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STUDSVIK'S R2 REACTOR - REVIEW OF THE CAPABILITIES AT A MULTI-PURPOSE RESEARCH REACTOR

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STUDSVIK'S R2 REACTOR - REVIEW OF THE CAPABILITIES AT A MULTI-PURPOSE RESEARCH REACTOR

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ABSTRACT

A general description of the R2 reactor, its associated facilities and its history is given. The facilities and range of work are described for the following types of activities: fuel testing, materials testing, neutron transmutation doping of silicon, activation analysis, radioisotope production and basic research including thermal neutron scattering, nuclear chemistry and neutron capture radiography.

1 INTRODUCTION

STUDSVIK AB performs R&D work and associated activities, primarily in the nuclear energy field. STUDSVIK AB is a commercial company, active in the areas of services, supply of special equipment and systems and also consulting. STUDSVIK NUCLEAR AB, which is the largest subsidiary within the group, is one of the direct offsprings of AB Atomenergi, the origin of the STUDSVIK group, which was formed in 1947. The STUDSVIK group has about 540 employees and a turnover of about 500 MSEK/year.

During the 1950's and 60's, an ambitious nuclear program was launched in Sweden. The experience and competence gained from a large number of advanced projects constitutes the basis upon which the present activities of STUDSVIK NUCLEAR are based. Since the 1970's, the efforts have been concentrated on light water reactor fuel and materials, and the originally domestic R&D programs have been expanded so that a large fraction is now financed by non-Swedish sponsors.

Neutron activation analysis and radioisotope production as well as beam tube experiments for basic research applications were started in the 1960's. In 1977 neutron transmutation doping of silicon began.

The facilities of interest in this connection are the R2 Test Reactor, the Hot Cell Laboratory, the Lead Cell Laboratory and various other laboratories, all located at Studsvik, 100 kilometers south of Stockholm.

2 THE R2 REACTOR

2.1 General Description of the R2 Reactor

The R2 reactor is a tank-in-pool reactor, see Figure 1 and Table 1, in operation since 1960 and originally similar to the Oak Ridge Research Reactor, ORR (1). The reactor core is contained within an aluminum vessel at one end of a large open pool, which also serves as a storage for spent fuel. Light water is used as core coolant and moderator. The reactor power was increased to 50 MW(th) in 1968. During 1984-85 the reactor was rebuilt and a new vessel was installed.

The R2 reactor has a high neutron flux, see Table 1, and special equipment for performing sophisticated experiments. An important feature of the R2 test reactor is that it is possible to run fuel experiments up to and beyond failure; that is not possible in a commercial power reactor.

Table 1. Technical Data for the R2 Test Reactor.

Power	50 MW(th)
Moderator/coolant	H_2O (D_2O)
Fuel length	600 mm
Fuel assembly length	924 mm
Fuel assembly cross section	79x82 mm^2
Number of fuel plates per assembly	19/18
Neutron flux in experimental positions,	
thermal	(0.3–2.5) x 10^{14} n/(cm^2-sec)
fast (>1 MeV)	(0.5–2.0) x 10^{14} n/(cm^2-sec)
Primary flow	1 100 kg/sec
Coolant temperature	<45°C

The present core configuration is shown in Figure 2. The components of the core are arranged in an 8x10 lattice. A normal core consists of 46 fuel elements, 6 control rods, about 12 beryllium reflector assemblies and a number of in-pile loops, irradiation rigs and aluminum fillers. Rows No 1 and 10 consist of beryllium reflector assemblies. The composition of the core can be altered to suit the experimental program.

The irradiation facilities in the R2 reactor have been described in the literature (2–4), and details are given below. Most base irradiations of test fuel (that is irradiations at constant power, where fuel burnup is accumulated under well-defined conditions) are performed in boiling capsules (BOCA rigs). Some base irradiations and all ramp tests (that is irradiations under power changes) are performed in one of the two in-pile loops, which can be operated under either BWR or PWR pressure and temperature conditions. The ramp tests, simulating power transients in power reactor fuel, are achieved by the use of ^3He as a variable neutron absorber. Structural materials, such as samples of Zircaloy cladding and candidate materials for advanced reactors can also be irradiated in special rigs either in the loops or in special NaK-filled irradiation rigs in fuel element positions with a well-controlled irradiation temperature.

The R2 core has an active fuel length of 0.6 meters. Most fuel rods irradiated are segments of power reactor fuel rods, so-called rodlets, with about the same length. Non-destructive examinations of fuel rods can be performed in the R2 pool during short pauses in the irradiation program or between various phases of an experiment, see Section 3. All the handling and all the examinations are performed with the fuel rods in the vertical position; this is advantageous with respect to possible movements of fuel fragments etc.

Associated with the R2 reactor is the 1 MW(th) swimming pool R2-0 reactor, which is located in the same pool, see Figure 3. The basic research performed by use of R2-0 is briefly described in Section 10.

2.2 Boiling Capsules (BOCA rigs)

The Boiling Capsule (BOCA) facility for irradiation of BWR and PWR fuel rods was introduced in 1973.

