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NUCLEAR RESEARCH CENTER

APPLICATION OF NON-DESTRUCTIVE TESTING AND IN-SERVICE INSPECTIONS TO RESEARCH REACTORS AND PREPARATION OF ISI PROGRAMME AND MANUAL FOR WWR-C RESEARCH REACTORS

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MOHAMED KHATTAB

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CONTENTS

	Page
ABSTRACT..	I
INTRODUCTION..	1
HISTORICAL BACKGROUND...	2
PHYSICAL DESIGN PARAMETERS...............................	3
FUEL SPECIFICATIONS...	3
REACTOR VESSEL MATERIAL AND INTERNALS.......	4
PRIMARY COOLING SYSTEM.......................................	7
SPENT FUEL STORAGE POOLS..................................	16
WATER CHEMISTRY..	17
ENHANCEMENT OF REACTOR SAFETY.......................	17
REFERENCES...	19

ABSTRACT

The present report gives a review on the results of application of non-destructive testing and In-service Inspections to WWR-C reactors in different countries. The major problems related to reactor safety and the procedure of inspection techniques are investigated to collect the experience gained from this type of reactors. Exchangeable Experience in solving common problems in similar reactors play an important role in the effectiveness of their rehabilitation programmes.

INTRODUCTION

Most of the research reactors type WWR-C was designed and built within the period 1955 to 1965 at the time when no special requirements were in force for reactor safety, and one had to follow general industrial recommendations, for example, at the time of the piping's fabrication, lack of depth of fuse penetration in welded seams were allowed,, which is forbidden by the present standards. Since that time, a large number of special standards have been developed, with most of them having been revised after the chernobyl disaster twords ensuring "General Safety Requirements for Research Nuclear Reactors", for example, the reactor vessel should meet the requirement of inherent safety, i.e. the core should remain under water in the case of a rupture in the primary circuit piping. There are naturally many other aspects of reactor operation that have to be included in the evaluation of the allowed service life. Reactor design should be checked for compliance with the regulations in force systematically. One should also carry out systematically the required actions to bring the reactor in fuel compliance with the recommendations. Quality assurance techniques have been applied in the earlier stages of components manufacturing. Preliminary and final safety analysis reports became essential before licensing permit for any nuclear installations. In-service inspection and non-destructive test programs have been developed. The water chemical properties must obey the standard specifications of water condition for nuclear facilities in order avoid corrosion and material degradation. The problems of of ageing, spent fuel storage tanks and decommisioning procedures have been evolved as a resuslt of the expanding operational history.

The present report study the problems related to reactor safety evolved during more than thirty years operation of the WWR-C research reactors in different countries.

2- Historical Background

Table 1 Illustrates general historical information about WWR-C reactors. Most of them started with low power 2-3 MW and then upgraded during the last two decades.

Table. 1
General Information.

No.	Country	Facility		Construction Start	Criticality Date	AGE Years	Reactors Power MW
		Ref. No.	Name				
1	Czech	CZ-0003	LWR-15 Rez	1955/01	1957/09	39	10
2	Egypt	EG-0001	ET-RR-I	1958/03	1961/02	35	2
3	Hungary	HU-0002	Budapest R.R.	1956/05	1959/03	37	10
4	Poland	PL-0001	EWA	1955	1958/06	38	10
5	Romania	RO-0001	WR-S Bucharest	1956	1956/07	39	3
6	Russian	RU-0002	WWR-2 Moscow	1950	1953/07	43	3
7	Russian	RU-0008	WWR-M Gatchina	1956	1959/12	37	18
8	Russian	RU-0019	WWR-TS	1961	1964/10	32	12
9	Ukraine	UA-0001	WWR-M Kive		1960/12	36	10
10	Uzbek	UZ-0001	WWR-CM Tashkent	1957	1959/09	37	10

3- Physical Design Parameters

Table 2 illustrates the physical design parameters of each reactor

Table 2. Physical Parameters

Ref. No.	Thermal Max Flux n/cm^2-s	Fast Max Flux n/cm^2-s	Horizontal Max Flux n/cm^2-s	Vertical Max Flux n/cm^2-s	No. Horizontal Channels	No. Vertical Channels	Core Irradiation Facilities	*
Cz-0003	1.0E +14	1.5E +14						
EG-0001	1.5E +13	3.6E +13	6.0E8	2.0E1	9	8	1	
HU-0002	2.5E +14	1.0E +14	2.0E9	1.6E14	10	35	28	19
PL-0001	1.5E +14	1.3E +14	2.5E9	1.0E14	10	51	7	27
RO-0001	1.0E +13							
RU-0002	4.0E +13							
RU-0008	4.0E +14	1.5E +14	1.4E14	4.0E14	17	17	40	17
RU-0019	1.8E +14	3.3E +14						
UA-0001	1.5E +14	7.0E +13						
UZ-0001	2.3E +14	1.0E +14						

** Reflector irradiation facilities*

4- Fuel Specifications

Table 3 shows fuel specifications of each type

Table 3. Fuel Specifications

Ref. No.	Enrichment %	Tubes per element	Type	Equilibrium Core Size	Burnup %		Spent Fuel Storage Capacity
					Max	Average	
EG-0001	10	16	EK-10	46	25	20	60
HU-0002	36	3	WWR-SM	228	55	50	752
PL-0001*	36	3	WWR-SM		60	40	2400
RU-0002	10						
RU-0008	90	5	WWR-MS	145	70	25	3500

** The reactor has been shutdown since January, 1995.*

5- Reactor Vessel Material and Internals

The reactor vessel was made of aluminium alloy CAB-1 is fairly close in properties to the 6061 alloy used in the USA with material composition [%]:

Fe=0.085	Zn=0.011	Si=0.81	Ti=0.004	Cu=0.0058
Mn=0.0026	Ni=0.001	Cd=0.00002	B =0.000043	Mg=0.48

The most representative sites of the reactor vessel are end caps of the horizontal channels, where the fluence is the highest, and the core support grid, where mechanical stresses develop and, at the same time, irradiation affects noticeably the metal properties,Fig.1. The mechanical properties of CAB-1 suffer degradation as a result of production in the alloy of nuclear reaction products in interaction with thermal and fast neutrons,and radiation defect generation in the material. The stresses in the horizontal channel end cap are governed by thermal stresses. The high thermal conductivity of the material and small wall thickness (13 mm) provide an upper estimate for the total equivalent stress of 1 kg/mm^2. The sudden brittle fracture cannot occur in the horizontal channel because of the low acting stresses.

