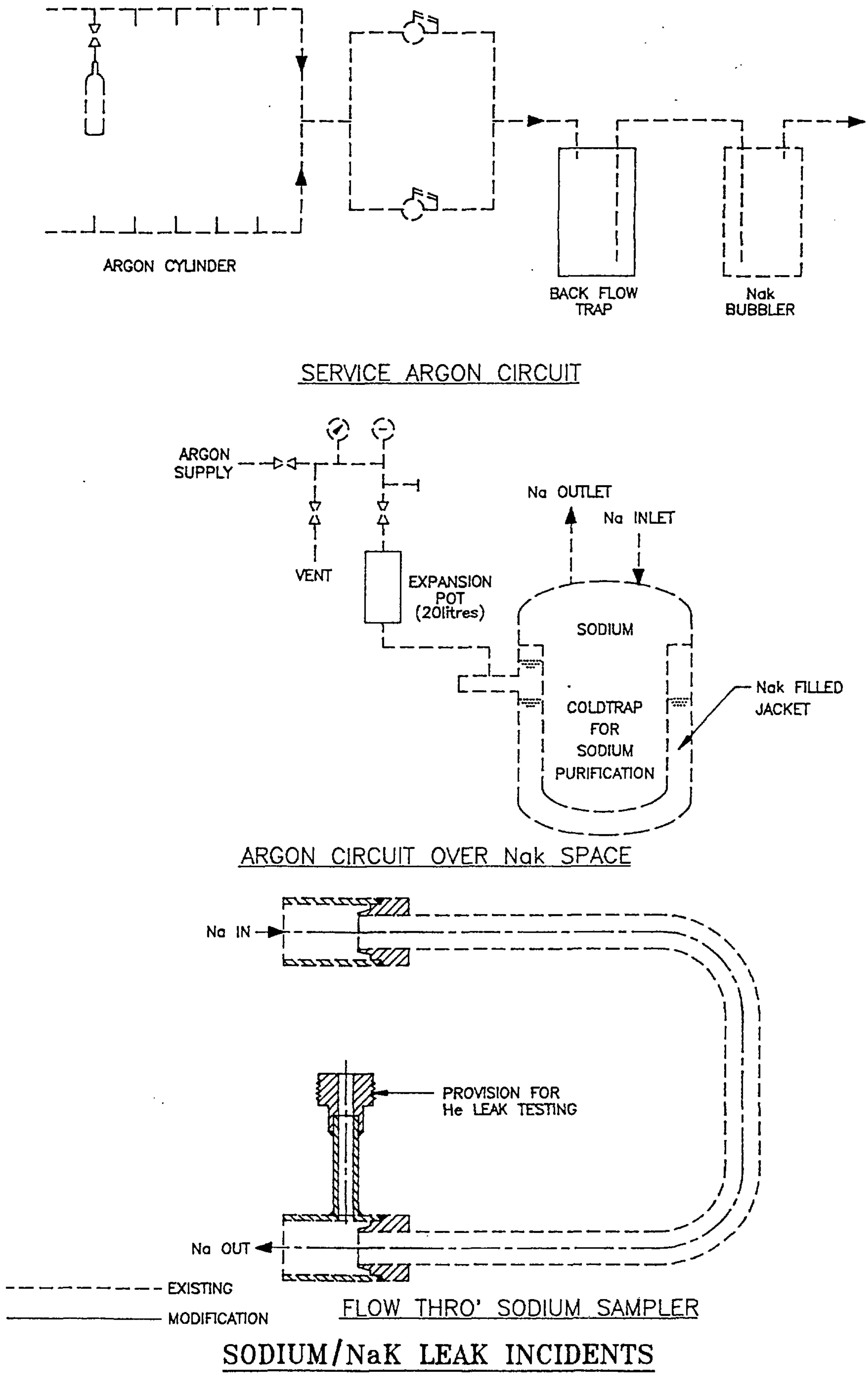

ARGON CYLINDER
BACK FLOW TRAP
Nak BUBBLER
SERVICE ARGON CIRCUIT
ARGON SUPPLY
VENT
EXPANSION POT (20litres)
Na OUTLET
Na INLET
SODIUM
COLDTRAP FOR SODIUM PURIFICATION
Nak FILLED JACKET
ARGON CIRCUIT OVER Nak SPACE
Na IN
PROVISION FOR He LEAK TESTING
Na OUT
EXISTING
MODIFICATION
FLOW THRO' SODIUM SAMPLER
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.