The in-pile part of a BOCA rig consists of a bare stainless steel pressure thimble containing a shroud with flow entrance ports at the bottom and exit ports at the top. The lower part of this shroud is located in the reactor core region. A fuel test rod bundle consisting of 4 (or 5) test rods is located inside the shroud. The BOCA is filled with highly purified pressurized water from a special pressurization system. Figure 4 shows a simplified BOCA flow diagram. BOCA system technical and operational data are given in Table 2.

Coolant circulation and cooling is brought about by natural circulation, although no net boiling occurs. The water is heated mainly by the fuel rods. Buoyancy forces make hot water with lower density rise in the riser shroud, leave the exit port, meet the cold wall of the pressure thimble and be cooled down while flowing downwards in the annular channel between the pressure thimble wall and the riser shroud.

The circulation flow rate is low, about 0.2 kg/sec, but the coölant is substantially subcooled. When the heat flux at the fuel rod surface exceeds about 60 W/cm², subcooled nucleate boiling will occur at the rod surface, rendering a surface temperature equal to or slightly higher than the saturation temperature for the actual static pressure.

Table 2 BOCA System Technical and Operational Data.

Permissible operating pressure range:	
– Pressurizer System BOCA A	30 to 100 bar
– Pressurizer System BOCA B&C	30 to 170 bar
Permissible total power generation in one BOCA rig	~ 90 kW
Permissible heat flux on test rod surface with regard to departure from nucleate boiling (DNB)	160 W/cm²
Possible range of thermal neutron flux achievable by:	
– Positioning in the core	0.5 to 1.5×10^{14} n/(cm²·s)
– Internal hafnium shielding	Flux reduction to ~ 50 %
– Internal hafnium shielding plus hafnium shielding of the Be–position rig adapter	Flux reduction total to 30 %

Power in the test rods is measured by a combination of nuclear measurements with the Delayed Neutron Detector (DND) technique, and coolant water temperature and flow measurements.

Up to five BOCA rigs can be operated simultaneously in the reactor. Two independent pressurization systems are available, each capable of supplying 3 to 5 BOCA rigs with water. Each BOCA rig is connected to a separate outlet circuit.

Each rig is constantly fed with a purging water flow in order to control the water chemistry in the self–circulating water volume of the pressure thimble. The same flow is let out to the drain. This inlet and outlet of water takes care of thermal expansion and contraction of the water in the thimble. During reactor start–up, when the water is heated up and expands, the flow is increased. The out–going water is monitored for radioactivity (fission products), and water chemistry is controlled.

In order to make it possible to irradiate power reactor fuel with standard enrichment in the R2 in–pile loops and BOCA rigs, it is often necessary to decrease the neutron flux. This is achieved with hafnium absorbers in the form of tubes or plates.

2.3 In–pile Loops

Data for the two loops used in the current LWR fuel R&D programs are shown in Table 3 (2–3). Data for other loops, e g for the testing of HTR fuel, are described elsewhere (5) and will not be discussed here. Each LWR fuel loop utilizes two diagonally adjacent fuel element positions in the R2 test reactor, see Figure 2.

The two loops are pressurized in order that test fuel rods can be investigated at realistic operating conditions for either PWR (loop No 1) or BWR (both loops) type power reactors.

Table 3 Characteristics of the R2 in-pile loops.

Loop No	1	2
Type	PWR	BWR
Pressure, bars	30–150	30–90
Coolant temperature, °C	220–325	220–285
Coolant flow rate, kg/sec	2.5–4.5	2.5–5.0
Max cooling capacity, kW	150	400
Neutron flux, $10^{14}/(cm^2 \cdot s)$		
Thermal, $\varnothing_{th}$	0.6–1.2	0.6–0.9
Fast (>1 MeV), $\varnothing_f$	0.8–1.5	1.2
Gamma heating in stainless steel, W/g	5–10	5–8

The loops can be used for irradiation at constant power of up to 4–5 test fuel rods simultaneously, and for power ramp tests of single rods. The test rods to be ramp tested are installed in the loop in special capsules, which in turn are inserted in a special test rig. The loops can also be used for irradiation testing of structural materials, e g Zircaloy test specimens, including creep specimens, and steel specimens of various types.

The in-pile part of the loops are of a U-tube design, taking up two core positions and thus providing two test positions in the R2 core, one of which can be used for ramp tests (Figure 5). The U-tube is isolated from the reactor primary coolant by a gas gap containing CO_2. Heat losses from the tube to the reactor coolant are therefore quite small, which facilitates accurate test rod power measurement.

The working in-core space inside diameter of these in-pile pressure tubes is 45.5 mm. Useful length in the core is 670 mm.

The main features of the loops are presented in Figure 6. This figure shows a simplified flow diagram of Loop No 1.

In the present Ramp Test Facility, introduced in 1973, the fuel rod power during performance of ramp tests in the loops is controlled by variation of the ^3He gas pressure in a stainless steel double minitube coil screen which surrounds the fuel rod test section. The principle of operation of this system is based on the fact that ^3He absorbs neutrons in proportion to its density, which can be varied as required by proper application of pressure.