The visual inspection and material testing of the Hungurian reactor vessel showed that it was in good condition after 25 years of operation. There was no deterioration, even on the rubber sealing between the flanges of the vessel outlet and the first section of the primary pipeline although the gamma radiation was very high there. The only weldings that showed certain enbrittelement were those which received a higher fluence than $10^{19} n/cm^2$(E>1 MeV) (they were in the core or very close to it). [1].

Systematic monitoring of the core support grid condition and measurements of the thickness of material separating the holes have been carried out in 1986 for WWR-M reactor of PNPI, Russia within a routine inspection of the vessel and of the assemblies inside it,[2]. The results revealed degradation of

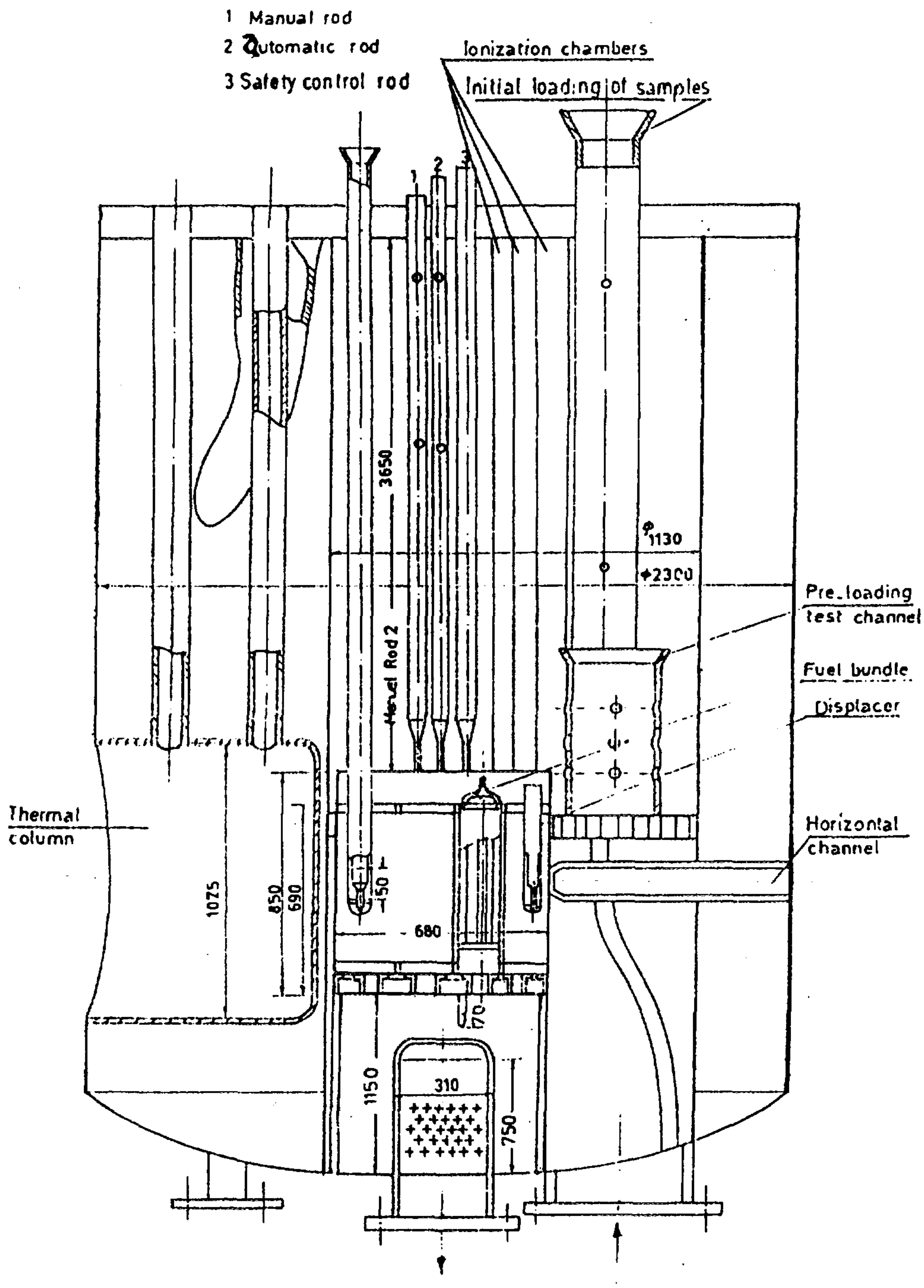


Fig.1. Reactor Construction

some holes in their lower part. It did not proceed further. It may be occurred apparently in the initial stage of reactor operation somewhere in the period 1962-1965, when the water in the primary circuit did not satisfy the optimum chemical parameters recommended for contact with aluminum alloys, and its analyses revealed considerable amounts of aluminum oxides, which attributed at the time to fuel element corrosion. The general corrosion rate of the other reactor vessel components, where the water movement velocity is several times smaller, did not exceed 0.005 mm/yr. Special measures have been taken to monitor grid condition. Inspection of the grid was performed once a year from top and from below, and thickness measurements were made on the material separating the holes. In view of the fact that fracture of the support grid can lead to fuel element damage, four devices measuring grid deformation in each loading and unloading cycle have been installed on the grid. The constancy of deformation amplitude is checked during reactor operation. As a result, one can predict safe operation of the reactor vessel until at least the year 2000, [2].

In early 1979 the presence of spots of different size which showed signs of corrosion upon the otherwise shining surface of the reactor pool lining was reported as a result of the regular checking procedures for visual observations to the internals of IRT-Sofia Research Reactor,[3]. Repair works of much larger scale have been performed eight years later, in 1987, when the reactor was not operational for two months. The lining has undergone a preliminary optical inspection by a remotely operated underwater TV system in order to specify the areas with signs of corrosion. The aluminum lining thickness was throughly checked by the ultrasound technique both before and after the repair. To do this the core was unloaded and the reactor internals except the grid were disassembled and stored away. Special care was taken in dismantling the control rods and the ionization chambers.

