WORKING GROUP "HOT LABORATORY AND REMOTE HANDLING"
PLENARY MEETING 1991

LHMA-Services to the Nuclear Industry

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ABSTRACT

In the frame of a general programme related to "Services for the
nuclear power plants", the hot laboratory infrastructure and know-how
have proven to be very important in the examination and evaluation of
degraded and failed components.
An overview is given of some exemplary hot cell examination campaigns,
covering the past few years; these relate to nuclear vessel internals
as well as to components of the pressure boundary and other specific
plant installations.

INTRODUCTION

The objective of this report is to give an overview of specific services available at the hot laboratory installations on request of the nuclear power industry.
In this context, we have not considered the programmes related to fuel examinations nor to pressure vessel steel surveillance; only addressed here are these activities relevant to reactor internals and other equipments suffering from degradation in radiation and or contaminating environments.
Operating experience in the LWRs as well as the modes of mechanical failure generally encountered in structures, suggest potential damage mechanisms that could affect the reactor internals during their life and that may not have been sufficiently considered at the initial design stage. In fact, international meetings on regulatory and life-limiting aspects of core internals have indicated, more than once, the less-developed state of knowledge of these systems as compared for example to pressure vessels; the need for more research and enhanced practical attention to these degradation issues has been emphasized.
The reactor internals are subject to a variety of degradation mechanisms, including irradiation, intergranular- and irradiation-assisted stress corrosion cracking, fatigue, thermal embrittlement and corrosion.
Actually, this is quite interesting a facet of hot cell investigation programmes.
Other installations or equipments in power plants have displayed certain degrees of degradation as well, while being unrelated directly to irradiation effects. Specific contaminating species are then the limiting elements for handling and examination to be performed outside the controlled area of a hot laboratory.

Examples of the situations briefly outlined above will be given in the next paragraphs, illustrating the importance of hot cell infrastructure to deal with the related needs.

IN-CORE INSTRUMENTATION THIMBLES

In order to continuously monitor the LWR core state, flux mapping neutron detector probes are periodically inserted in thimble tubes. These thimble tubes extend through the bottom of the reactor vessel and the lower internals up to the central guide tubes of the fuel assemblies.

The thimbles are guided at different locations in order to ease their introduction into the fuel assembly and to avoid excessive vibration in normal operation. The thimbles must be able to function as a part of the pressure boundary keeping the core sealed as securely as possible. Fig. 1 gives the general arrangement and incore guiding of the instrumentation system.

The primary coolant flow, circulating through the clearance between thimbles and guides, induces vibration of the thimble; this leads to wear, wall thinning and even perforation, by impact of the thimbles against the adjacent components.

In-pile eddy current examination is applied to control this event; the wear volume is measured assuming a certain concentration of wear over the tube circumference in order to define the depth.

Thimble tube sections (304 stainless steel material) showing extended wear after limited operation, as well as their corresponding flow limiters - installed on the lower core plate in order to reduce the vibration level - have been examined in the hot cell facilities. Fig. 2 illustrates the observed degradation.

Non-destructive measurements - profilometry and wear depth control - on thimbles and flow limiters were performed in order to identify and characterize the degraded areas and to obtain possible correlation between in-pile EC measurements and actual wear, so that thimble life can be better predicted.

A limited metallographic examination was a part of the programme as well.

Results of the hot cell examination can be summarized as follows :
- pronounced wear areas up to tube wall perforation;
- wear located over the entire tube circumference, but not uniformly;

- wear depth changes very smoothly;

- outside the degraded areas, the thimble tube remains intact;

- aspects of wear-tapering is measured on the corresponding tube flow limiters;

- localised tube deformation due to coolant pressure is observed in areas where the remaining tube wall is very thin (collapse effect);

- optical microscopy provides no microstructural evidence for the abrasion, which is attributed to tube-to-flow limiter wear;

- no basic microstructural changes occur in the stainless steel and no variation in grain size is observed;

- microhardness measurements on both irradiated thimbles and reference tubes confirm the similarity.
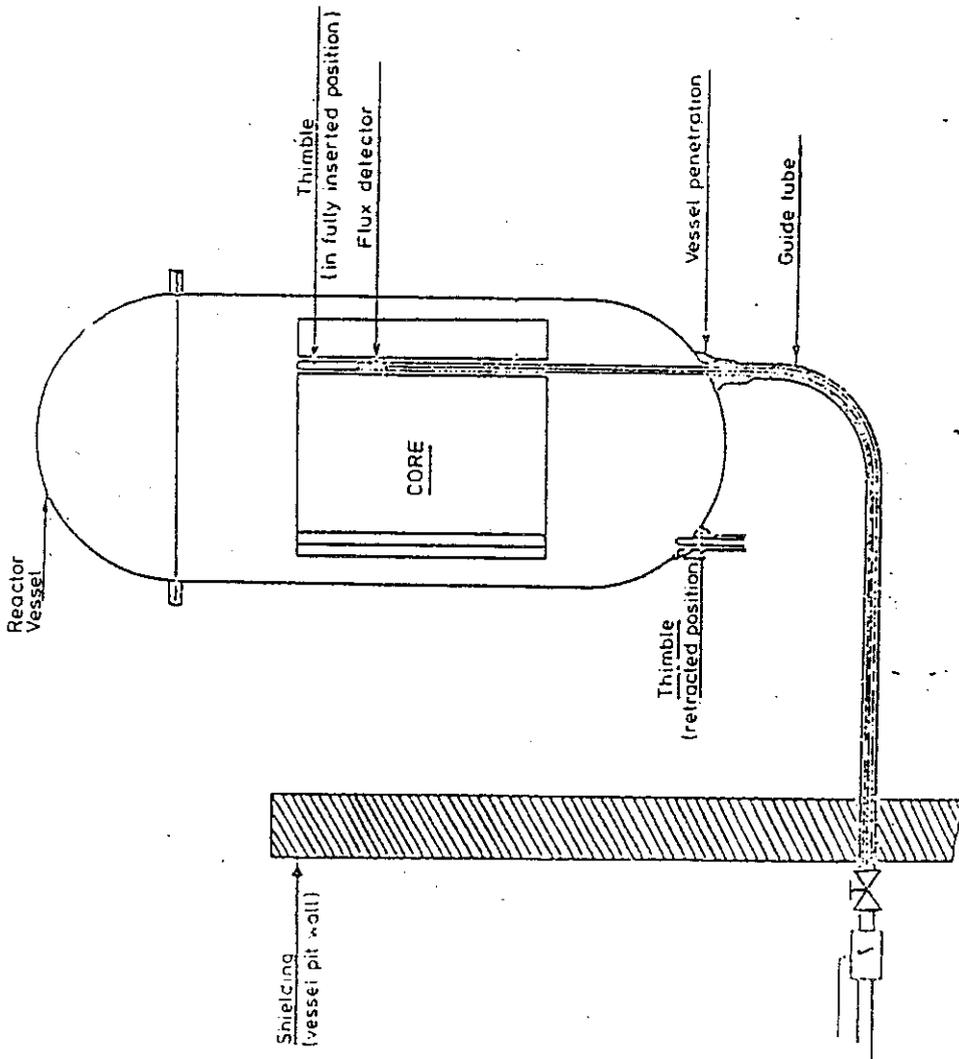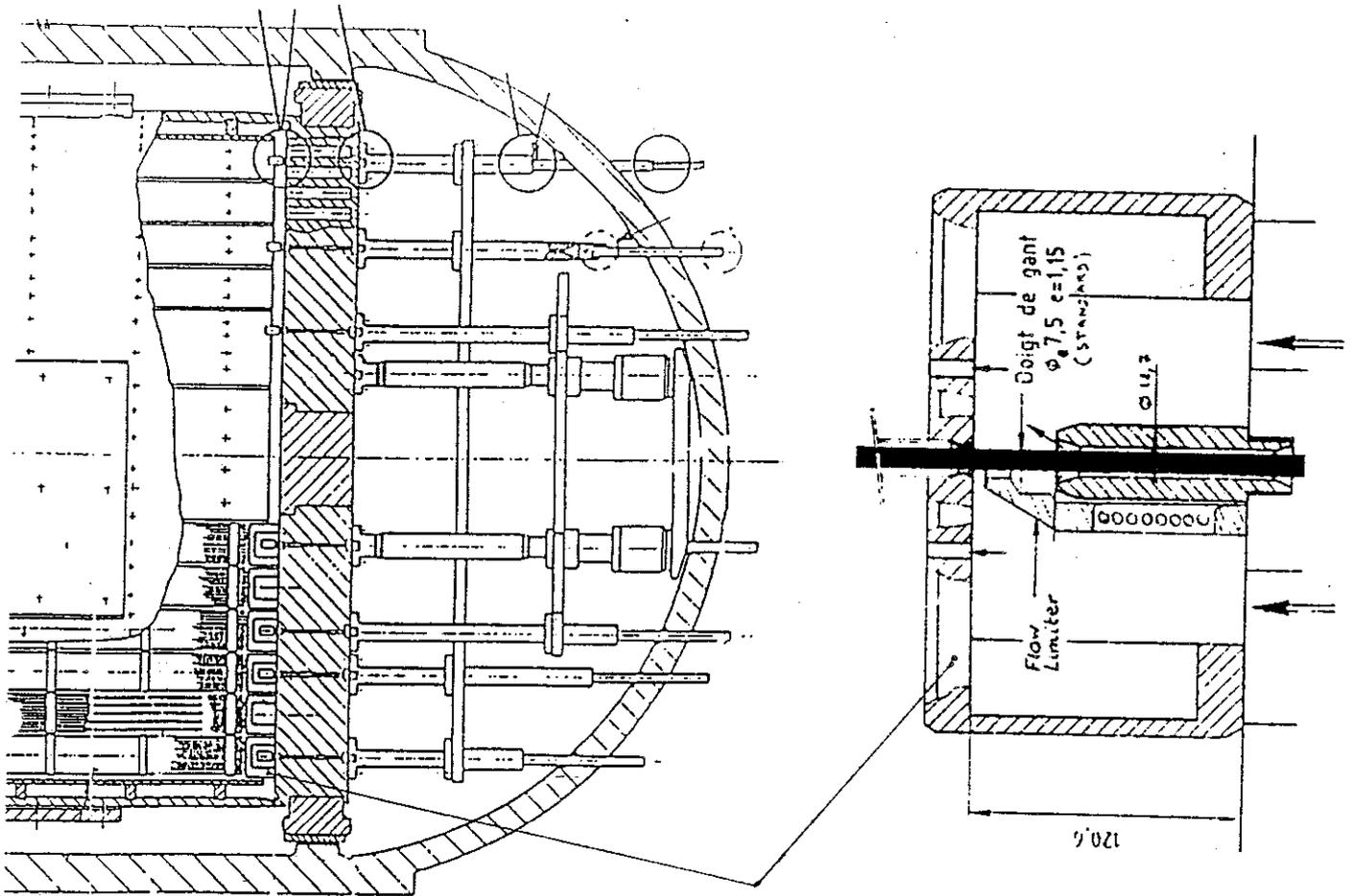
Shielding (vessel pit wall)

Reactor Vessel

CORE

Thimble (in fully inserted position)

Flux detector

Vessel penetration

Guide tube

Thimble (retracted position)

Doigt de gant $\varnothing_e 7,5$ $e=1,15$ (standard)

$\varnothing 4,3$

Flow Limiter

120,6

Fig. 1.