The axial location of the minitube coil screen in the loops' U-tubes is shown in Figure 6. The efficiency of the ^3He neutron absorber system makes it possible to increase test rod power by a factor of 1.8 to 2.2 (depending on the fissile content of the fuel). The ^3He absorber system is designed to achieve a 100 % power increase within 90 seconds, when operating with the normal pressure variation (bellows system).

In order to achieve a higher power increase than a factor of about 2, the reactor power must be increased before or simultaneously with the "^{3}He ramping". This variety of combined ramp systems is called "double step up-ramping", Figure 7. The technique makes it possible to increase the test fuel rod power by a factor of about three.

An important advantage of the R2 Ramp Test Facility is that test rods, one at a time, can be loaded and unloaded during reactor operation. This is done by means of a lock vessel built onto an axial drive mechanism with about 3.5 m stroke. This lock vessel is bolted on top of a lock valve (ball type valve) fixed on top of the ramp rig. For BWR pressure conditions and normal rod lengths there is a 4-rod revolver lock vessel with a mechanical chain drive. For tests of PWR fuel rods and longer than normal rods, there exists a selection of rod lock vessels and dedicated hydraulic drives. Figure 6 shows a ramp rig with a hydraulic drive.

In the Ramp Test Facility ramp rates can be achieved in the range of 0.01 W/(cm·min) to about 3 000 W/(cm·min).

The power (linear heat generation rate in the fuel rod) is measured calorimetrically by the use of two inlet thermocouples, two outlet thermocouples, a venturi flow-meter and a pressure gage. The special calibration techniques employed have been described (3). The estimated uncertainty ($\pm 1\sigma$) is 2.3 % when the most common rod lengths (0.3–1.4 m) are used. The reproducibility obtained, when a fuel rod is irradiated several times in the same ramp rig, is ± 1 %. For fast ramps the discrepancy between the terminal power aimed at and the one obtained is less than ± 1 kW/m.

The axial thermal neutron flux distribution is measured by activation of cobalt wires in dummy rods and by gamma scanning of the ramp tested fuel rods.

Each test rod is mounted in a separate stainless steel "capsule" (a shroud open at both ends), primarily as a safety measure and to facilitate the removal and handling of test rods that fail in the course of a power ramp. The "capsule" with the fuel rod is connected to the actuating rod which is used to move the fuel rod axially between the rod changing device and the in-pile section of the rig. There is a small floating push-rod built in at the bottom guide plug of the capsule. This push-rod transmits the elongation movements of the test rod to the LVDT type elongation detector built into the bottom of the ramp test rig.

2.5 Fuel Rod Failure Detection System

Fuel rod failures in the loops are detected by a Cerenkov-type radiation sensor which monitors the activity of the loop coolant water. The Cerenkov detector is installed in a by-pass circuit in order to increase the detection system sensitivity by decreasing the background ^{16}N activity produced in the loop coolant water. The ^{16}N background activity is decreased by the introduction of a delay time due to the fact that the Cerenkov detector is positioned in the by-pass circuit.

The system detects fuel rod failure after 155 ± 10 seconds. This degree of failure detection capability has been verified experimentally. An example of system operation during a ramp test where the rod failed, is shown in Figure 8.

However, the moment of failure is also registered instantaneously by the rod elongation measurement system as a sudden rod contraction and also often by the power measurement system as a small thermal "spike", (3,6–7).

3 POOL–SIDE EXAMINATION

The general appearance of the irradiated fuel rods can be studied by visual inspection in the R2 pool (3). The following phenomena can also be investigated in detail:

Dimensional changes, ridge formation, rod bow and creep–down can be investigated with equipment for profilometry and length measurements. The existence and location of fuel rod defects can be established by means of eddy current testing.

The axial distribution of certain nuclides is determined by axial gamma scanning of fuel rods or of cladding samples. Data obtained before ramp tests are used as a check on the burnup profile during the base irradiation. Data obtained after ramp testing are used to check the power profile during the R2 irradiation and for studies of the fission product redistribution.

Neutron radiography can be used to study the general appearance and dimensions of the fuel, the extent of filling out of pellet dishing, of center porosities and of center melting (8, 9). This type of examination also reveals the presence of special fuel cracks, interpellet gaps etc. Indications of cladding failure and of structural changes in the fuel can also be observed. In cases where there is no leakage of fission products from failed fuel rods neutron radiography is an important tool since cladding leaks are indicated by the existence of hydrides in the cladding or by the presence of water in the fuel cracks.

4 POST IRRADIATION EXAMINATION

Post–irradiation examination of irradiated fuel is performed in STUDSVIK's well–equipped Hot Cell Laboratory, which has been described in a separate publication (10).

Post–irradiation examinations of structural and cladding materials are performed in STUDSVIK's versatile Lead Cell Laboratory, which contains equipment for post–irradiation mechanical testing, corrosion testing and metallography.