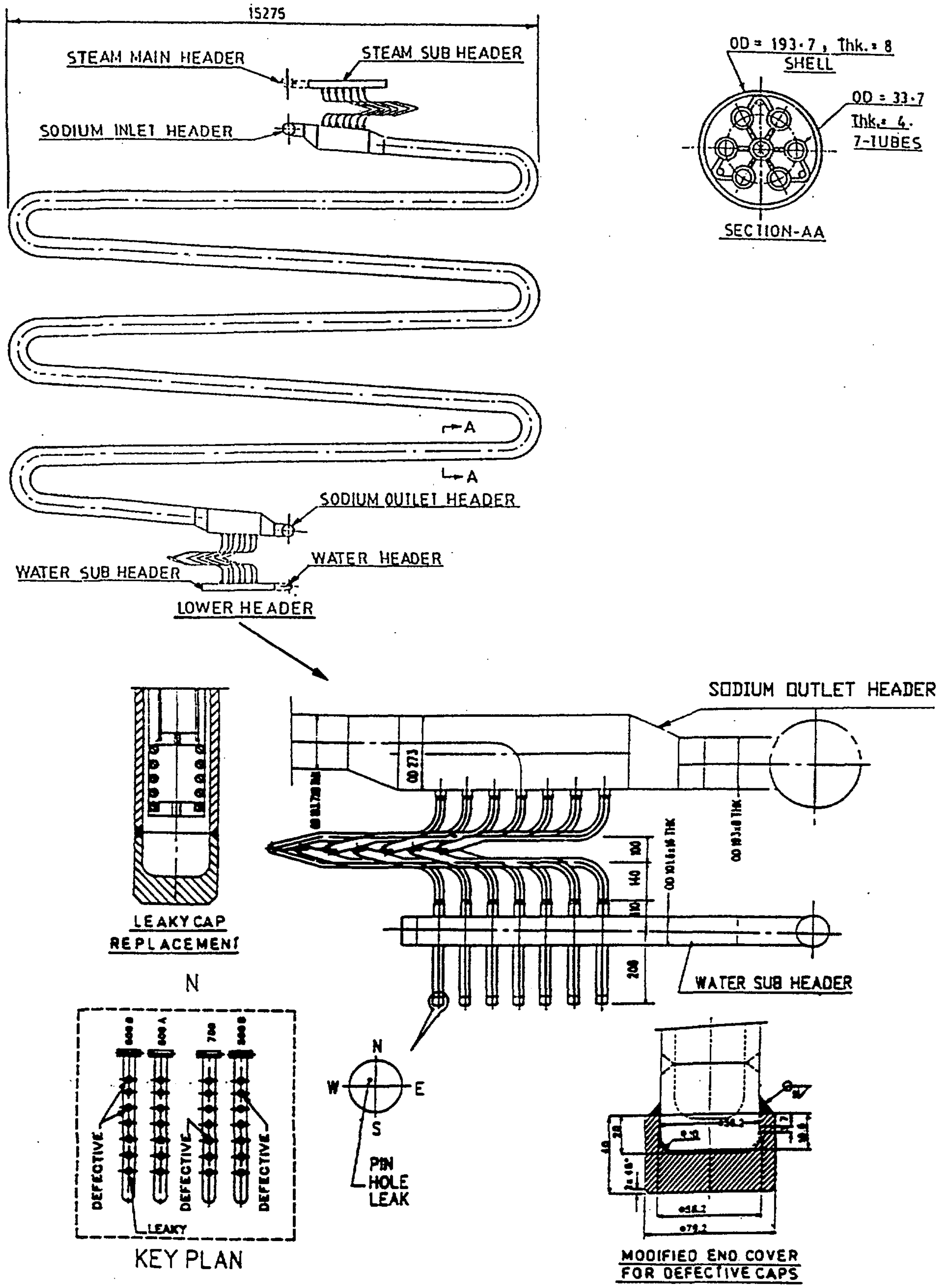
3.2 Water leak incidents in SG subheaders

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.