Argon arc welding works and quality assurance procedures have been carefully prepared, approved by the Regulatory body and then carried out. The arc welder accompanied by a health physicist with suitable dosimetric equipment were working in a specially designed lead shield of 8 tons weight moved cautiously within the reactor pool by the 10 tons reactor bridge-crane. About 31 hours of welding works were needed. A dose of 100 milirads was received each of this two men. The 21 employees involved in all kinds of repair work of 60 days duration received a total of 1.545 rads. The maximum of individual doses did not exceed 115 milirads.

The lining since then is optically inspected and cleaned periodically by a high pressure water-jet system. The same is valid for the reactor grid and other safety related in-pool equipment, [3]. The modernization project submitted for consideration in autumn 1989 included:

(1) a new lining of stainless steel covering the existing one;
(2) replacement of the core with a new one using IRT-2M fuel elements;
(3) replacement of some aged reactor systems;
(4) use of beryllium as moderator.

The core basket of EWA reactor was tested by both radiographic in 1967 and ultrasonic method in 1979,[4]. A reference basket was prepared and tested:

a) radiographically,
b) ultrasonically in the air,
c) ultrasonically under water.

Using the correlations obtained in reference basket testing no defect propagation was found.

6- Primary Cooling System, PCS

The PCS sucking pipe is joined to the vessel right under the core and a sudden break in this joint could provoke abrupt loss of water from the reactor, including uncovery of the core. These joints are situated under the reactor, in an area which is accessible only for short periods of time due to high intensity of gamma radiation. In EWA reactor,emergency support of PCS was installed under the most critical joints to minimize the gap estrangement in case of sudden drop of the pipes connection under reactor vessel. The studies showed that begining of fuel exposing takes about 8 seconds in the case of LOCA without the emergency support, and 400 seconds after installing it. Gap measurement probe was introduced to presage with the emergency state, Figs.(2-7). Thermal hydraulic studies were undertaken to determine the possible course of a reactor accident after break of these joints. The studies comprised analyses of heat removal from a fuel element. The upgraded reactors have provided with emergency core cooling system (ECCS). This system is controlled by electromagnatic valves supplied from emergency electric power sources. The valves would open due to spring and gravity forces whenever the electric power is lost. Two out of three monitoring system for water level and pressure measurements were installed, providing an activation signal for valves opening in case of a LOCA.[4].

The initial radiographic testing performed in the PCS of EWA reactor in 1975,i.e. after 17 years of reactor operation, showed poor quality of welded joints,[4]. Therefore, it was decided to undertake regular testing of all welded joints in the PCS. A summary of inservice inspections is shown in table 1. The results of PCS testing are contained in 13 reports of about 1000 pages volume. No crack propagation has been detected. The weakening of the weld area was minor, especially in view of low pressures and large wall thickness. However, the defects were too large for a responsible weld of a nuclear reactor. Subsequent controls over 12 years have not shown any crack propagation and the reactor has been safely operated throughout this time, Fig.8.