5 FUEL R&D

Much of STUDSVIK NUCLEAR's R&D work in the fuel area has been concentrated on fuel testing, which can be made in the R2 test reactor with high precision under realistic water reactor conditions. This type of work was started in the early 1960's. In a very general sense the purpose of fuel testing can be described as follows:

- Increase of reactor availability by decreasing fuel–related operational power restrictions, defining the operational power limits
- Acquisition of experimental data for fuel–related safety considerations
- Decrease of fuel costs by making increases in fuel burnup possible.

The fuel testing activities can be divided into a number of well–defined steps as follows:

- Base irradiation, performed
  - in a power reactor, or
  - in STUDSVIK's R2 test reactor
- Power ramping and/or other in–pile measurements, performed
  - in STUDSVIK's R2 test reactor
- Non–destructive testing between different phases of an experiment, performed
  - in STUDSVIK's R2 test reactor pool or in the Hot Cell Laboratory

- Destructive post-irradiation examinations, performed
  - in STUDSVIK's Hot Cell Laboratory, or
  - in the sponsor's hot cell laboratory.

Fuel examination can be performed on standard (full-size) fuel rods from power reactors, which can be investigated in the Hot Cell Laboratory. If required, some types of tests could also be performed on such fuel rods in the R2 test reactor. However, due to the rather large initiation costs, such tests have not yet been performed. It should be noted, however, that short fuel rodlets, suitable for ramp testing and other on-line measurements in the R2 test reactor, can now be fabricated from irradiated full-size power reactor fuel rods by the STUDFAB refabrication process.

Fuel testing in the R2 test reactor is usually performed on fuel segments of 350-1000 mm length. However, tests have also been performed on full-size demonstration reactor fuel rods with up to 2.5 m length. In those cases only the lower 0.6 meters were irradiated. Irradiation at constant power is performed in boiling capsules (BOCA rigs) in fuel element positions, or in pressurized water in-pile loops operating under BWR or PWR pressure/temperature conditions, as described in Sections 2.2 and 2.3 above.

Ramp tests incorporating a very fast-responding test rod power measuring system and associated on-line measurements, such as rod elongation and noise measurements for studies of the rod thermal performance, are performed in the pressurized water loops.

The ramp tests are a form of integral performance tests where the complex interplay between the pellets and the cladding of a power reactor fuel rod is reproduced. The primary test objectives are:

- Determination of the failure boundary and the failure threshold, see Figure 9.
- Establishing of the highest "conditioning" ramp rate that safely avoids failure occurrence.
- Study of the failure initiation and progression under short time over-power transient operation beyond the failure threshold.
- Proof testing of potential pellet-cladding interaction (PCI) remedies.

Other, more specific test objectives have also been pursued in some projects.

The fuel testing projects executed at Studsvik have been organized under three different types of sponsorship:

International (multi-lateral) fuel projects

- Jointly sponsored internationally on a world-wide basis
- Project information remains restricted to the project participants through-out the project's duration and some pre-determined time after project completion.

Bilateral fuel projects

- Sponsored by one single organization, or a few cooperating organizations
- Project information remains restricted to the sponsor, sometimes published later.

In-house R&D work

- Sponsored by STUDSVIK NUCLEAR.

Several new hot-lab techniques have also been introduced in recent years (10). The STUDFAB process for fabrication of rodlets from full-size fuel was mentioned above. Fuel ceramography can include scanning electron microscopy (SEM) and electron probe microanalysis (EPMA).

Descriptions of the fuel testing facilities and the techniques used were given in Section 2 above. The noise measurements introduced for studies of the rod thermal performance have been described elsewhere (11-13). Several other novel testing techniques have also been introduced (14). A very fast ramp rate, up to 3 000 W/(cm·min) can be used to obtain fast power transients and to determine the pellet-clad interaction/stress-corrosion cracking (PCI/SCC) failure boundary. The double step up-ramping technique was described in section 2.4. On-line elongation measurements can be performed during ramp tests, Figure 8. Test fuel rods can be fitted with on-line pressure transducers through a refabrication process.

6 MATERIALS R&D

The R&D work in this area consists of studies of irradiation effects in structural materials. These types of studies have been concentrated on pressure vessel steels, stainless steels and nickel-base alloys (for super-heater and fast reactor cladding) during the 1960's and on zirconium alloys since the 1970's. The early pressure vessel steel work comprised investigations of different potential pressure vessel materials, of different materials variables, and of the influence of different irradiation conditions (neutron fluence, irradiation temperature etc). Recent work has been concentrated on accelerated irradiation of materials actually used in pressure vessels under as realistic conditions as possible. Recent work on stainless steels has to a large extent been concentrated on fusion reactor materials within the Next European Torus (NET) program, where tensile tests, fatigue tests, and stress corrosion tests have been performed after irradiation to a displacement dose of 10 dpa. Some work, including post-irradiation creep and fatigue tests and crack propagation studies, has also recently been performed on potential FBR vessel materials. The work on nickel-base alloys has been discontinued. The work on zirconium alloys is continuing and is being expanded in order to include in-pile corrosion.