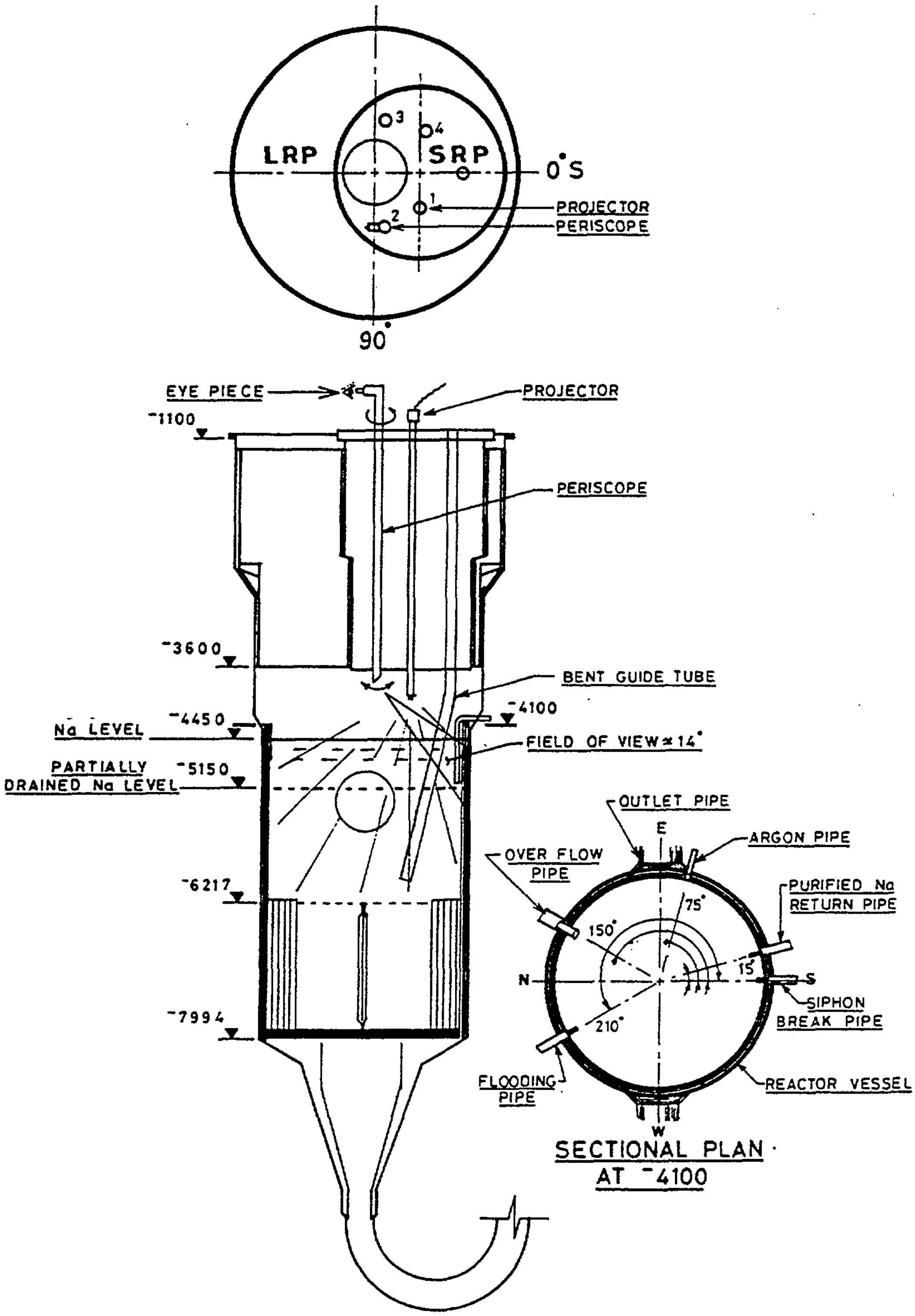
4.1.1 Incident Description

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

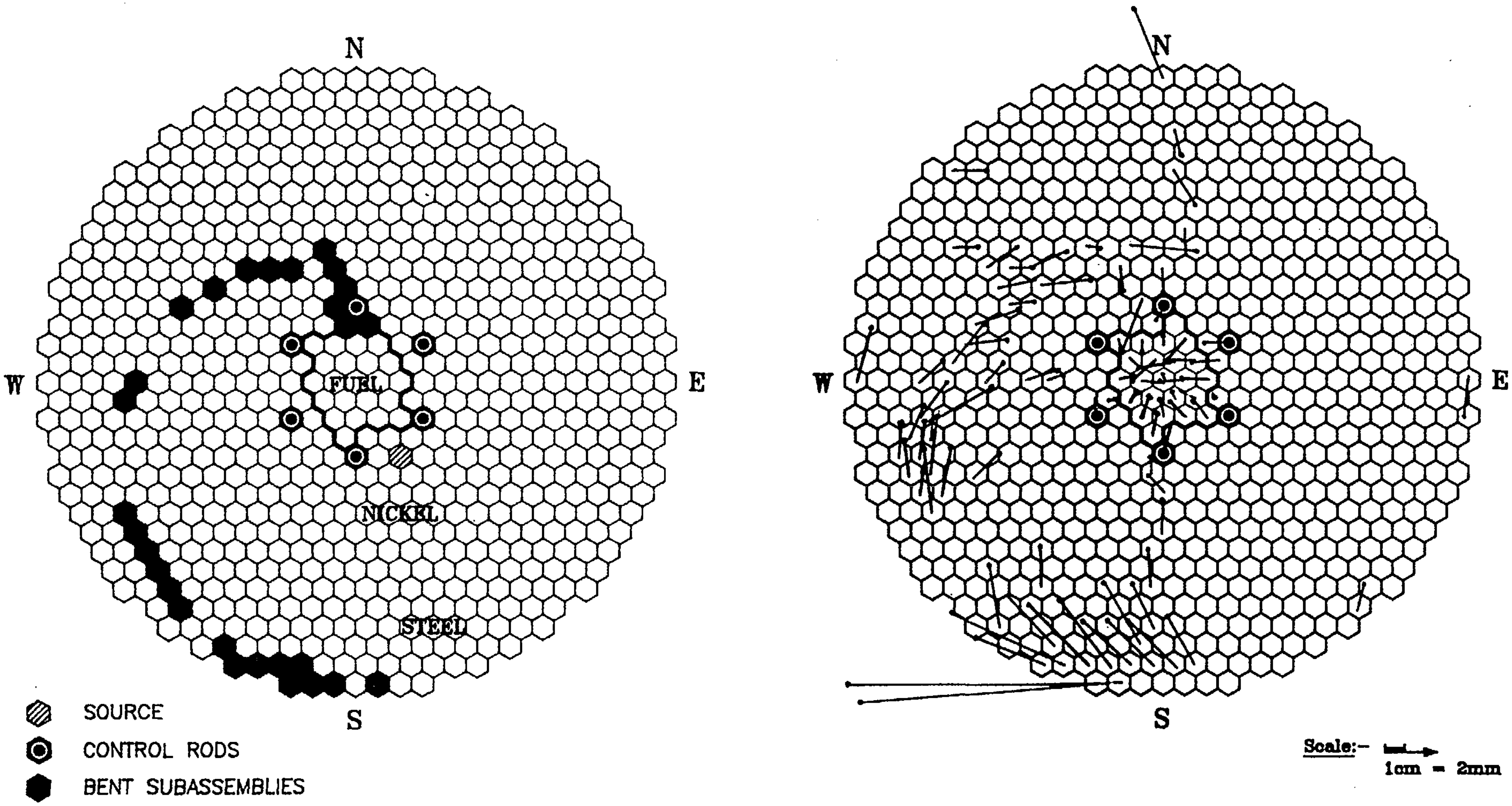
Three techniques were developed and utilised to assess the bend of the guide tube, viz., optical inspection, ultrasonic air gauging and mechanical disc gauging. Optical inspection was carried out with a periscope/projector system (Fig 7). Sodium was drained to expose the heads of SA and the bend of guide 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.