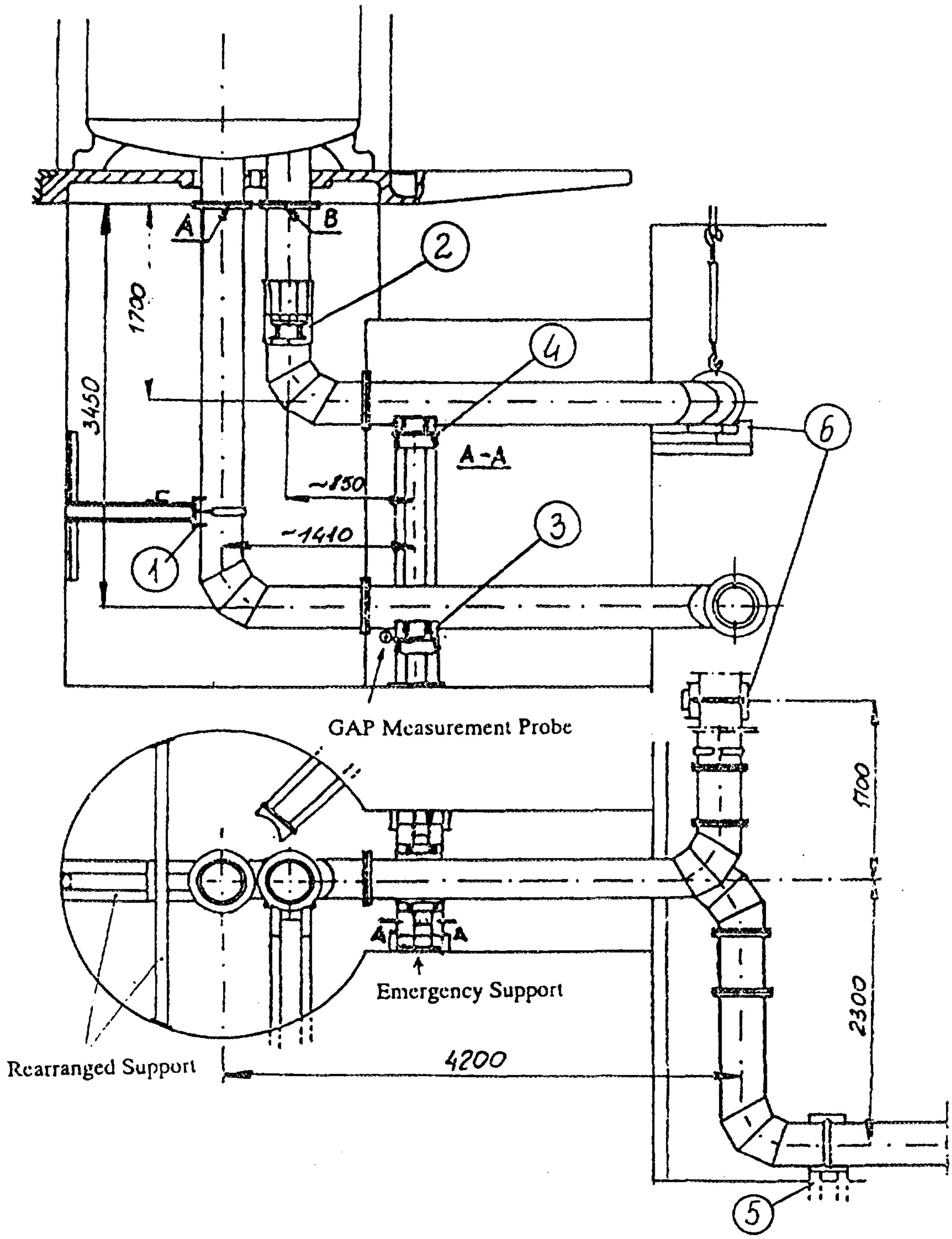


Fig.2. Primary Circuit Supports

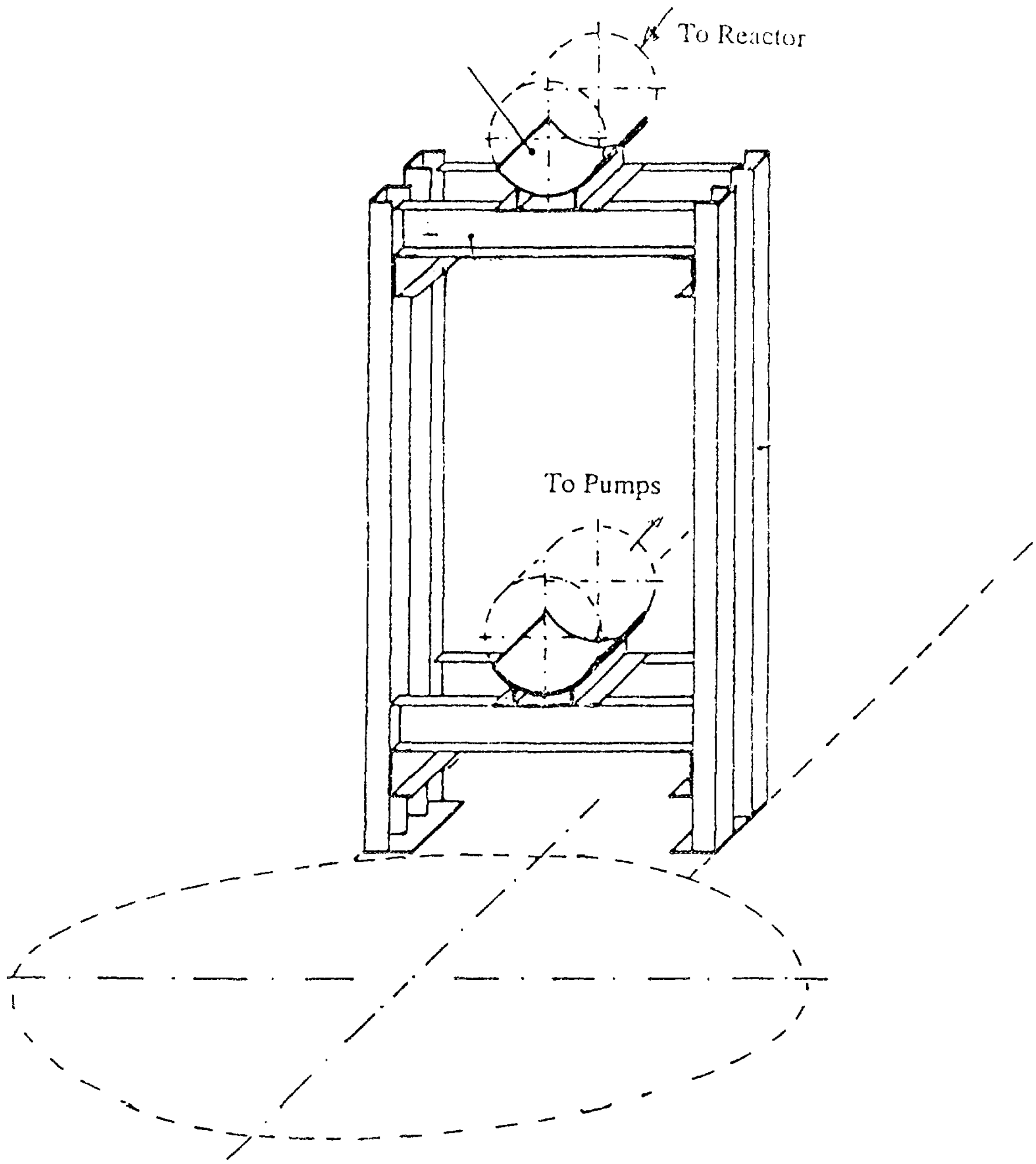


Fig.3. Emergency Support of Primary Circuit

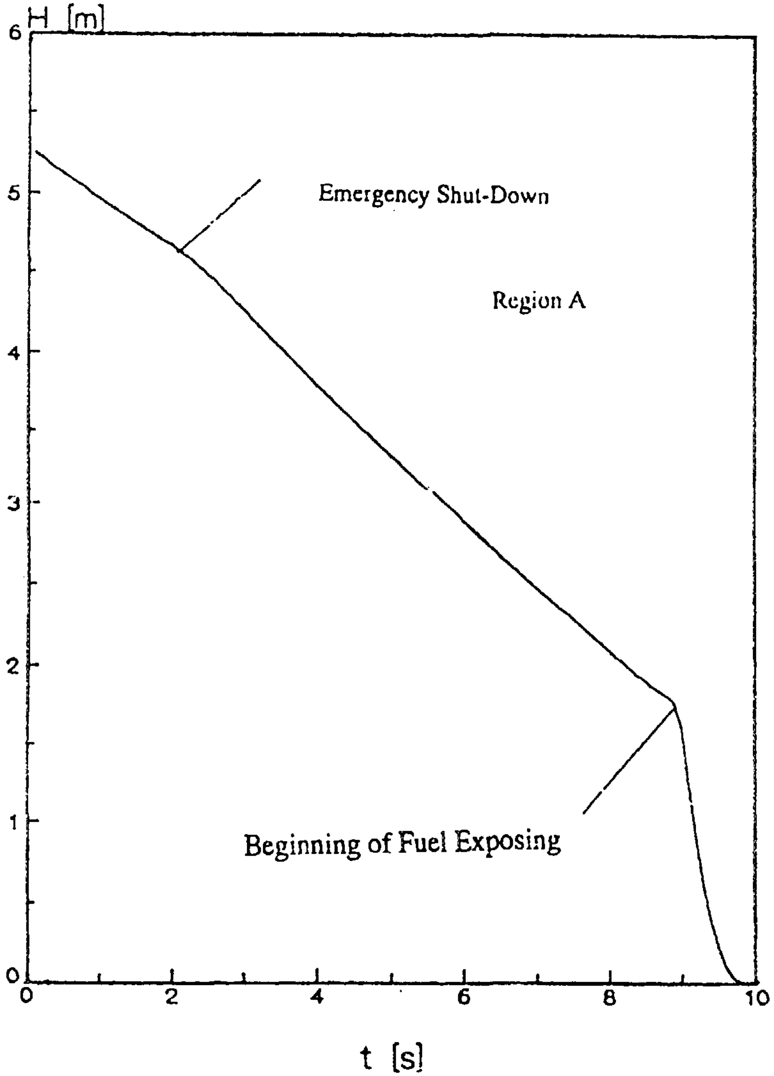