Instrumentation system – General arrangement
and
Incore instrumentation guiding system.
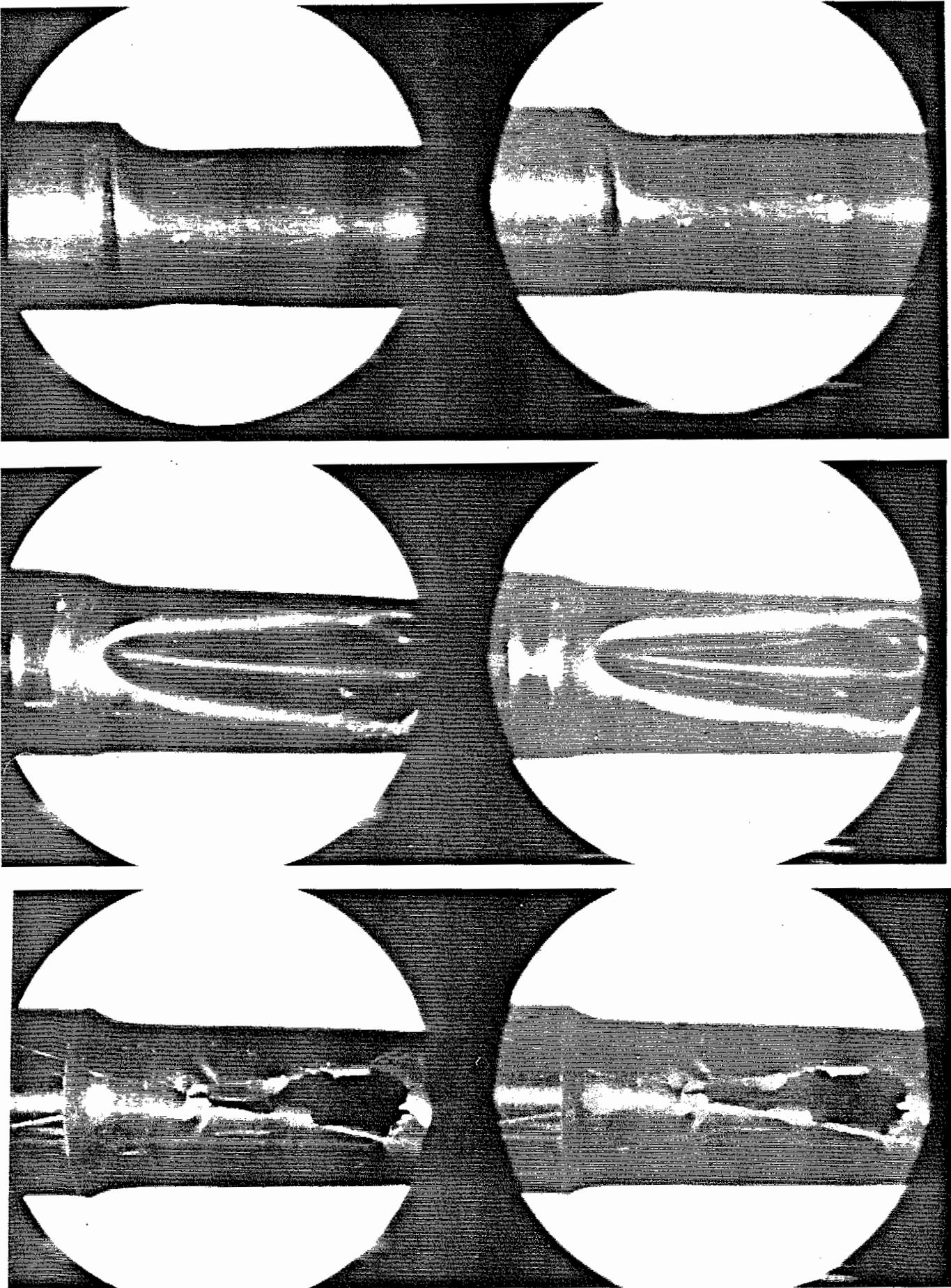
Fig. 2. Thimble tube wear and perforation.

FRACTURE ANALYSIS OF BOLTS OF FUEL ASSEMBLY EDGE CLAMPS

As a result of bolt rupture, the edge clamp and hold-down spring, at the top of fuel assemblies, can be displaced; this might lead to blocking of the control rod movement. Fig. 3 shows the hold down spring clamp system.

An investigation of the fracture aspects of the cracked bolts, and an evaluation of a possible relation of crack length to ultrasonic on-site inspection signal, was made in the hot laboratory.

The following programme was performed :

- macroscopic examination and photography of the bolts in the as-received condition and identification of cracks;

- measurement of the tensile rupture of bolts in order to 1)permit an estimation of the in-service crack size, 2)reveal possible cracks in apparently intact bolts and determine the rupture mode of intact bolts;

- macroscopic examinations and stereophotography of the fracture surfaces;

- detailed fracture surface analysis by scanning electron microscopy (SEM) and EDX analysis of cracked bolts ;

- metallographic analysis of a longitudinal section of a crack-free bolt.

Fig. 4 shows some examination results.

The main conclusions of this investigation can besummarised as follows :

- intergranular stress corrosion cracking (SCC)is responsible for the observed damage in the Inconel 600 bolts in primary cooling water at operating temperatures;

- residual surface stresses induced by the peculiar bolt manufacturing techniques at (and close to) the head-shaft connection, are the major source of the stresses required for the SCC mechanism;

- the microstructure of the bolt material is typical of a rather vulnerable Inconel 600 condition;

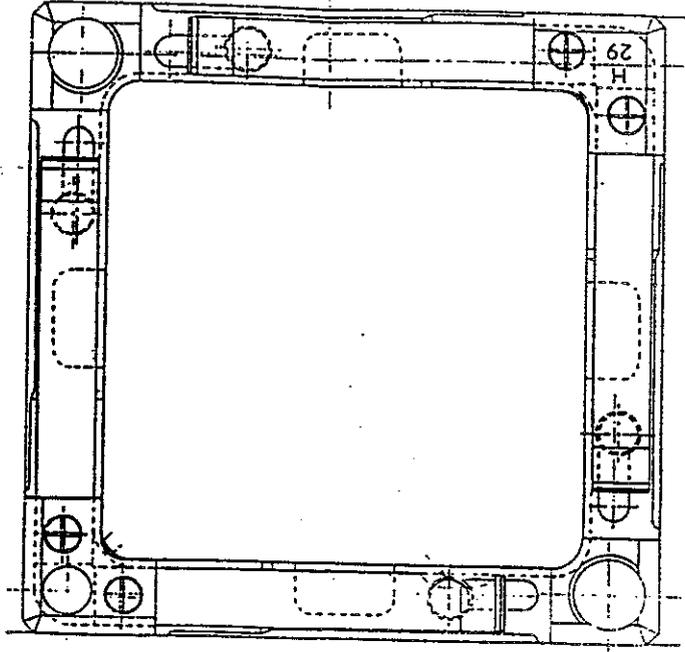- on-site ultrasonic testing is not very efficient to identify cracked bolts.