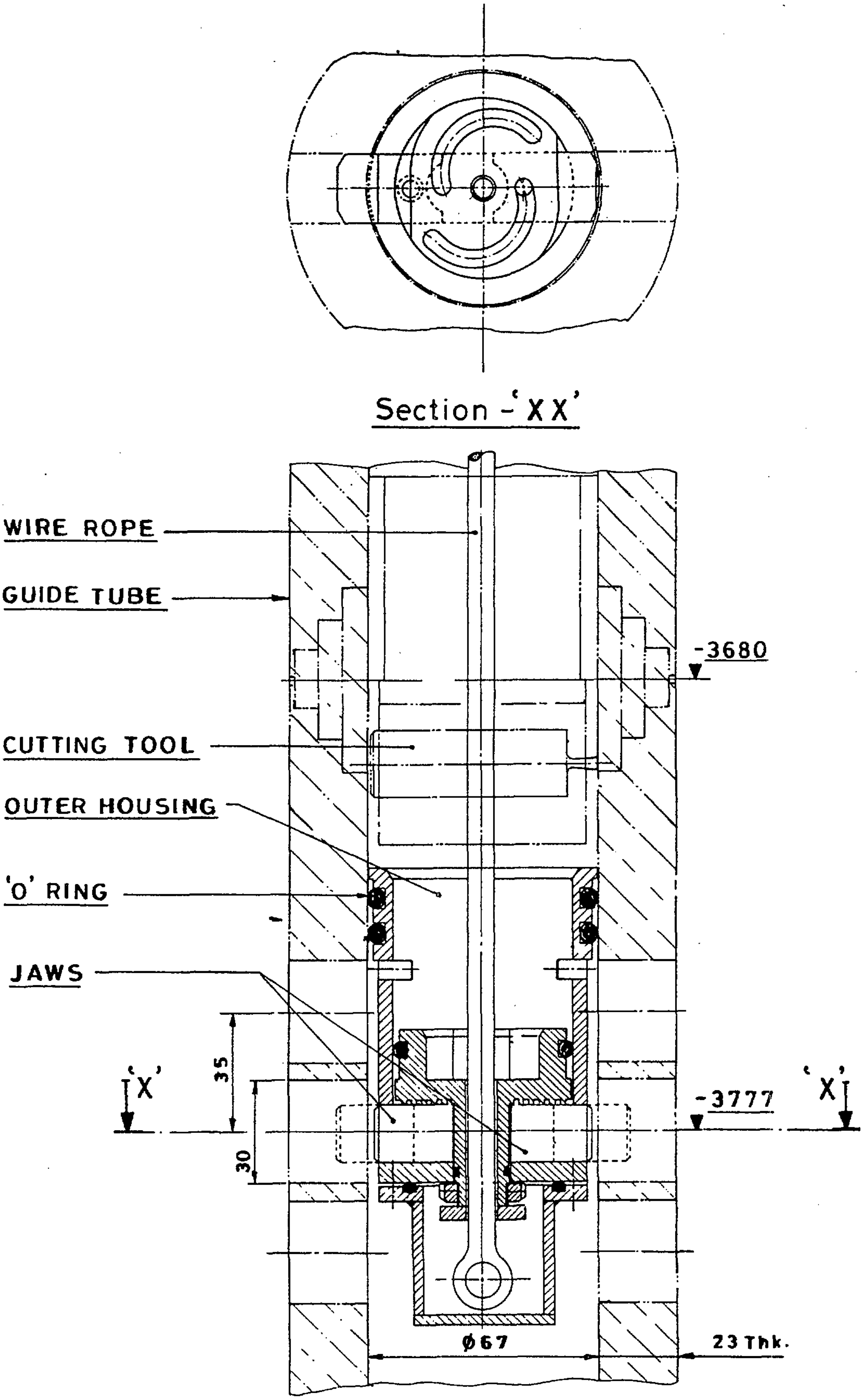
4.1.3 Retrieval of damaged components

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.

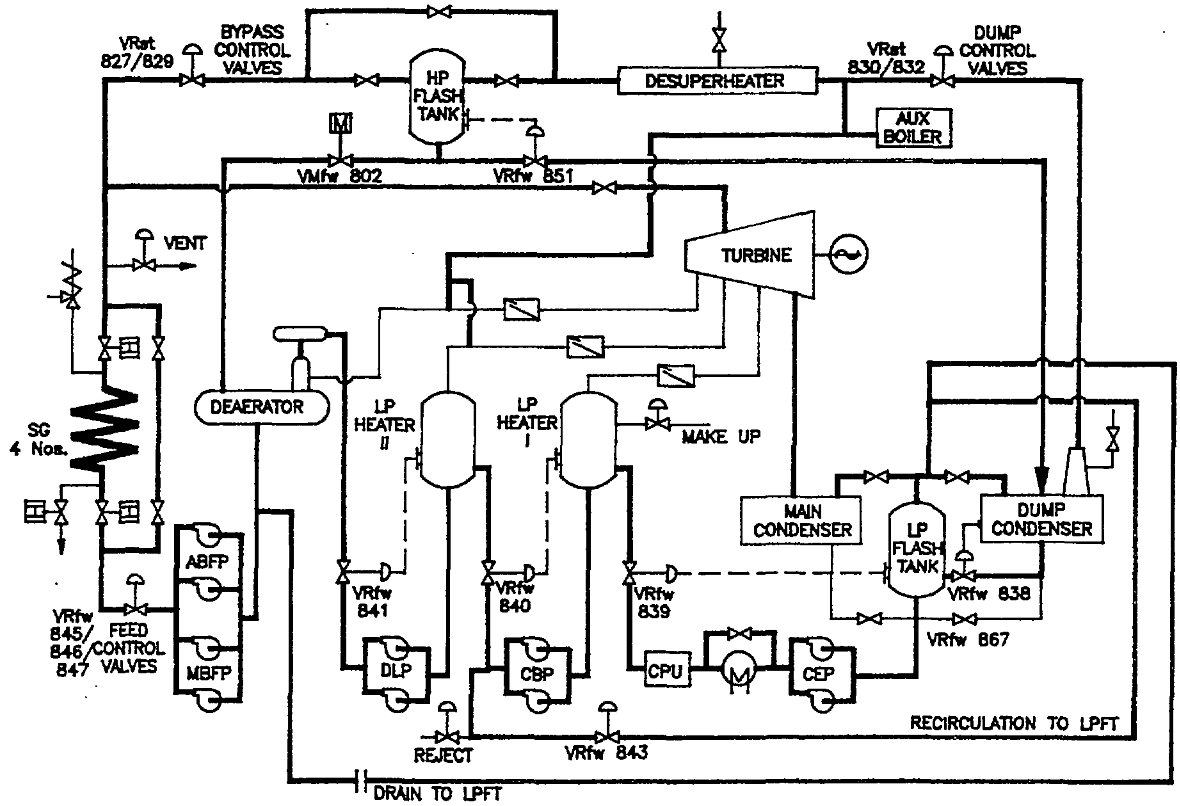
4.3.2 Investigations

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].