7 NEUTRON TRANSMUTATION DOPING

Neutron transmutation doping of silicon for industrial use in electric power components is performed. With the present facility, from 1982, it is possible to treat simultaneously three shelves with silicon ingots, see Figure 10, where only the bottom shelf is shown. The silicon ingots are loaded manually onto the triple-shelf irradiation facility, which is situated in the R2 reactor pool, in front of the reactor vessel. The ingots perform a horizontal helical movement on the shelves in front of the core. The neutron flux is monitored through self-powered neutron detectors and the velocity of the ingots and hence the neutron fluence is controlled by a computer.

After completion of the irradiation the ingots are removed from the shelves to a conveyor which slowly transports them to the pool surface. A permanent radiation instrument monitors the dose rate and in order to avoid hand doses to the operators the ingots are lifted with a crane. They are then stored for a few days in order to let the ^{31}Si and ^{32}P activities decay. The decontamination is done by rinsing in demineralized water.

Silicon ingots with lengths up to 600 mm and diameters from 60 to 152 mm are treated routinely. The target resistivity of the resulting n-conducting material usually lies in the 30-300 ohmcm range. The high uniformity and precision of the irradiation guarantees less than 5 % axial variation and a radial gradient which is better than 2 and 4 % for the minimum and maximum diameters, respectively. The day-to-day constancy of the operation of the facility is monitored by means of cobalt monitors attached to some of the silicon ingots.

The irradiated material is shipped in compliance with IAEA regulations. The bulk material must have a specific activity of less than 7.4 Bq/g before it is shipped as non-radioactive material. For the nonfixed surface contamination a limit of 0.4 Bq/cm² is maintained taking the ALARA principle into consideration. Application of these values to the process normally gives a turnover time of three weeks at Studsvik.

8 NEUTRON ACTIVATION ANALYSIS

The present set-up for neutron activation analysis (NAA) permits multi-element determination by instrumental and radiochemical NAA of around 50 elements in trace concentrations. The samples which are investigated are of biological, environmental, industrial, or geological origin.

The samples usually require little or no pre-treatment, and after weighing they are pneumatically transported in plastic capsules to a position close to the reactor core. Having been irradiated the samples are left to decay in a storage unit for a suitable period of time, before the gamma radiation is registered with a Ge(Li) or a high purity Ge detector. The thermal and epithermal fluxes are measured with a Zr monitor. Thus the automatic data evaluation system gives an absolute determination of the composition of the sample from the intensity of the gamma spectrum, which is characteristic of each element.

Neutron activation analysis in theory permits determination of around 70 elements. In practice the number is limited to some 30 elements when using instrumental NAA. Application of radiochemical NAA increases this number.

Many elements can be determined at sub-ppb(10^{-9}) levels, but high concentrations of disturbing elements may be troublesome because of spectrum interferences. Sometimes corrections have to be made for other reasons, for instance due to interfering nuclear reactions. Neither the lightest elements nor lead and sulphur can be detected. Some examples of STUDSVIK NUCLEAR projects for which instrumental NAA has proved to be an efficient method are:

- Multi-element studies of geological samples, with special interest in rare earth elements and iridium.
- Uranium, thorium and other elements in sediment.
- Trace elements in metallurgical products.
- Trace elements in foodstuff.

9 RADIOISOTOPE PRODUCTION

Radioisotopes can be produced over a wide range of conditions in several irradiation positions in and around the R2 vessel. The operational cycle of the reactor, however, to some degree limits the number of isotopes that are produced routinely.

There are six permanent rigs in the reactor core which are used for radioisotope irradiation. One of them can be loaded and unloaded during reactor operation. The maximum flux which can be obtained for irradiation is as high as 4×10^{14} n/(cm²-s). The permanent rigs can be supplemented with temporary installations. ^{192}Ir is produced by irradiation in the core positions. The specific activity of the resulting product is higher than 350 Ci/g. The encapsulated isotope is used industrially for gamma radiography. ^{169}Yb, which is also produced, has the same application.

Eu is irradiated with the aim to extract ^{153}Gd, which is used for sources in bone scanners. ^{32}P and ^{35}S are examples of isotopes produced in R2, which are used for biological research. There in also a wide variety of radioisotopes being produced for medical research and therapy, such as ^{85}Sr, ^{89}Sr, ^{86}Rb, ^{155}Cd, ^{110}Ag, ^{51}Cr, ^{59}Fe, ^{45}Cs, ^{47}Cs, ^{90}Y,^{186}Re and ^{63}Ni.

A few other isotopes are produced routinely. ^{24}Na for example is delivered to the Swedish defence forces for training purposes.

The reactor has not been designed for the production of ^{60}Co. However, this isotope has previously been produced in limited quantities.

10 BEAM TUBE EXPERIMENTS

The R2 and R2-0 reactors serve as sources of thermal neutrons for a wide variety of basic research applications. The beam tubes at the R2 reactor are used for thermal neutron scattering experiments, see Figure 11 and Table 4. The R2-0 reactor, which is mobile in the pool, is in one position used as source for a neutron capture radiography facility and in the other position as source for a facility for nuclear physics and nuclear chemistry experiments based on an on-line isotope separator. Researchers from the universities have easy access to the facilities through the Studsvik Neutron Research Laboratory (15). The laboratory is organized as a department at the University of Uppsala but serves users from all Swedish universities. The instruments are also available for outside users.