HOLD DOWN SPRING

FUEL ROD

THIMBLE TUBE

THIMBLE SCREW

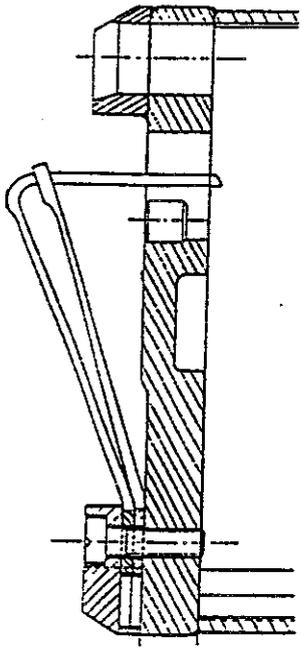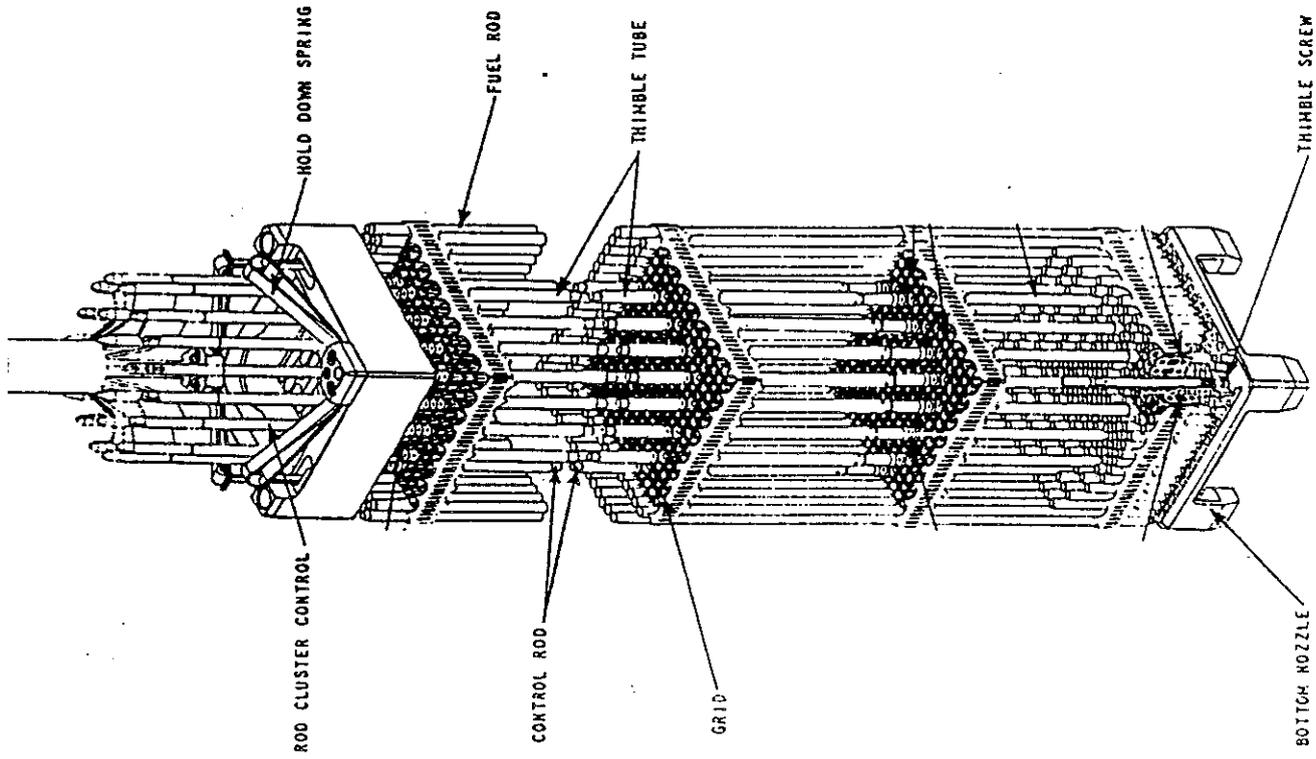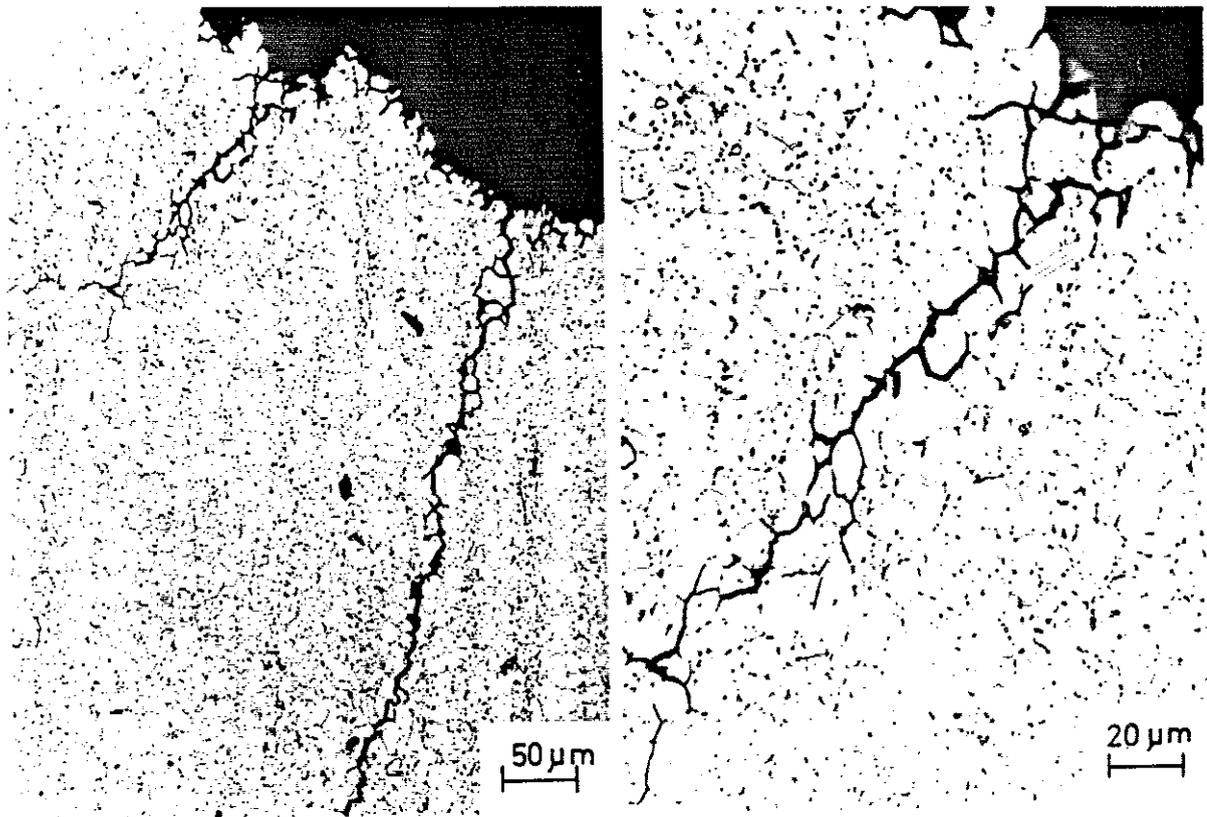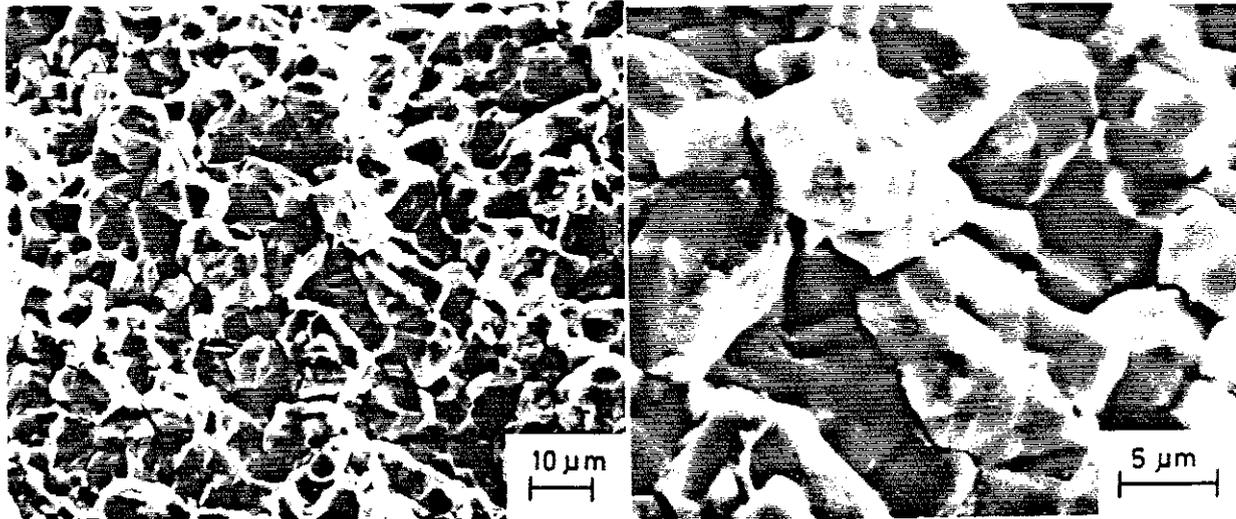ROD CLUSTER CONTROL

CONTROL ROD

GRID

BOTTOM NOZZLE

H 29

Fig. 3. Fuel assembly and hold down spring clamp system.

Bottom : Branch of in-service crack, mixed
nital-phosphoric acid etch. longitudinal
banding is visible at lower magnification.

deposit-free intergranular in-service fracture

Fig.4. Microstructure and SEM examination.

FAILURE INVESTIGATION OF FLEXURE PINS IN INCONEL X750 MATERIAL

The considered flexures are part of a system – housing removable inserts and retainers – installed to guide the drive rod of the control rod cluster assembly. Fig. 5 shows the control rod hardware and flexure system.

In several PWRs, extensive cracking has been observed in a number of these flexure pins.

A first evaluation of the fracture mechanism has been made by detailed fractographic analysis in the hot laboratory, including macroscopic examinations and extensive SEM examination of fractured as well as cracked flexure heads; this was supplemented by detailed metallographic and microprobe analysis, as well as macrohardness measurements.

Fig. 6 illustrates the observed cracks and microstructure analysis.

All observations indicated that failure has been caused by the slow progression of a brittle intergranular crack, with no indication typical of a fatigue mechanism.

Stress corrosion cracking was therefore proposed as the failure mechanism.

It was found that initiation and growth of the cracks occurs in the shaft-flexure head transition.