Fig.4. Drawdown in Reactor Vessel in the Case of LOCA in Region A Without the Emergency Support

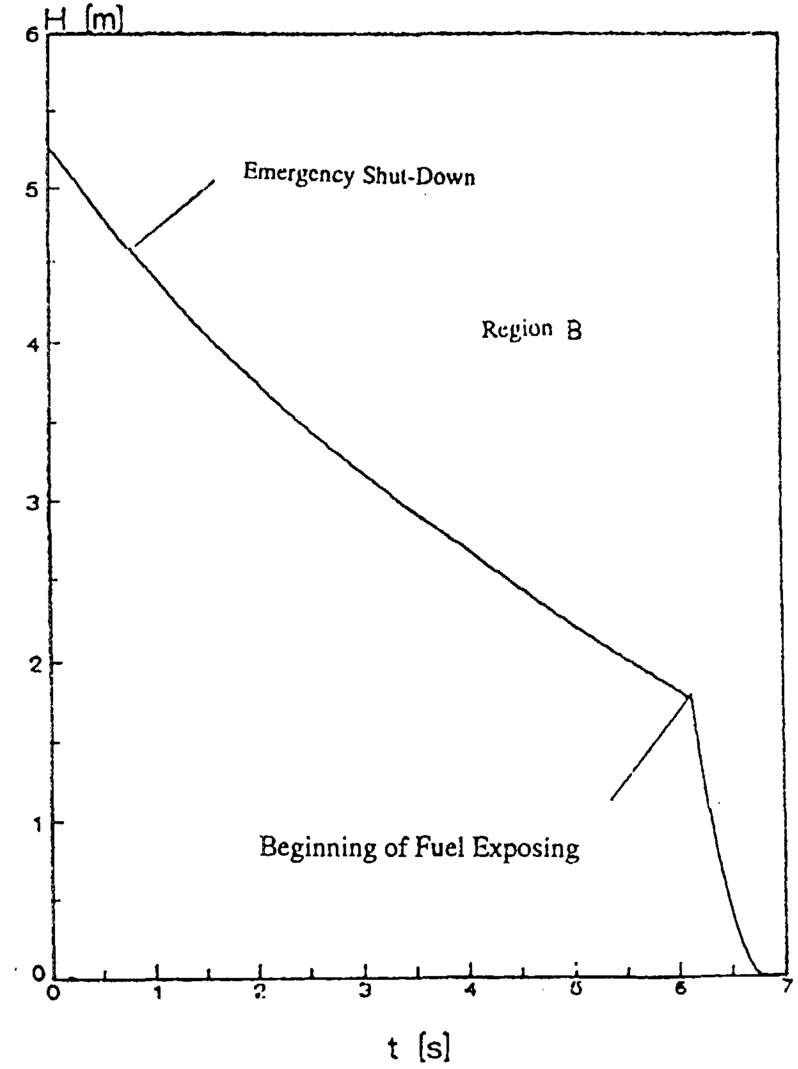


Fig.5. Drawdown in Reactor Vessel in the Case of LOCA in Region B Without the Emergency Support