In connection with the change of the reactor vessel, a large D_2O box was installed on the outside of the core box. The re-entrant beam tubes end at positions close to the thermal flux peak in the D_2O. Two tangential beam tubes were installed through a region in the biological shield which was previously inaccessible. The new beam tubes are rectangular with height 18 cm and width 8 cm. The larger vertical divergence of the beams increases the flux at the experimental positions. With these modifications substantial improvements in the thermal flux and in the ratio of thermal to fast flux at the experimental positions are achieved. The research performed involves, for example, structure determinations in crystals and amorphous systems, studies of magnetic phenomena in condensed matter and of excitations in disordered systems.

Table 4 Neutron scattering instrumentation at the R2 reactor.

Neutron beam tube	Instrumentation
H1	Service instrument
H3	Time-of-flight spectrometer for thermal neutrons
H5	Triple-axis spectrometer
H6	Diffractometer for protein studies
H7	Diffractometer for disordered structures
H8	Powder diffractometer
H9	Single-crystal diffractometer
H10	Spectrometer for polarized neutrons

A thermal neutron facility for neutron capture radiography (NCR) is located at the R2-O reactor, see Figure 12. A very pure thermal neutron field is produced by moderation of the fast neutrons from the reactor in a large D_2O volume positioned immediately outside the pool liner and adjacent to the reactor core, which is located immediately inside the pool liner. At the outer edge of the D_2O volume irradiations can be made in a thermal flux over a large area (30x30 cm^2). The thermal flux used in recent experiments has varied between $6 \cdot 10^8$ and $5 \cdot 10^9$ n/($cm^2 \cdot s$), corresponding to a reactor power of 25 to 200 kW. The facility is used extensively for biomedical research and has proved to be an efficient tool for studying the distribution of boron loaded compounds with a specific affinity for certain tumors.

A variety of nuclear physics and nuclear chemistry research programs are based on the on-line isotope separator OSIRIS at the R2-0 reactor. The main activity is aimed at studies of the properties of short-lived neutron-rich nuclides. The programs include determination of fission yields including branching ratios for gamma decay from fission products and determination of the anti-neutrino spectrum at a nuclear reactor. The system utilizes a novel method for plasma creation and allows higher temperatures, up to 2500 °C, and thereby shorter delay times for the released fission products. This has considerably increased the number of nuclides available and has increased the production yields of many short-lived isotopes by factors of 10^2–10^3.

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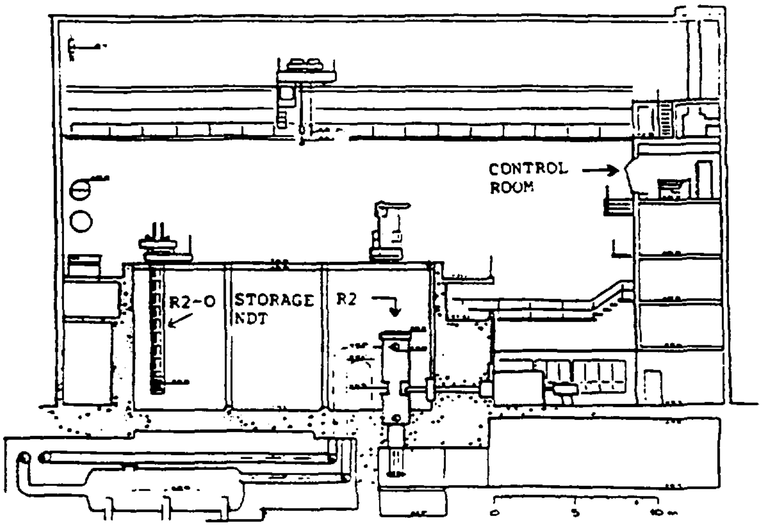