Potentially aggressive elements, such as Cl and S, have been detected at the fracture surface; the origin of these contaminants was however not clear.
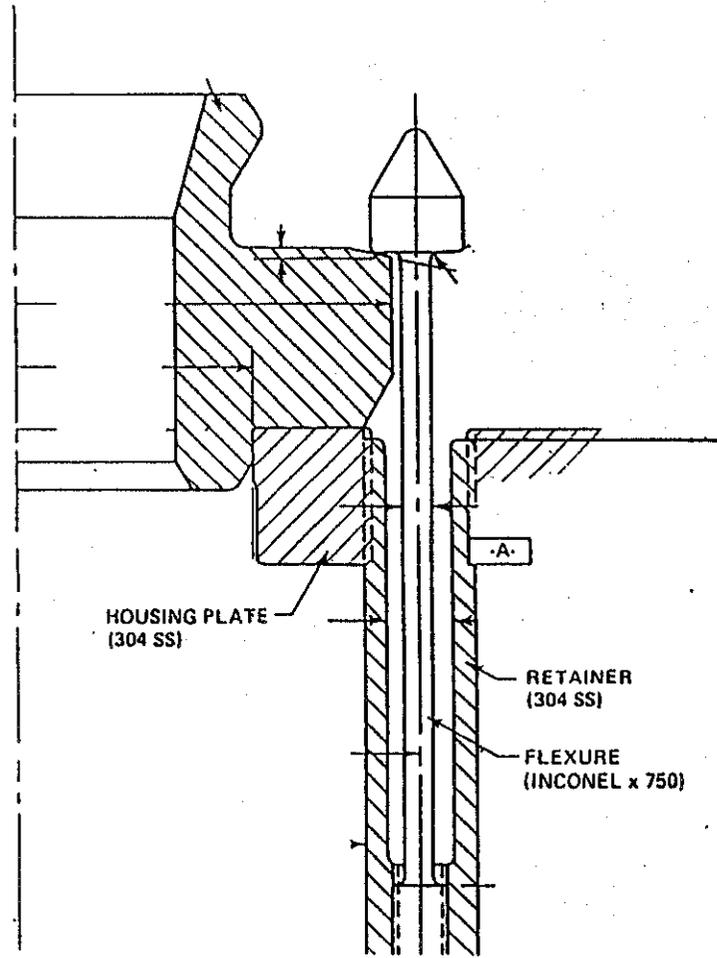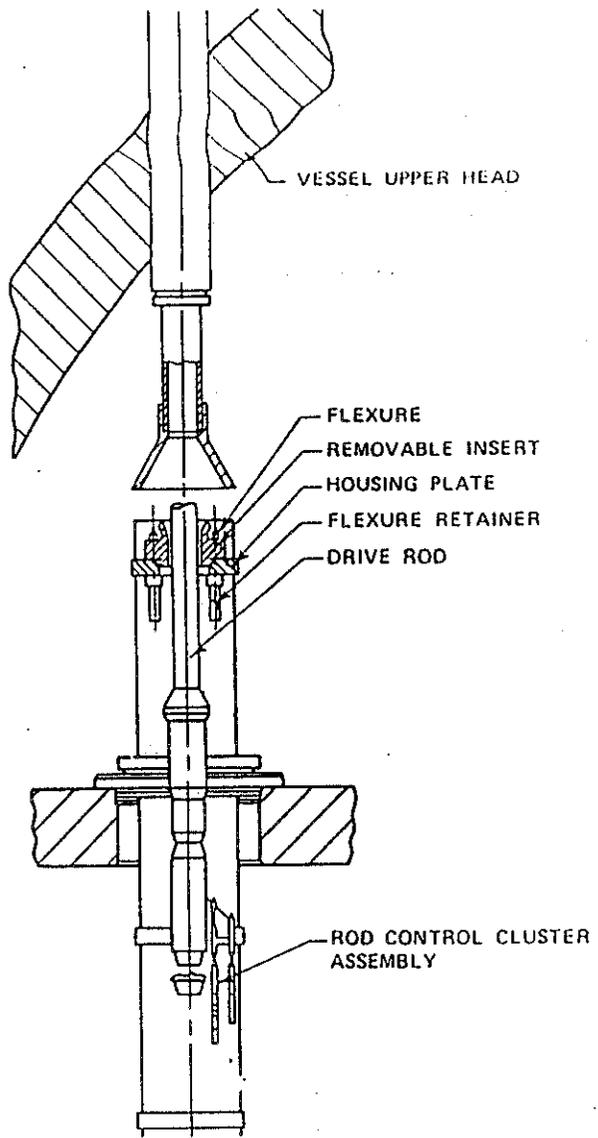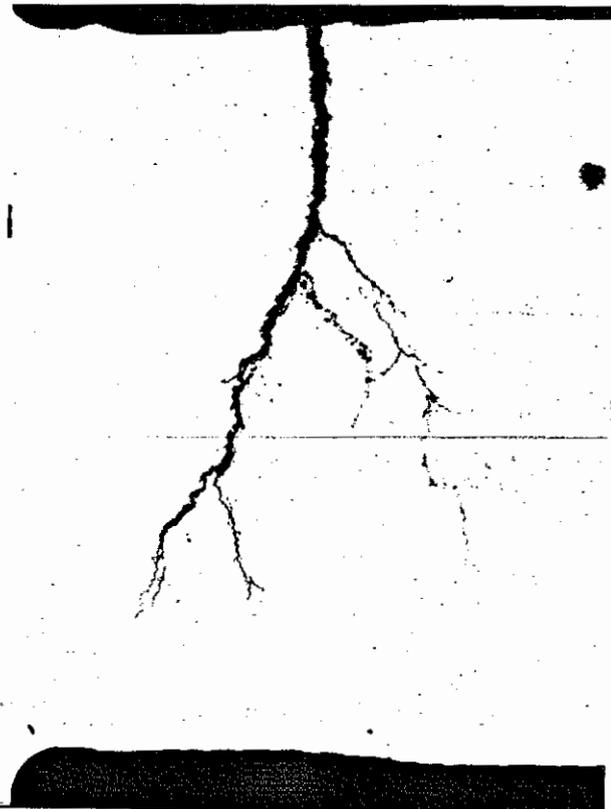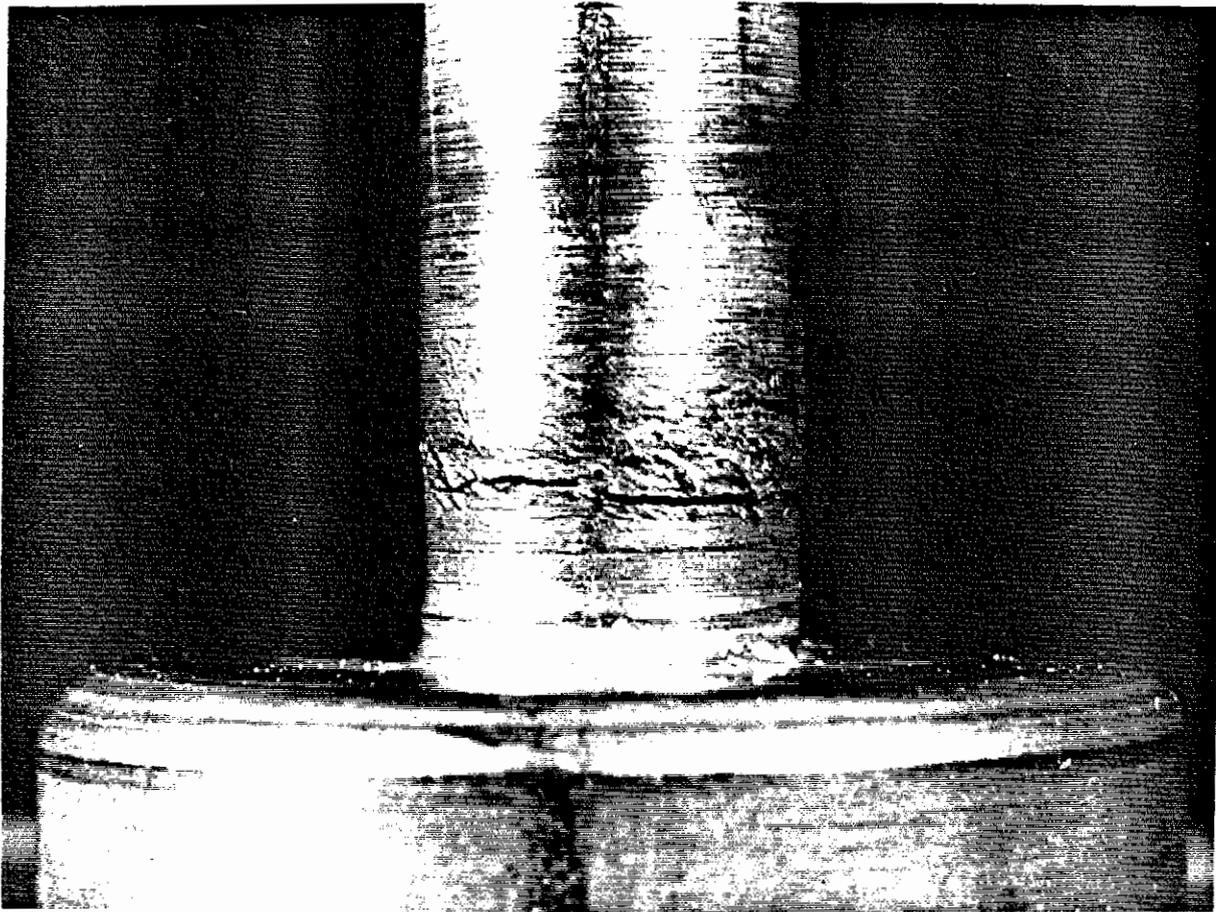
VESSEL UPPER HEAD

FLEXURE
REMOVABLE INSERT
HOUSING PLATE
FLEXURE RETAINER
DRIVE ROD

ROD CONTROL CLUSTER
ASSEMBLY

HOUSING PLATE
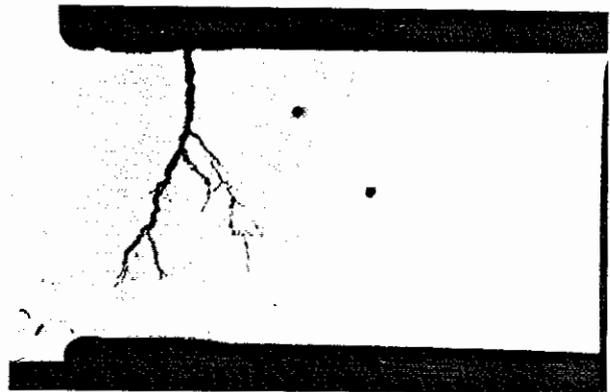(304 SS)

RETAINER
(304 SS)

FLEXURE
(INCONEL x 750)

·A·

Fig.5. Control rod hardware and flexure system.

Fig.6. Flexure pin cracking and microstructure
analysis.

DEFECT ANALYSIS OF CONTROL ROD ASSEMBLY

In some of the older PWRs, a cruciform array of cylindrical rods of absorbing material was adopted as reactivity control system. See Fig.7. The absorber section of these rods is made of extruded bars drawn from a Ag 8% - In 15% - Cd 5% alloy clad in cold-worked type 304 stainless steel tubing.

In this design, the stainless steel serves as a load-carrying component; it encapsulates the absorbing material, without metallurgical bonding. The assembled tubes are placed in a jig and ferrules are electron-beam welded to them at about 250 mm elevations.

After a certain number of reactor operation cycles, typical degradation of these arrays has been observed.

Hot cell examinations based on detailed overall visual inspection, metallography, microprobe and SEM analysis of a full size cruciform array, 2980 mm in length with an overall cross-section of about 200 mm, revealed the following degradation.

Wear related defects were observed at the lower part of the array, caused by displacements and vibrations of the control rod in contact with the adjacent fuel assemblies. Fig. 8 shows some rod perforations. In supplement to these fretting features however, cracks have been observed in the weldings of the ferrules against the rods at different levels, as well as longitudinal cracks in rod segments. See Fig. 9.

These defects have been related to constraints resulting of vibration and of interactions between the absorbant and the cladding.

Irradiation-induced swelling of the absorbant, as well as plastic deformation due to the axial impact forces during the insertions and extractions of the control rod, promote stress increase in the cladding. Embrittlement of the cladding at the lower part of the array is also to be considered as contributing to the cracking processus.