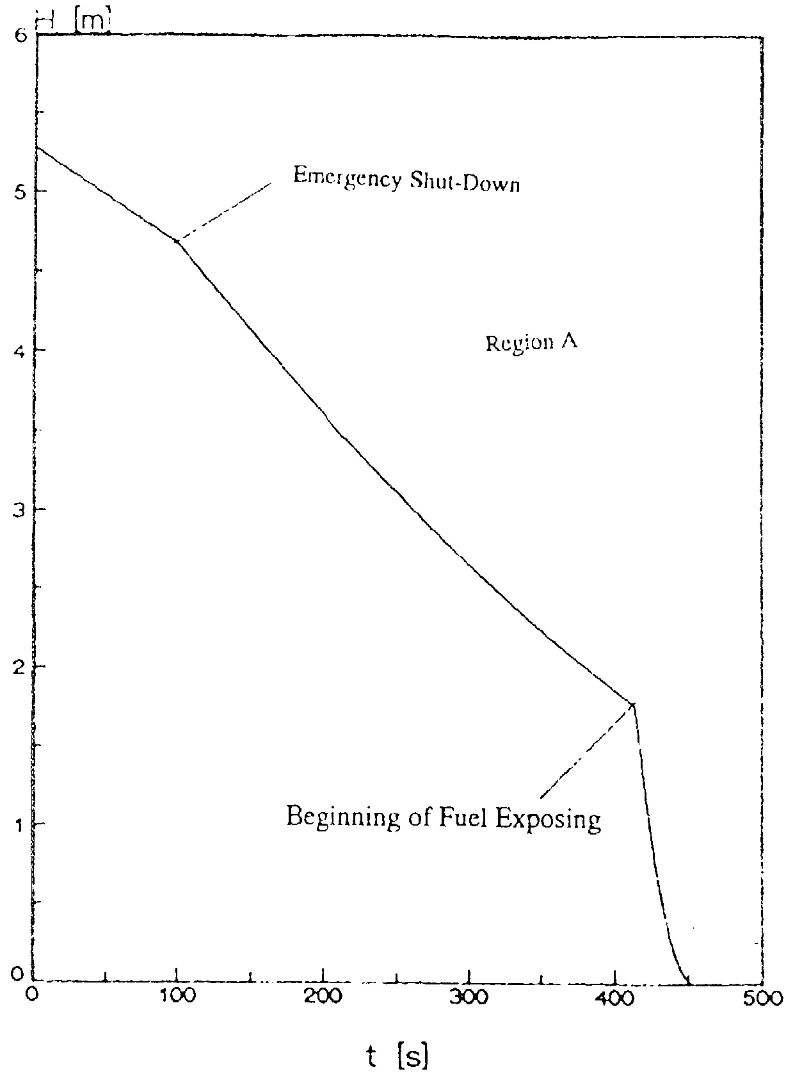


Fig.6. Drawdown in Reactor Vessel in the Case of LOCA in Region A With the Emergency Support

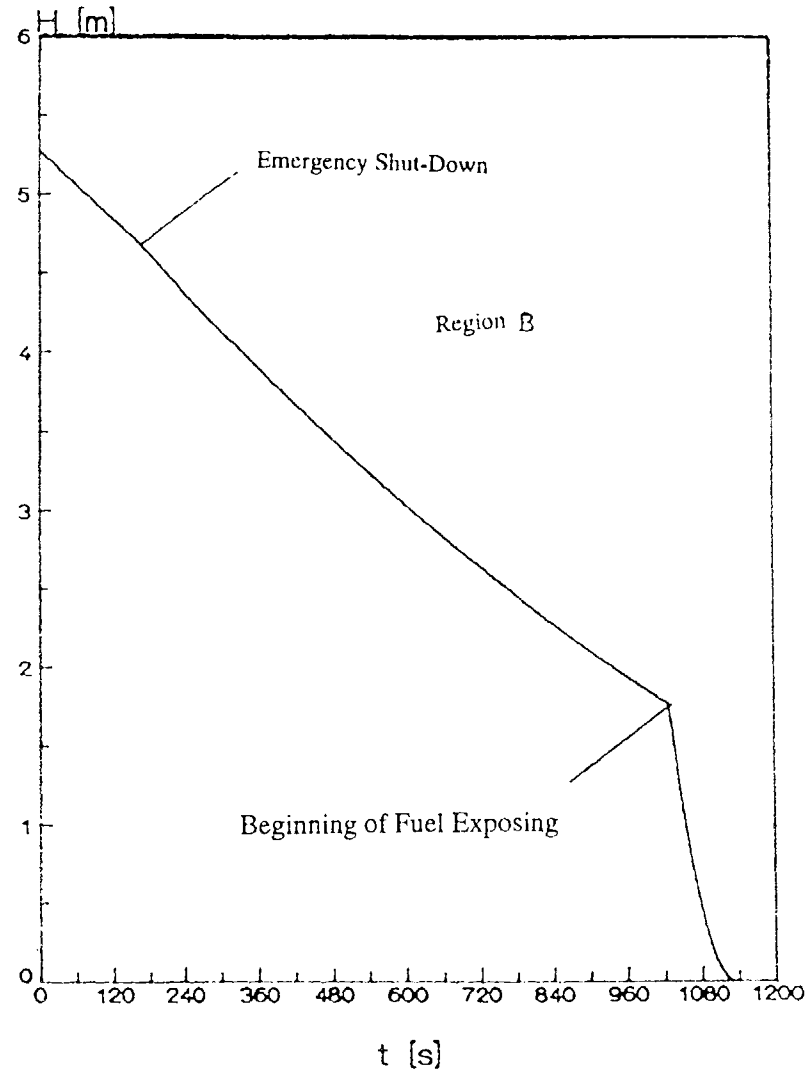


Fig.7. Drawdown in Reactor Vessel in the Case of LOCA in Region B With the Emergency Support