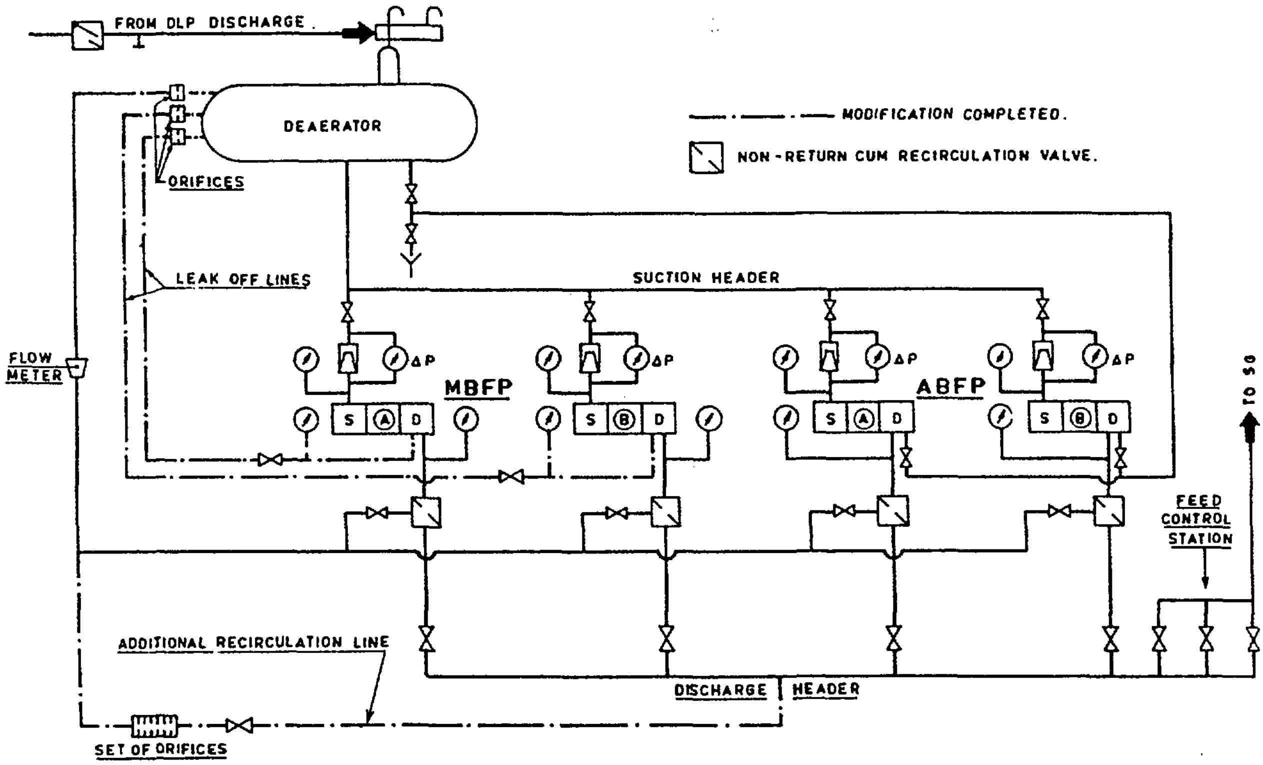
4.3.2.1 Process parameter changes

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

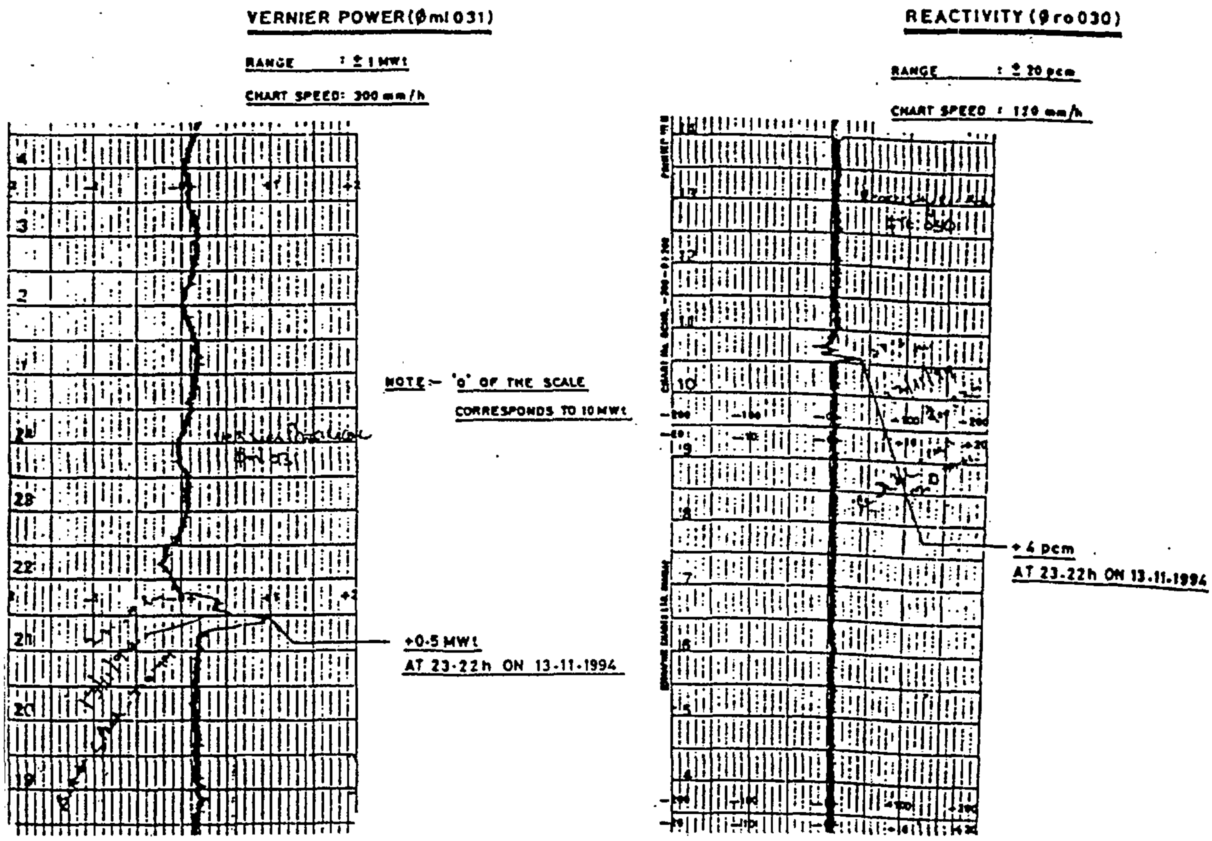


STEAM-WATER SYSTEM FLOW SHEET

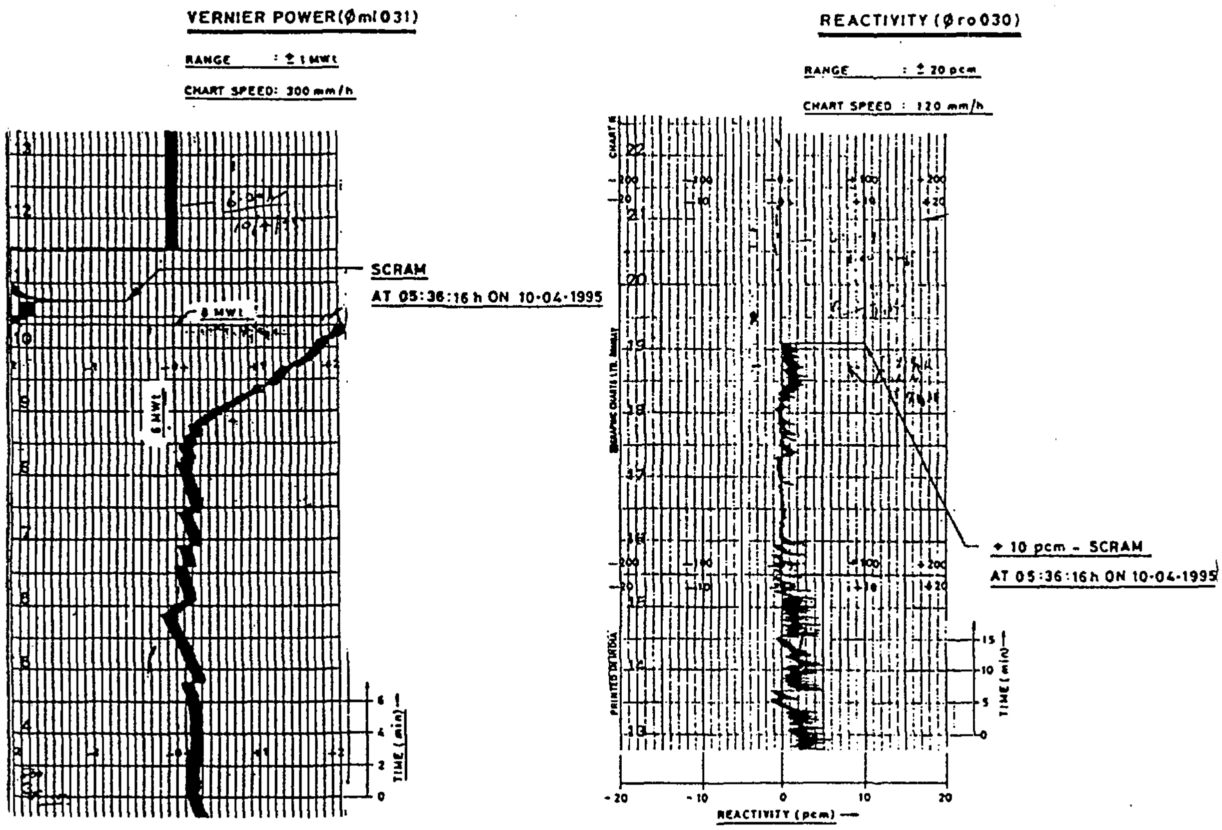