Metallographic- and SEM examinations have revealed typical inter-granular cracks, initiating at the outer surface of the cladding.
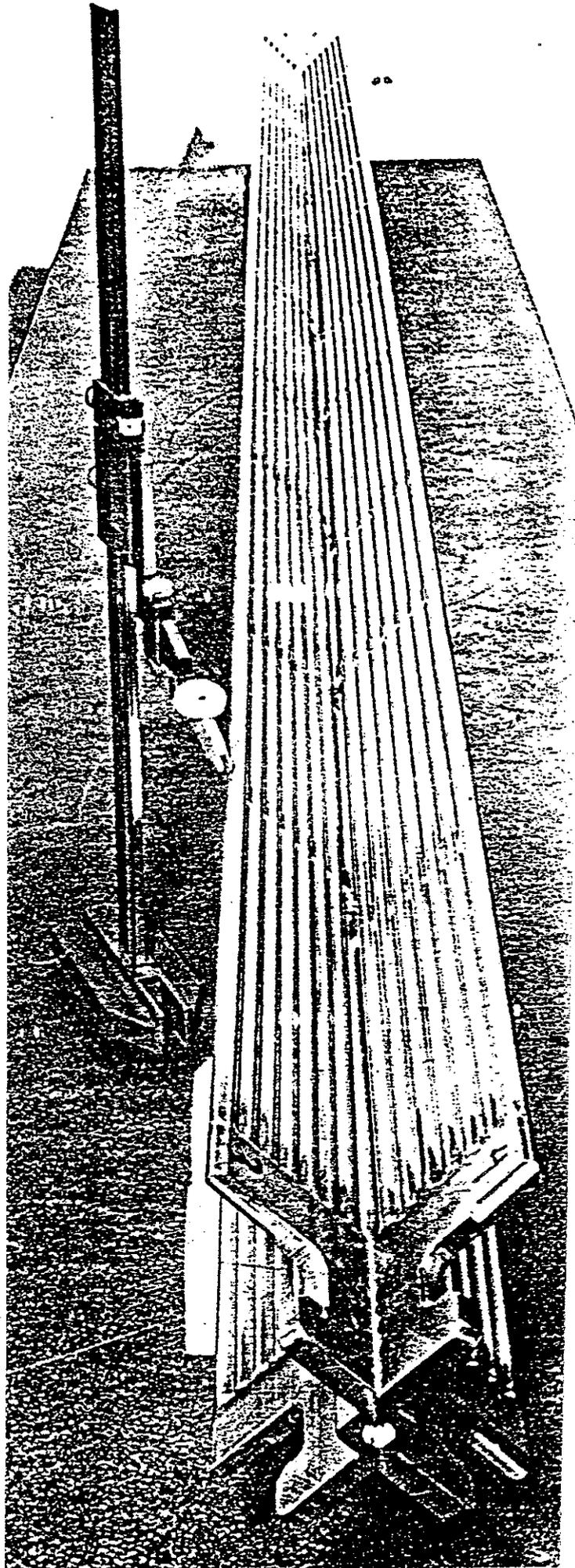
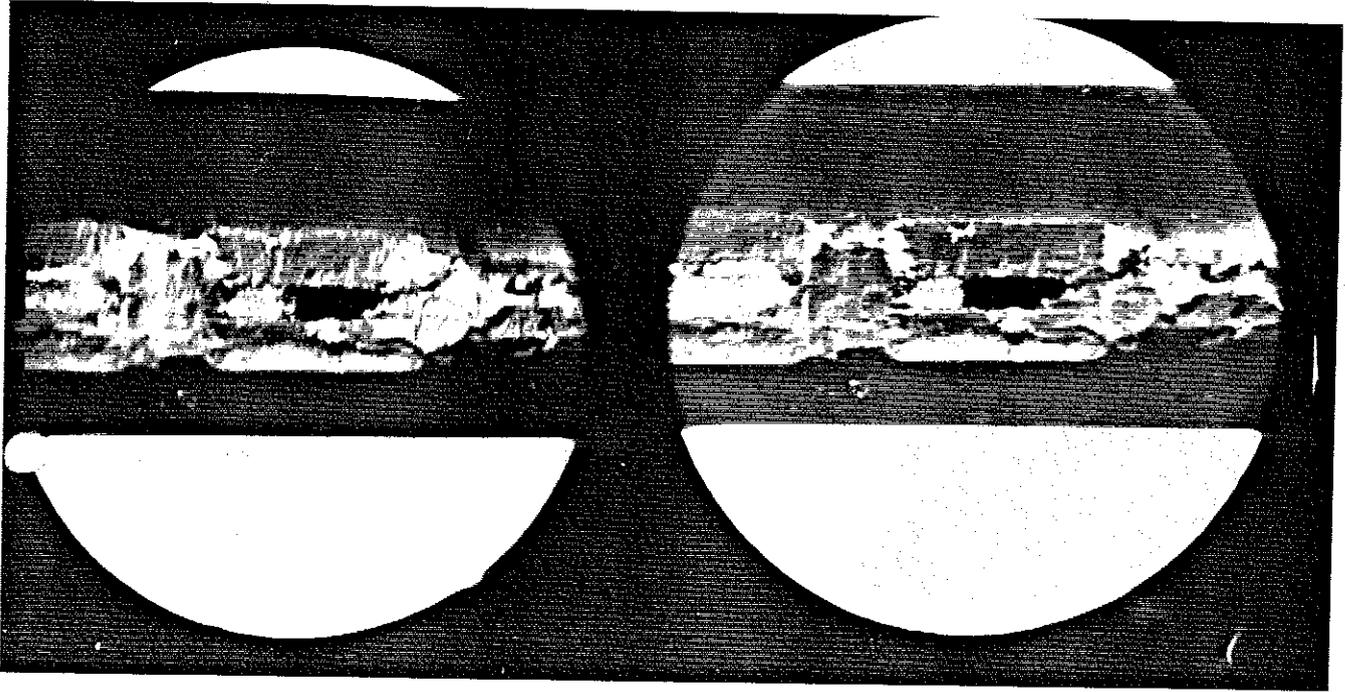Fig.7. Cruciform array of cylindrical rods.
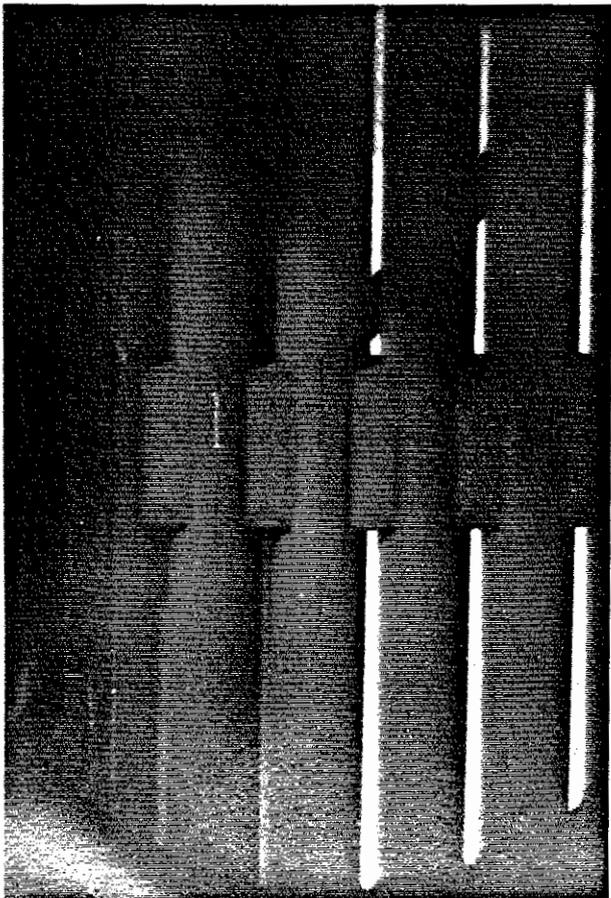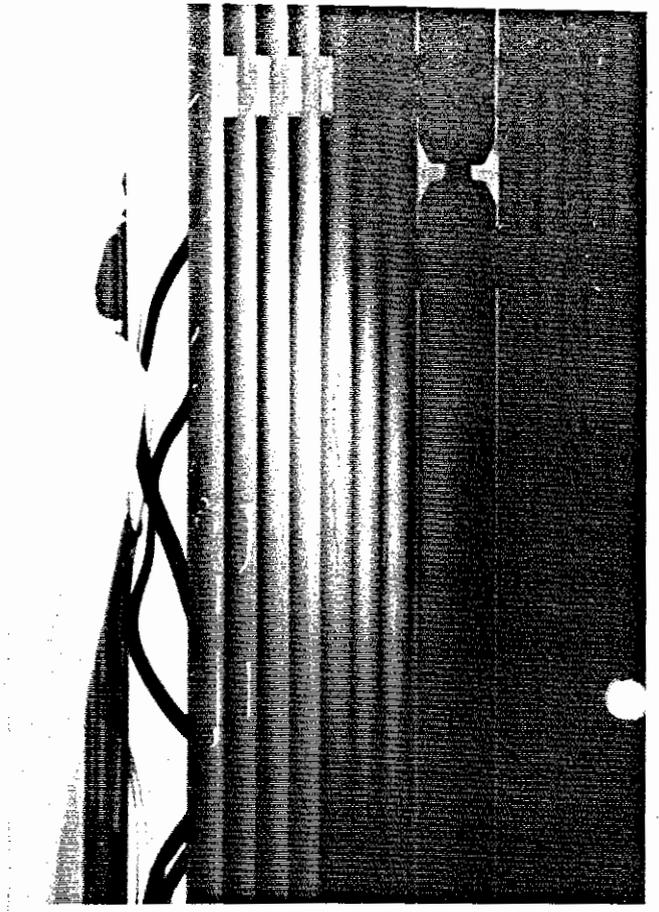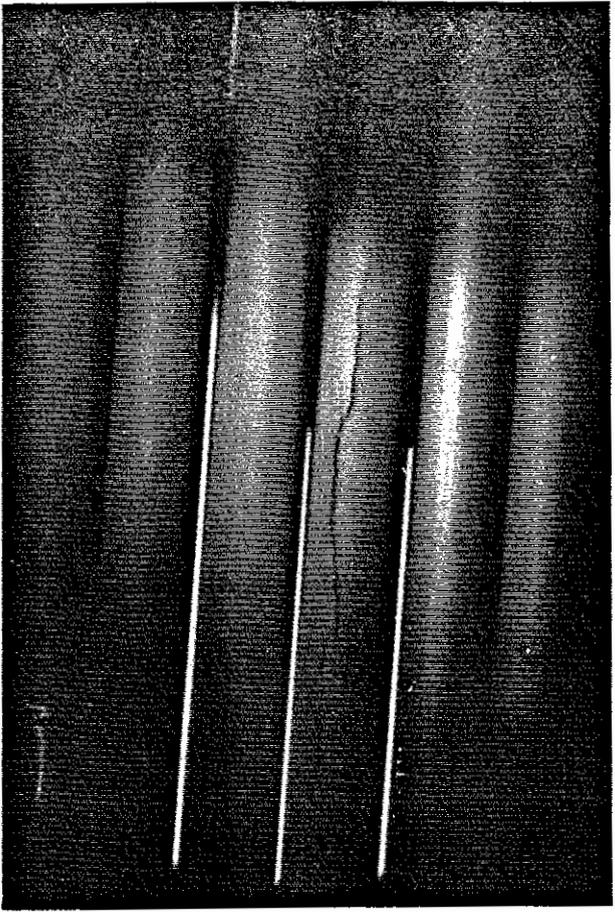
Fig. 8. Wear and rod perforations.

Fig.9. Cracks in welded ferrules and longitudinal cracks in rods.

UP-FLOW CONVERSION

In order to overcome the problem of water jetting between plates of the core barrel (stepped box structure surrounding the core of square fuel assemblies), an up-flow conversion has been applied in several European PWRs, resulting in an upward flow of coolant water through the barrel structure.

This operation however necessitates : 1) the plugging of holes around the top of the core barrel, where the flow of coolant water enters the structure, 2) the drilling of holes around the top former of the barrel, to allow an upward flow to emerge above the core.

This implies first that all holes drilled through the thermal shield, be aligned with the holes in the concentric core barrel; second, specially designed plugs must be fit through these holes in order to block them off. Fig. 10 illustrates the operations involved.

Drilling operations were performed by electro-erosion; stainless steel blocks extracted out of the thermal shield have been dispatched to the hot cells for further investigation.

The main purpose of the hot cell examinations was to identify a certain degree of degradation over the thickness of the stainless steel plate and to look for gradients.

So, out of the massive blocks, different test specimens have been manufactured. See Fig. 11.

Examinations were related to tensile testing, hardness measurements and dosimetry of the material.

Sulphur content was determined as well.