Figure 1
The R2 Test Reactor –
General Arrangement

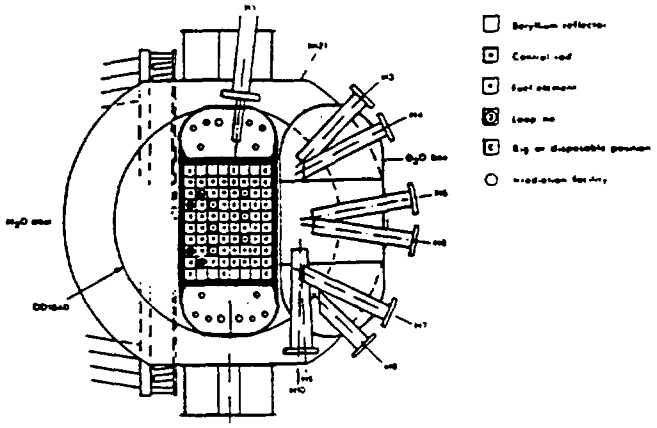

Figure 2
The R2 Test Reactor –
Core Configuration

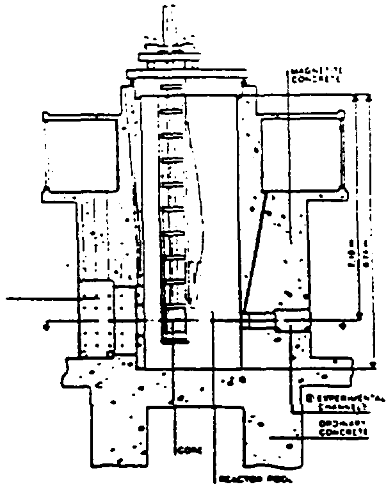

Figure 3
The R2–0 Reactor

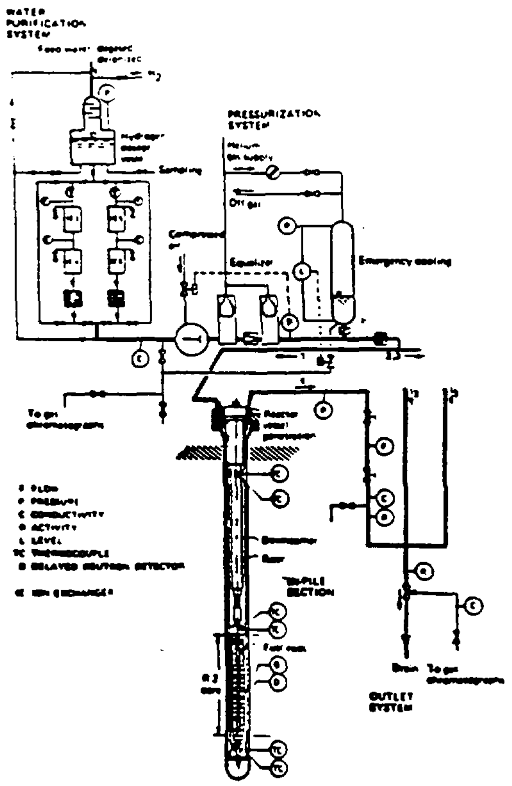

Figure 4
BOCA Rig in R2 –
Simplified Flow Sheet

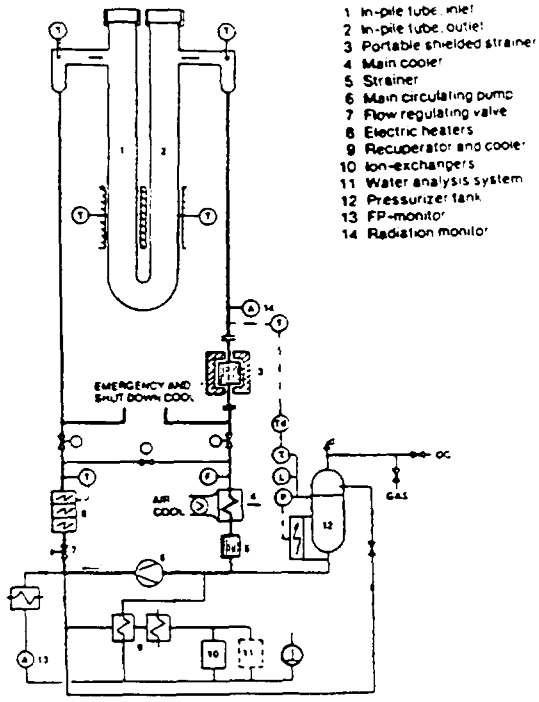

Figure 5
Loop No 1 in R2 –
Simplified Flow Sheet

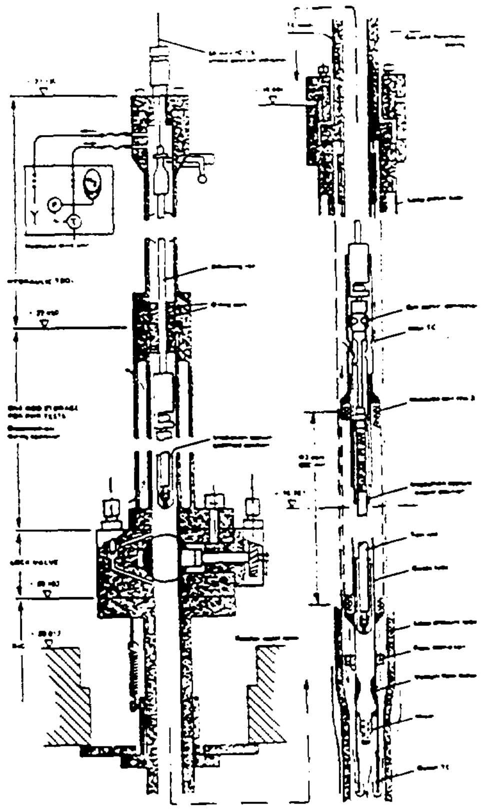