STEAM WATER SYSTEM MODIFICATION

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.

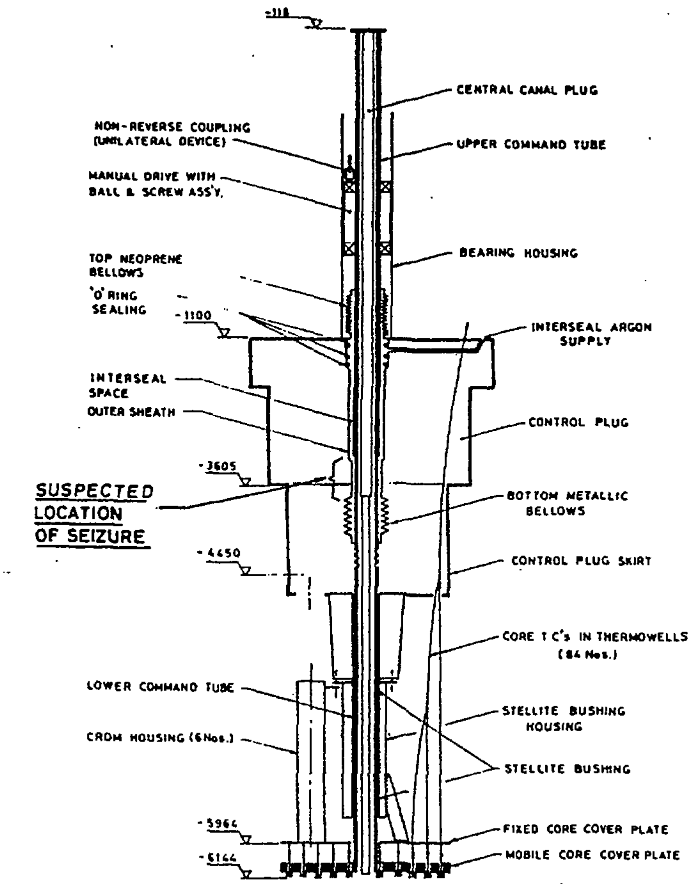
4.3.2.7 Present Status

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.

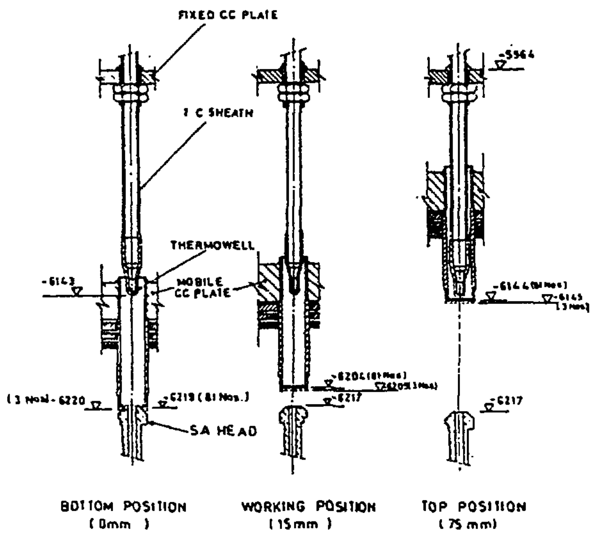
Since the incident in Apr 95, reactor has operated for about 6800 h with 4600 h at high power and the transient has not recurred.

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.

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