Finally, besides a small metallographic examination, workshop-operations were performed (turning, milling, drilling) in order to check changes in machining suitability.
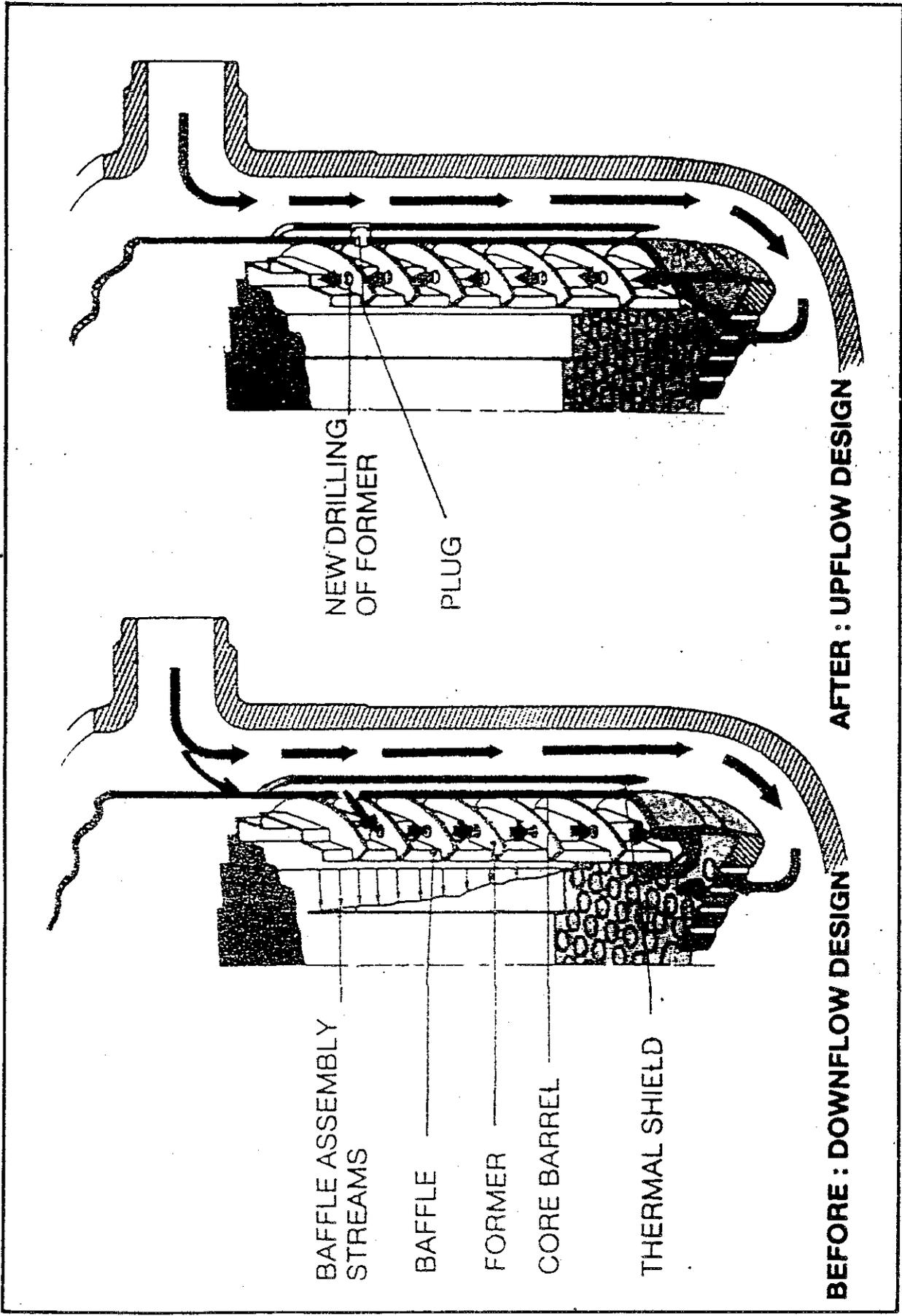
NEW DRILLING
OF FORMER

PLUG

AFTER : UPFLOW DESIGN

BAFFLE ASSEMBLY
STREAMS

BAFFLE

FORMER

CORE BARREL

THERMAL SHIELD

BEFORE : DOWNFLOW DESIGN

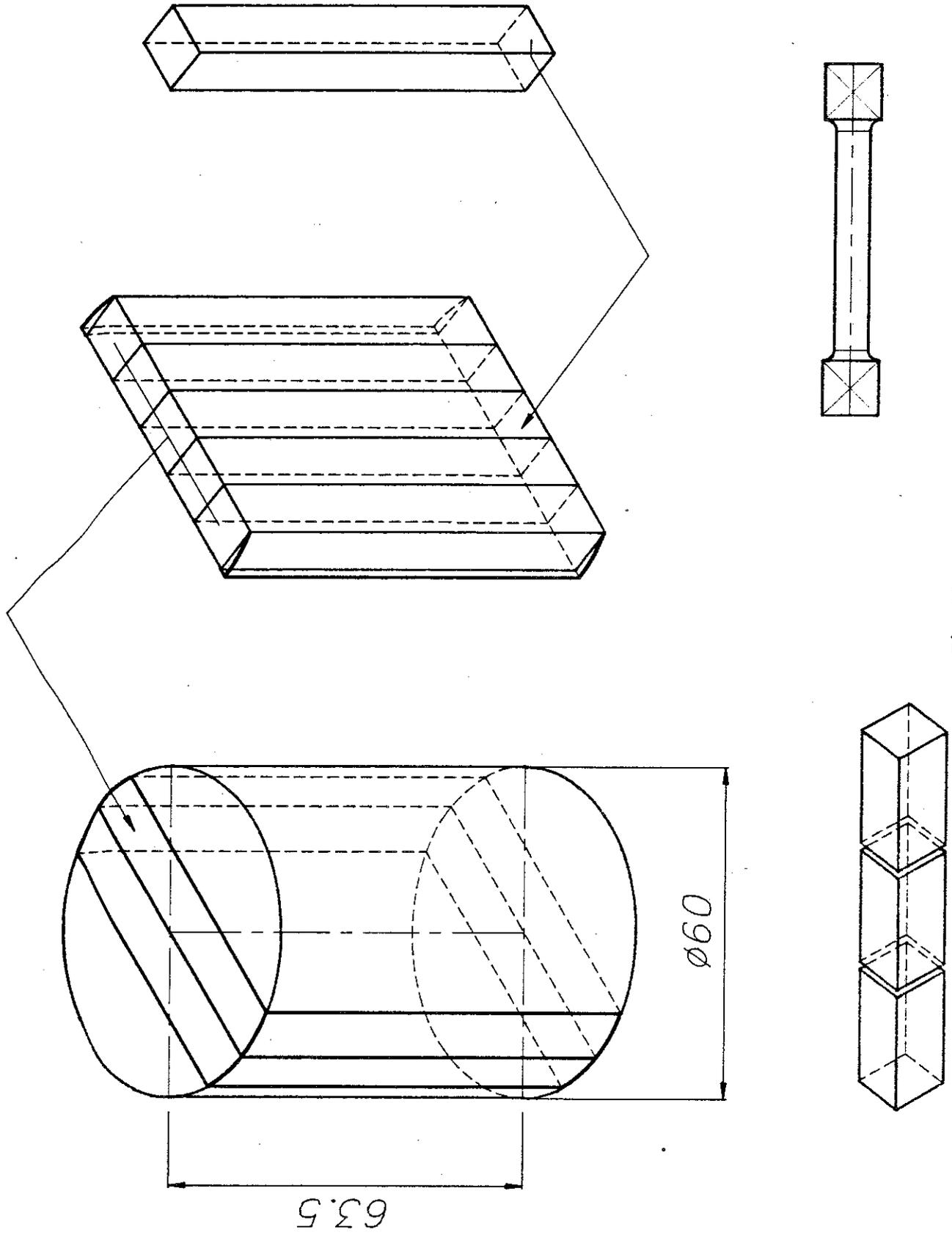Fig. 10.   Results of an upflow conversion

Fig.11. Thermal shield steel block and specimen conditioning.

Ø60

63.5

EXAMINATIONS OF A STEEL SAMPLE EXTRACTED FROM THE OUTER SURFACE OF THE SAFE END RING OF THE PRESSURE VESSEL

Severe surface cracking was observed at the outer surface of a ring segment at the cold leg safe-end of the pressure vessel. A small sample was mechanically removed from the damaged area and transferred to the LHMA for detailed analysis.

The objectives of the examinations were, besides material identification, evaluation of the damage and identification of aggressive species in the damaged areas.

Examinations revealed an extensive intergranular network of cracks, showing equiaxed microstructure with a very large grain size and numerous recrystallisation twins and glide bands.
Fig. 12 illustrates the crack pattern at the outer tube surface.

The material used for the safe-end ring was definitely AISI 316 in the fully-annealed condition; this material can very easily be sensitized, probably during welding of the ring to the reactor vessel in the cold leg duct.
Residual stresses combined with service stresses may have promoted the initiation and extension of the damage.
Aggressive chemical substances necessary to cause the damage of the sensitized material could not be revealed; damage in pre-service aggressive environment cannot be ruled-out.
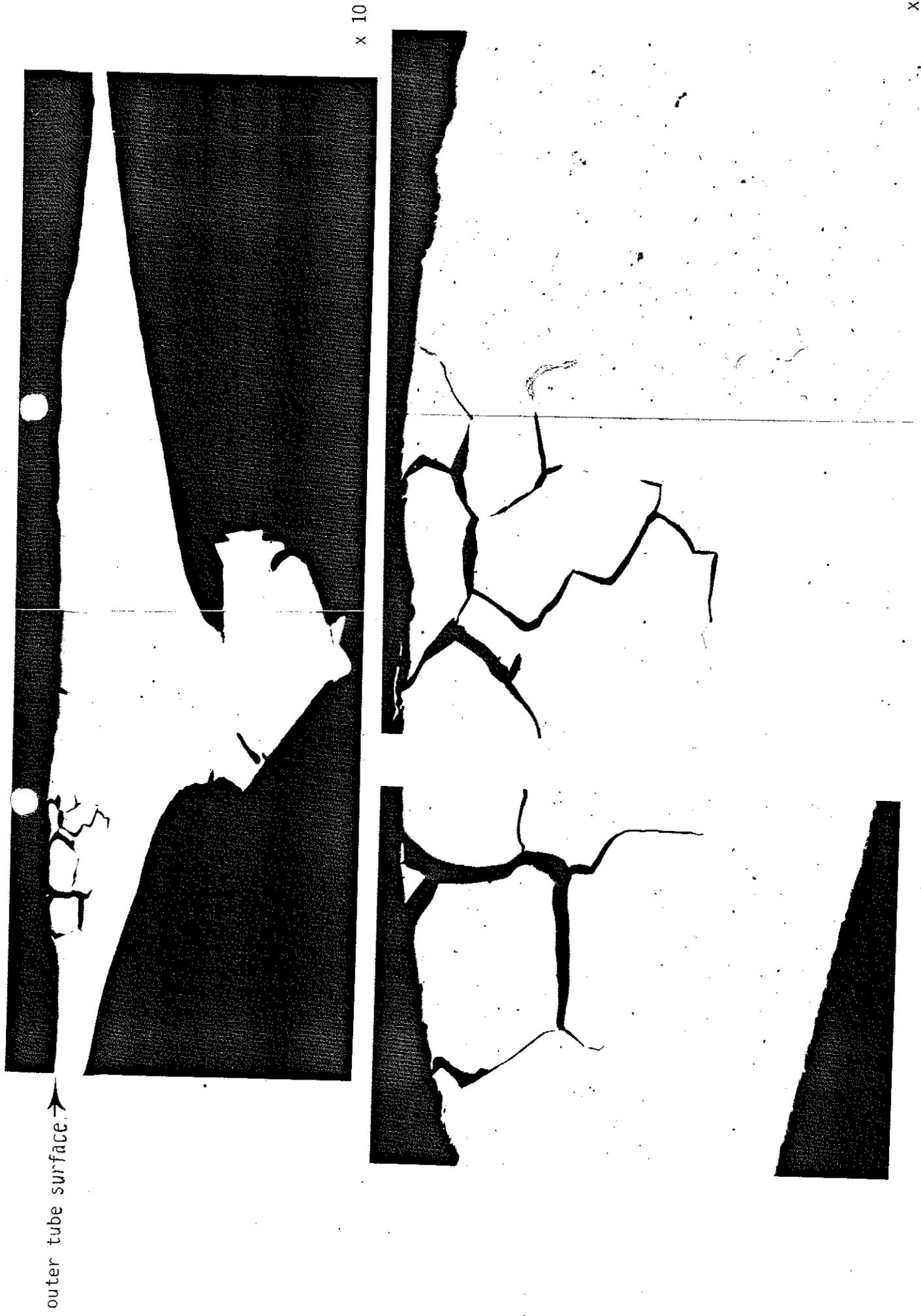
outer tube surface →