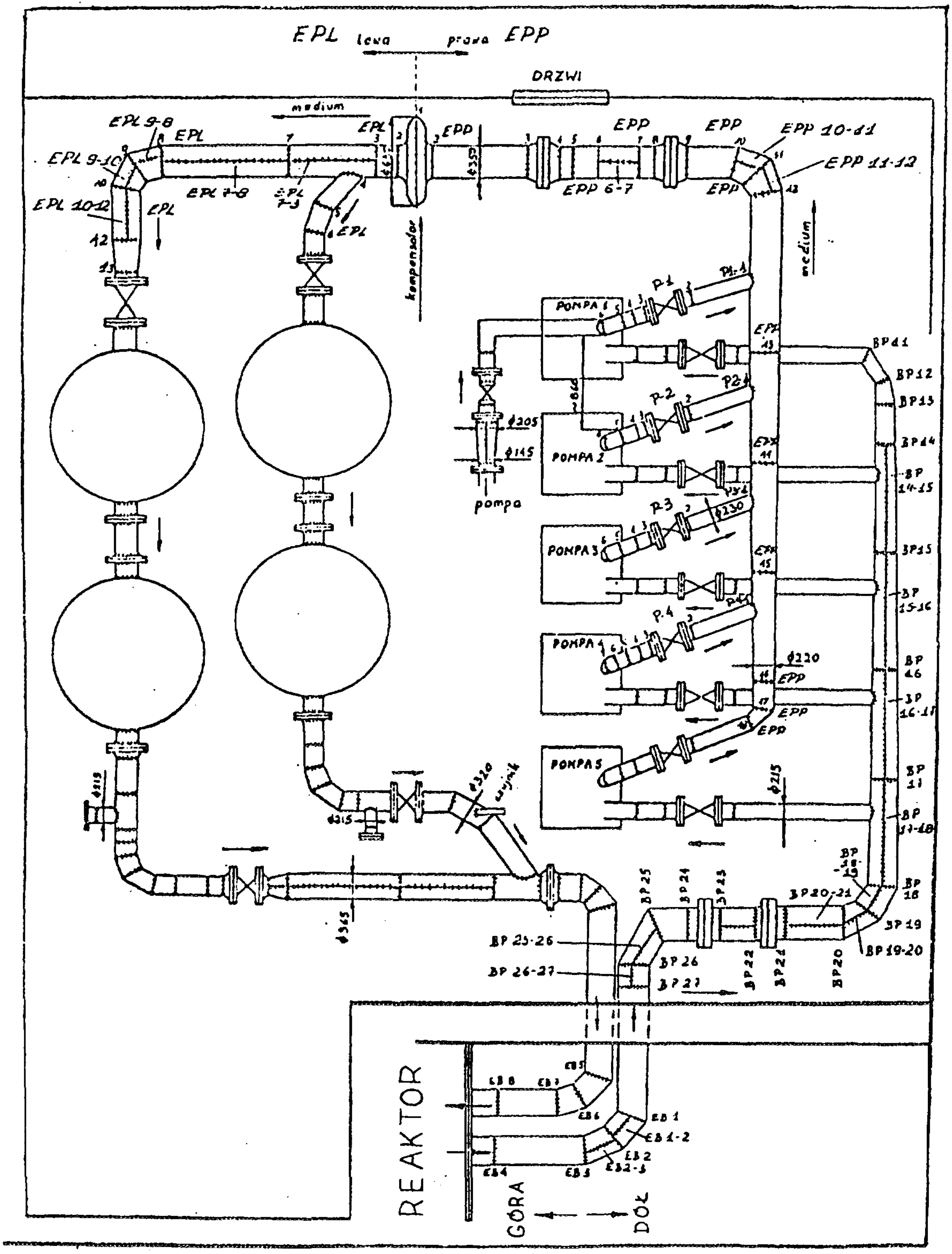


Fig.8. Welded Joints in Primary Cooling Circuit of EWA Reactor

Table 1. List of Radiographic (R) and Ultrasonic (U) Studies of PCS Welded Joints in EWA Research Reactor

Year	Scope of testing	Number of points, method
1970	PCS welded joints	3 R
1975	PCS welded joints	17 R
1976	PCS welded joints	21 R
	PCS walls thicknesses	all sections, U
	PCS materials checking	complex studies
	Pump No. 3, Valves	whole body, R
1977	PCS welded joints	3 R + 5 U
1978	PCS welded joints	5 R + 1 U
1979	PCS welded joints	5 R + 4 U
	Reactor vessel	All welds, U
	Core basket	All welds, U
1980	PCS welded joints	10 R + 8 U
	Vibration analysis	strain gauges,all sections
1981	PCS welded joints	12 R
	stress measurements	6 strain gauges
1982	PCS welded joints	13 R + 8 U
1983	PCS welded joints	10 R
1984	PCS welded joints	11 R
1985	PCS welded joints	9 R
1986	PCS welded joints	9 R + 4 U

Ultrasonic control has been repeatedly used to check wall thickness of heat exchangers and ion exchange column vessels to detect deep local corrosive pitting in the vessel walls. The results showed that the chemistry of the agent in the vessels affects on the progression of corrosive pitting in the metal, [4].

The current reactor problems could be deduced whenever the in-service inspection programme are implemented on ET-RR-1 reactor, [5]. Corrosion accompanied by leakage have been detected near the weld joints of the ion-exchange filter vessel, Fig.9. The vessel should be replaced by a new design which could facilitate the process of discharging the radioactive resins and charging the fresh resins. The PCS are made from 1X18H9T austentic stainless steel, whose chemical composition [%] is: C<0.12 Mn<2.0 Cr 17.0 Ni 8.0 Ti 0.5 S 0.03 Si 0.0 P 0.36.

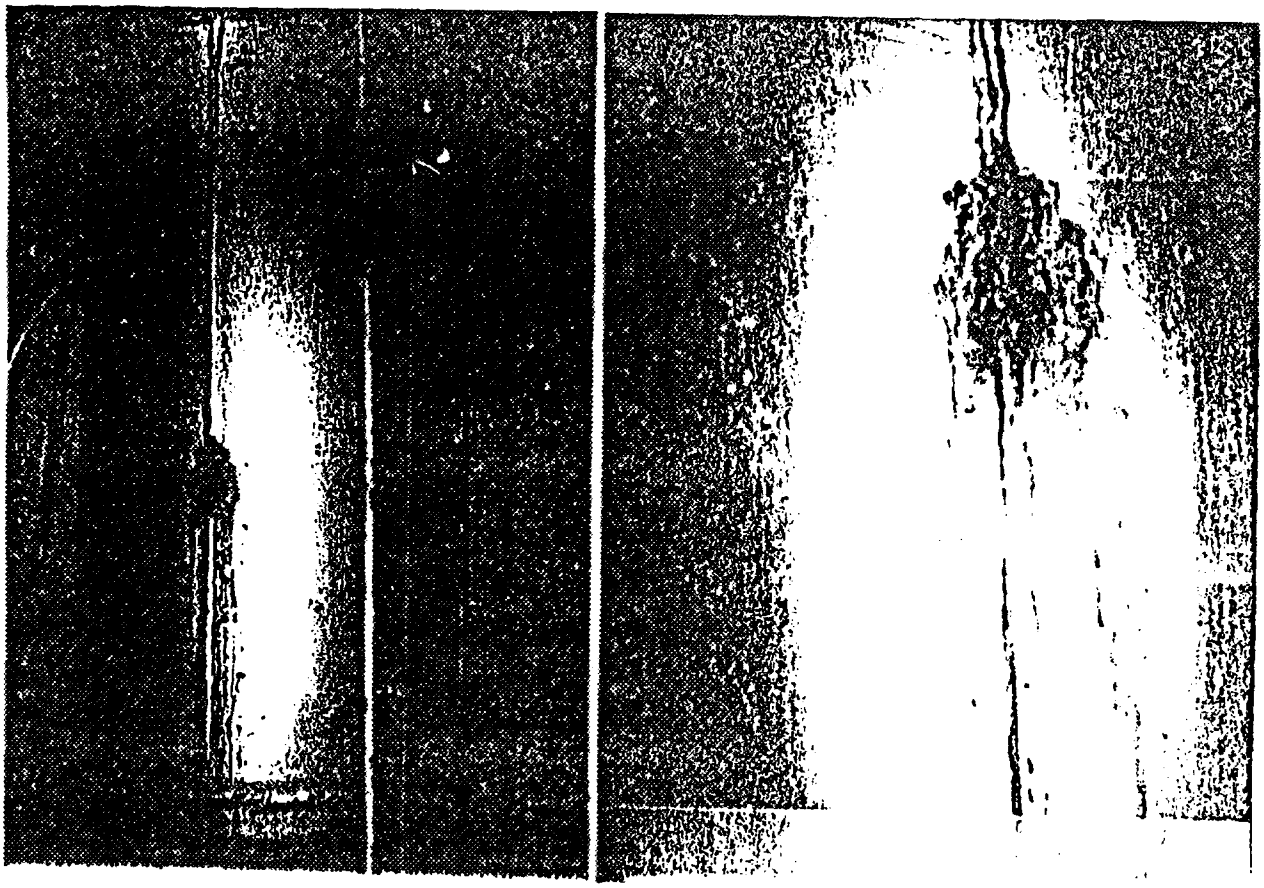
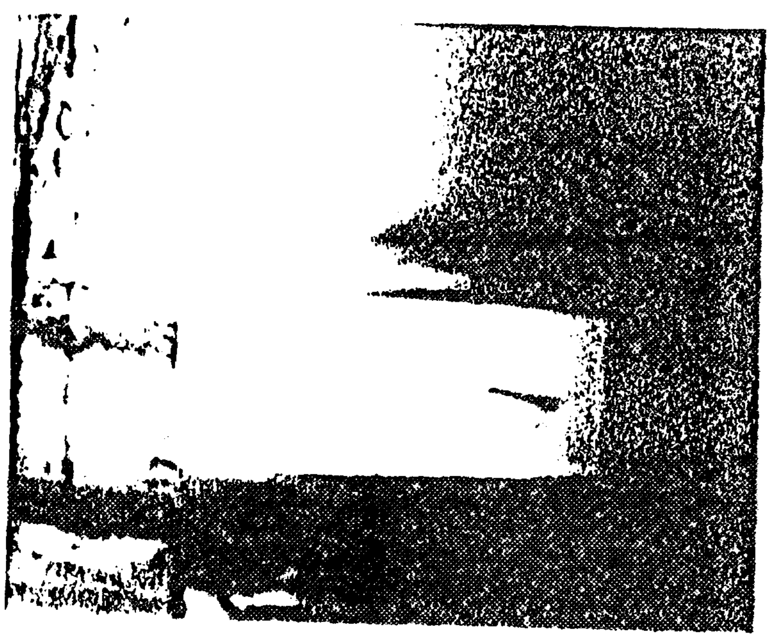