x 10

x 50

Fig.12. Section M2 normal to the tube surface. Crack pattern at the outer tube surface.

METALLURGICAL AND CHEMICAL EVALUATION OF TUBES FROM BELGIAN NUCLEAR POWER PLANT STEAM GENERATORS

Primary side stress corrosion cracking, as well as secondary-side degradation of the inconel 600 tubes of steam generators were observed; this required an important examination programme to be initiated.

Since identifying the cracking, an important number of tubes were expanded beyond the original rolled region in order to reduce the primary-to-secondary leakage; further more, in an effort to establish a permanent repair for the roll transition cracking, a kinematically bonded mini-sleeving operation was performed.

After several weeks of operation however, excessive leakage was observed again.

Finally, several steam generator tubes were removed from the hot leg and cold-leg sides of the steam generator and brought to the laboratory for analysis and evaluation.

The objectives of the programme were as follows :

- evaluation of the secondary side IGSCC occurring in the tube sheet crevice;
- evaluation of the primary side roll transition cracking;
- evaluation of the regions affected by the re-rolling and mini sleeving techniques.

After detailed visual inspection and identification by surface analysis of the probable causative species, a complete metallographic charac-terisation of the different tube defects was performed, followed by microprobe and SEM/EDX-analysis.

Fig. 13 & 14 illustrate some of the analysis.

The main conclusions are :

- microstructure and grain size, i.e. tube processing temperature, has an important influence; this confirms the effect of the annealing temperature range on the susceptibility to IGSCC;
- secondary side cracking, however, seems to be determined by the crevice chemistry and associated formation of particular surface layers, rather than by the material microstructure;

- the role of Si, Al, S and oxygen found as major elements in the outer tube surface layers is not very clear in relation to the secondary side stress corrosion cracking process;
- extensive circumferential cracking was observed in the case of an improperly applied sleeve. Limited circumferential cracking has also been seen under the lower unwelded sleeve end. Metallography and the overall geometry of these defects seem to indicate that these have been mechanically induced during the sleeving operation.
- a substantial hardness increase of the tube material and the development of small circumferential cracks at the lower-sleeve end appear to be the two major phenomena resulting from the mini-sleeve application.

400 μm

20 μm
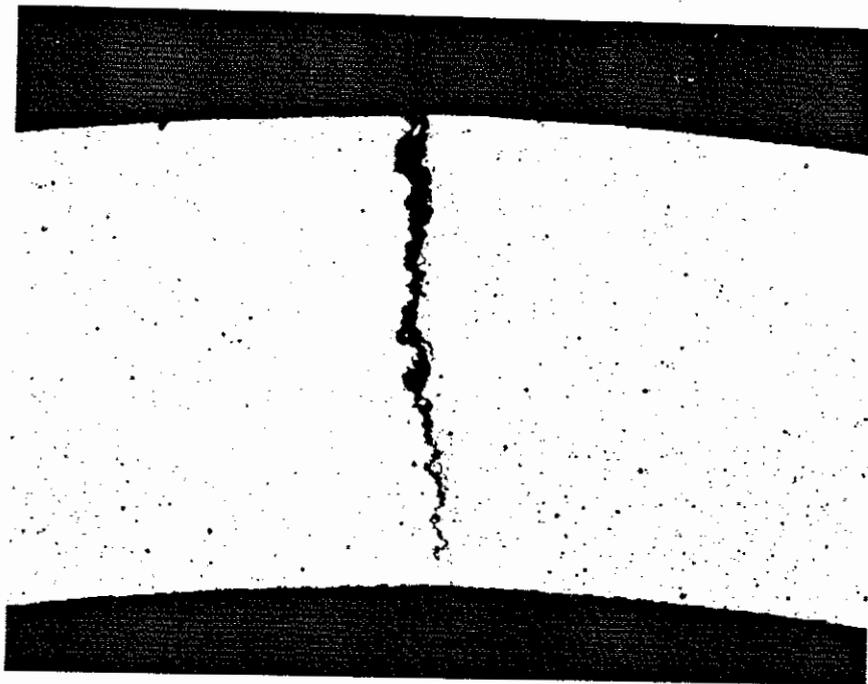
Fig.13. Secondary side longitudinal crack
        Top   : general view
        Bottom : detail of crack tip and inner tube surface

Fig.14a.

Circumferential cracks initiated at lower sleeve end of tube.

1 mm

40 μm
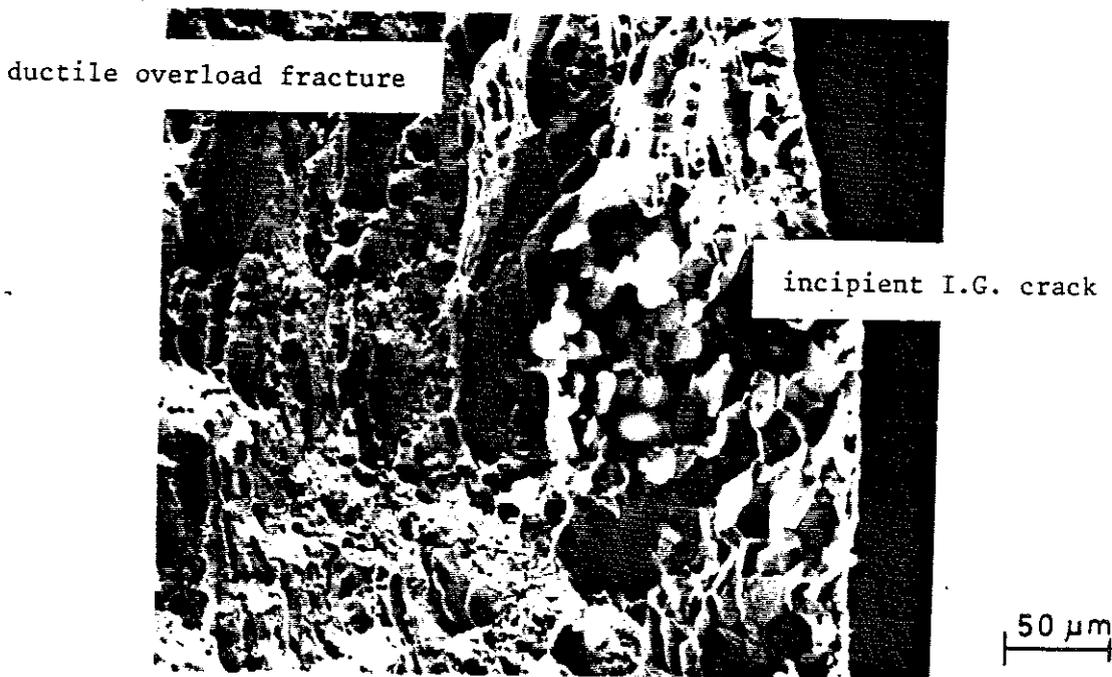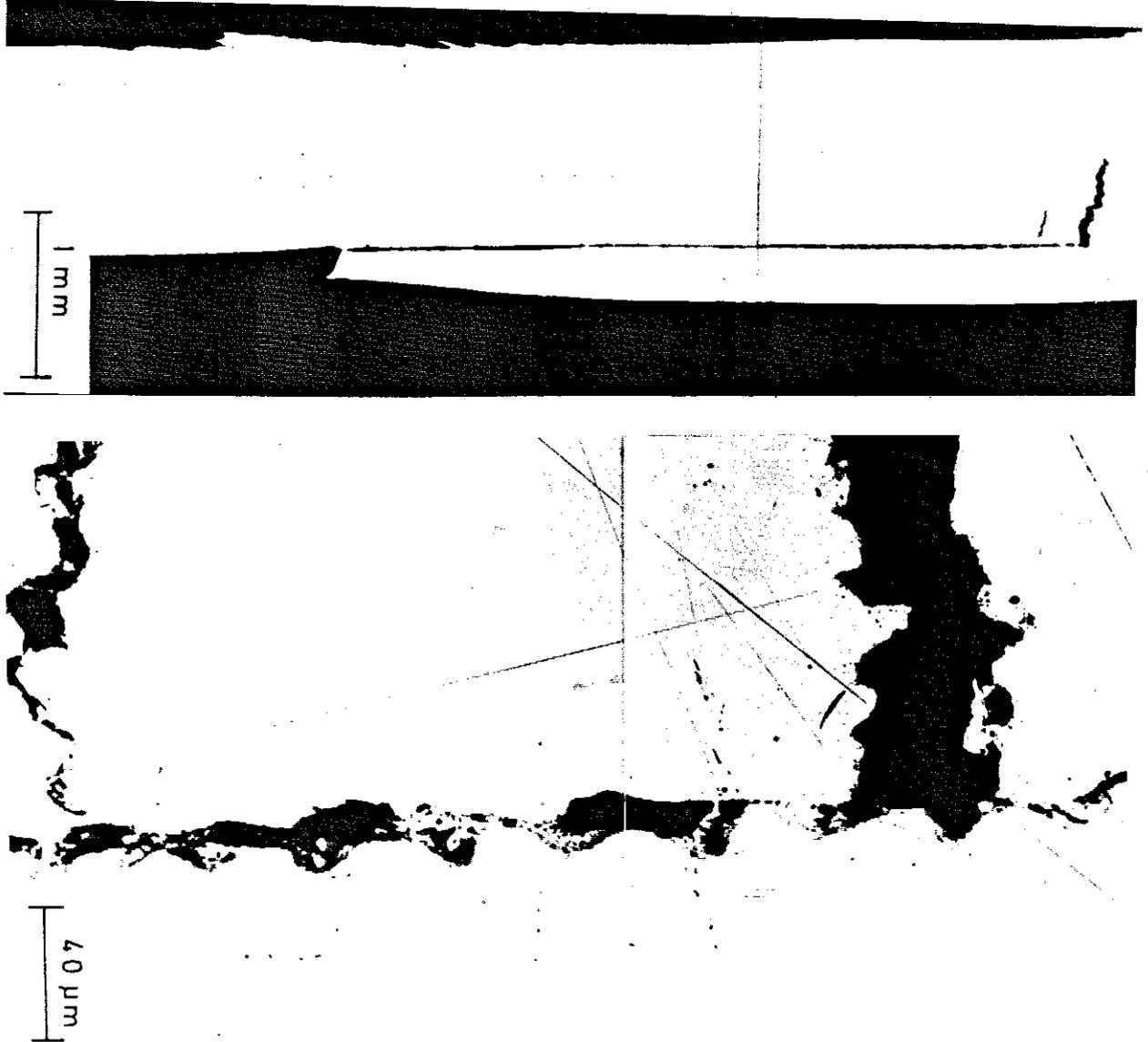

ductile overload fracture

incipient I.G. crack

50 μm

Fig.14b. Tensile rupture of tube
The overload fracture meets semi-circular I.G. cracks at the
outer tube surface. At higher magnification loose particles
can be seen in these areas.

EXAMINATION OF PRIMARY PUMP PARTS

Important surface cracking has been observed in a bush of a primary pump.

Die penetrant control revealed a lot of cracks of about 20 mm length, almost impossible to see with the naked eye, and all axially oriented at equidistance from each other. See Fig. 15a.

Although the contamination level was rather low, the hot laboratory was asked to examine in more detail the observed degradation of that particular bush of AISI 304 austenitic stainless steel.

Stereomacroscopic inspection of the cracks (Fig. 15b) followed by a material structural examination gave evidence of transgranular stress corrosion cracking (TGSC).
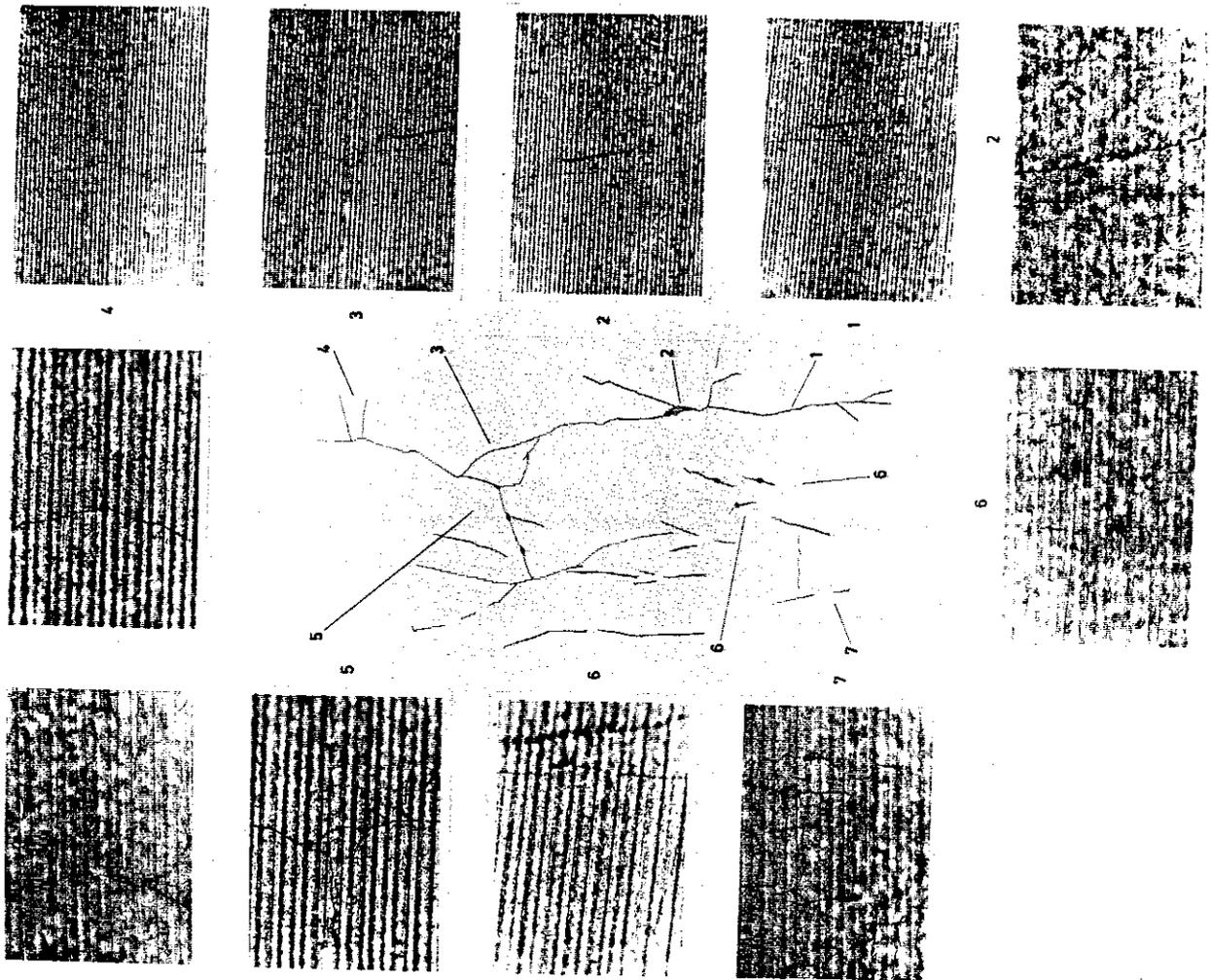
CRACK PATTERN

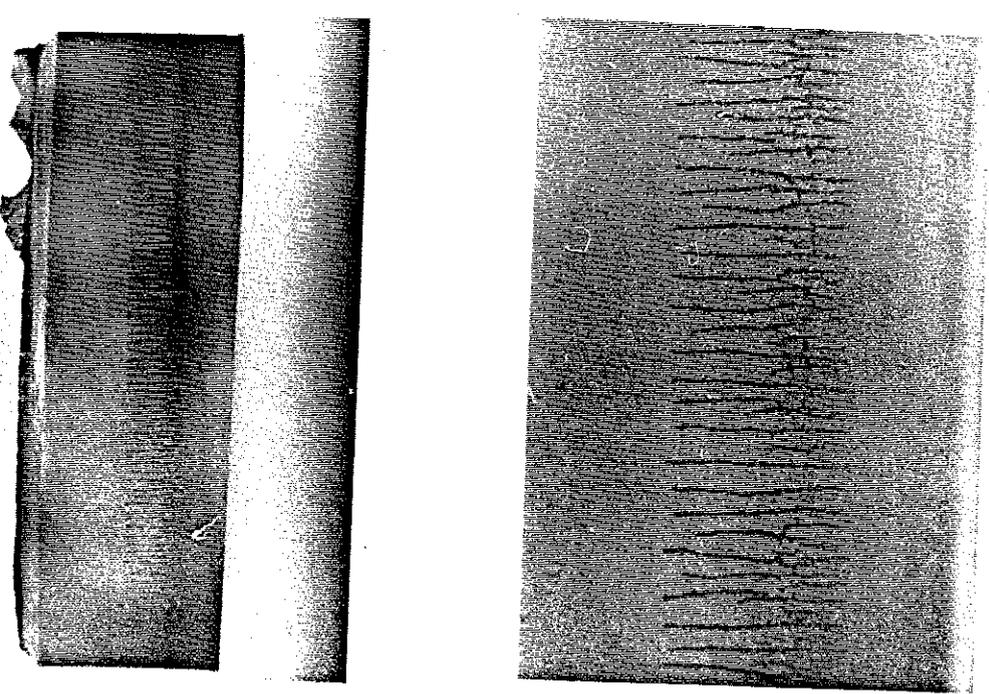Fig.15b. Stereomacroscopic examination.

Fig.15a. Die penetrant revealed axial cracks.

EXAMINATION OF CORRODED PIPE SECTIONS OF 304 L STEEL OUT OF THE POWER PLANT WATER TREATMENT SYSTEM

Different corroded and contaminated pipe sections extracted of the water treatment circuit were dispatched to the hot laboratory to examine for possible corrosion causes.

Pipe sections - a 3/4" diameter pipe welded to a 4" diameter pipe - have been handled in glove boxes; after specimen cutting and conditioning, metallographic examinations and SEM analysis followed in the shielded installations.

Metallographic examinations revealed pitting corrosion on the surface of the tube sections. Cracks initiate at these pitting corrosion sites and corrosive products are present in the microcracks. See Fig. 16.

SEM and EDX examination of overload- and corroded fracture surfaces indicate the presence of Cr, K, Ca, Cl, S and traces of Si.

The main conclusion of this investigation is :

- pitting corrosion and cracking are caused in the conjunction with the presence of local residual stresses induced by heavy cold work of the tube wall material and the welding structure.

Cracking at the periphery of the opening in the
4" diameter tube
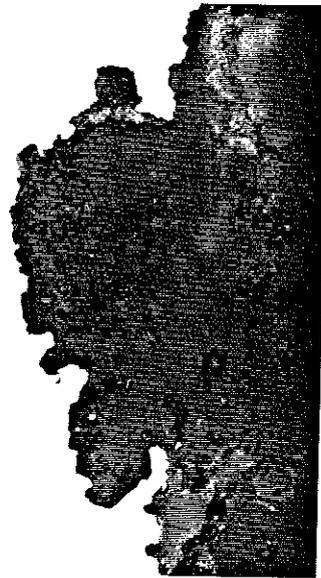
Pitting corrosion of inner tube wall

Fig.16. Pitting corrosion and cracking aspects.

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Fracture analysis of bolts of edge clamps of fuel assemblies

Fracture analysis of flexure pins
TEC/39.Z1911/23/HT/JK/JVdV/CVL

Examen d'une barre de contrôle de réacteur
TEC/39.Z1930/90/JVdV/JK/CVL/AG

Programme "Up-flow" carottes d'acier
Essais sur matériaux irradiés
TEC/39.Z1945/36/JVdV/CVL/JK

Examinations of a steel sample extracted from the outer surface of the
safe end ring of the pressure vessel
TEC/39.B3236/55/HT/JVdV/CVL/JK

Metallurgical and chemical evaluation of tubes from a steam generator
SCK/CEN - EPRI NP-5022-LD

Examinations of corroded tubes sections in 304 L steel out of power
plant water treatment system
TEC/39.B3002/10/JK/JVdV/CVL

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