Fig.9. Corrosion in Weld Stem of the Ion Exchange Filter of ET-RR-1 Reactor



7- Spent Fuel Storage Pools

Spent fuel assemblies are stored on racks in a separate storage pool of about 12 m^3 deionized water, connected with the reactor by an inclined channel for refuling operations. The pool lining is of welded Al sheets 6 mm thick, (made of the same material as the reactor vessel). The condition of the Hungarian vessel was bad, as pitting corrosion traces could be found on the surface, mainly at the bottom of the vessel. The depth of the traces exceeded 4 mm (half of the wall thickness). The most probable reason for this corrosion was the lack of circulation which probably resulted in poor water quality in the lower part of the tank, [1]. The controlled parameters are water level and water quality. The spent fuel storage tank inside the reactor hall was replaced by a stainless steel one with high density storage racks, capable of storing irradiated fuel produced during more than five years operation. The storage capacity has been increased in many countries by fixing additional racks in compliance with criticallity calculations to store up more fuel assemblies, in addition to built new second vault outside the reactor hall. Mobile system is used for water purification.

8- Water Chemistry

Water quality: electrical conductivity, chemical composition and pH, play main role in controlling corrosion products and corrosion rate. The amount of chlorides, sulphates, iron, aluminum and copper must be controlled by water purification system before introducing into the reactor to be within the standard specification of water for nuclear reactors, [8], as follows:

Electric conducting	< 2	µS/cm
pH	5.5-6.5	
Al^{+++}	0.05	ppm
Cu^{++}	0.02	ppm
Cl^{-}	0.05	ppm
Fe^{++}	0.05	ppm

Additional ionizers have been installed to the feedwater supply systems to improve the quality of the deminralized stored water in many countries. The primary cooling water is purified by the ion-exchange filter to retain its quality due to reactor operation. Continuous monitoring to PCS conductivity, pH value and radioactivity indicate reactor status during operation.

9- Enhancement of Reactor Safety

The operating experince of many countries proved that there was no need for the degassing tank in the primary circuit (dearator), of WWR-C research reactors. In Hungary,it was replaced by a gravity tank. IF the primary flow stops (e.g. in case of blackout),the natural flow from the reactor vessel to the gravity tank is sufficient for cooling the remanent heat until the start up of the diesel generators, [1].
The pump room is beneath the reactor vessel therefore for LOCA cases three water feedback systems are available. Two of them can feed back the water pouring below the reactor vessel, while the third feeds from the drainage of the pump room.
The new secondary cooling system has dry cooling towers with big fans in the middle in order to decrease the water consumption and save the environment from the evaporated water.
The reactor hall sealing was improved to decrease the leakage rate and absolute filters in the new ventilation system were introduced. For emergencies a recirculation ventilation system was designed with absolute filters, [1].

10- Conclusions

In spite of the mentioned safety problems of WWR-C research reactors, they have worked successfully for more than thirty years in different countries. Most of them have been upgraded to higher powers. Handling these problems provide experience to the reactor staff and designers. Safety rules have been developed through the gained experience.

CAB-1 aluminum material of WWR-C reactor vessels have been thermally treated to CAB-T for the modified WWR-M reactors. Water chemistry control becomes very important for reactor safety. QA-QC, ISI and NDT as well as SAR represent principle documentations of nuclear reactor installations. Operational procedure and maintenance manuals in addition to efficient utilization of the research reactor became important technical and scientific tasks related to reactor safety. The IAEA, have published several safety series issues for ISI of nuclear power plants and management of research reactor ageing, [9-12] . These issues are good guidance for preparing similar issues for ISI manual to research reactors.

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