Figure 6
Loop No 1 in R2 –
PWR Ramp Test Facility

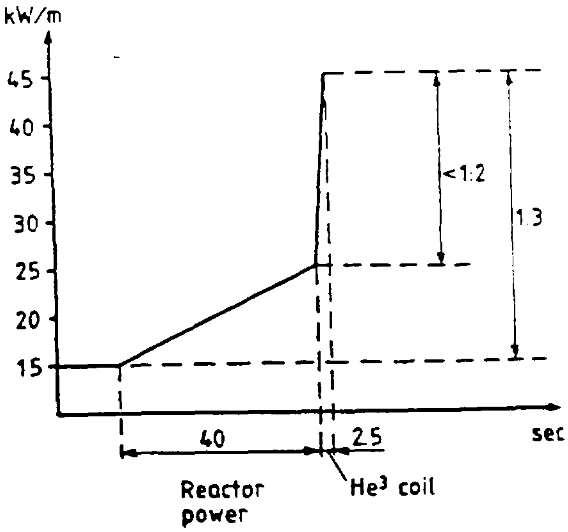

Figure 7
Rod Power vs. Time Diagram
for Double Step Up-Ramping

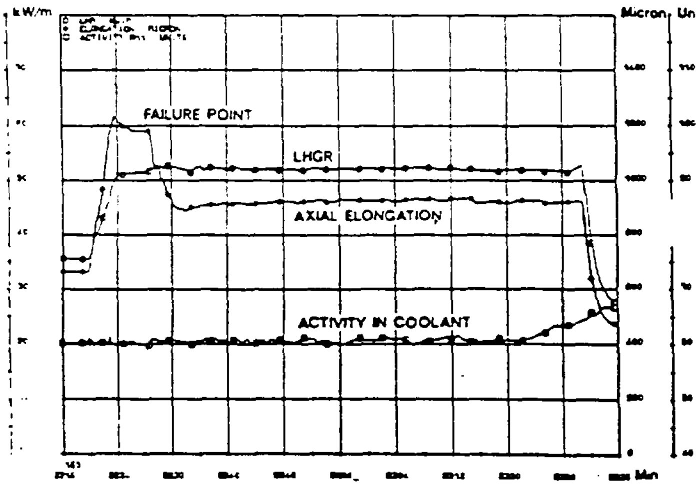

Figure 8
On-Line Measurements
During a Ramp Test

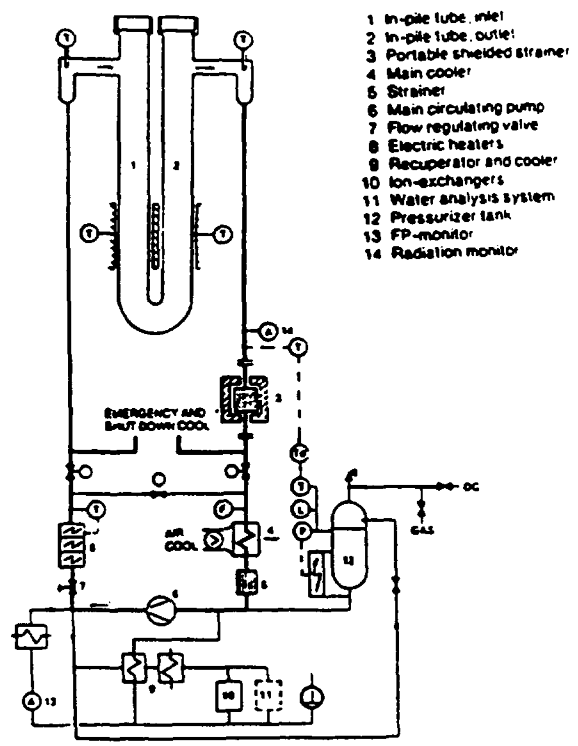

Figure 5
Loop No 1 in R2 –
Simplified Flow Sheet

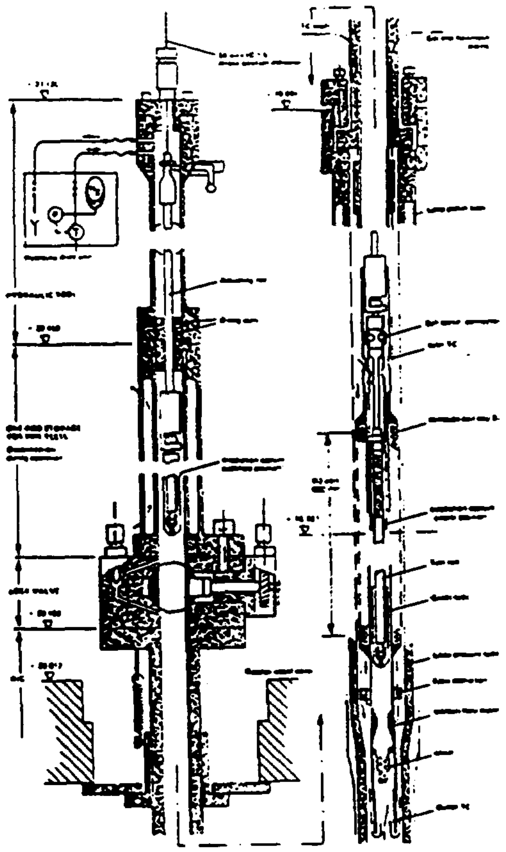

Figure 6
Loop No 1 in R2 –
PWR Ramp Test Facility

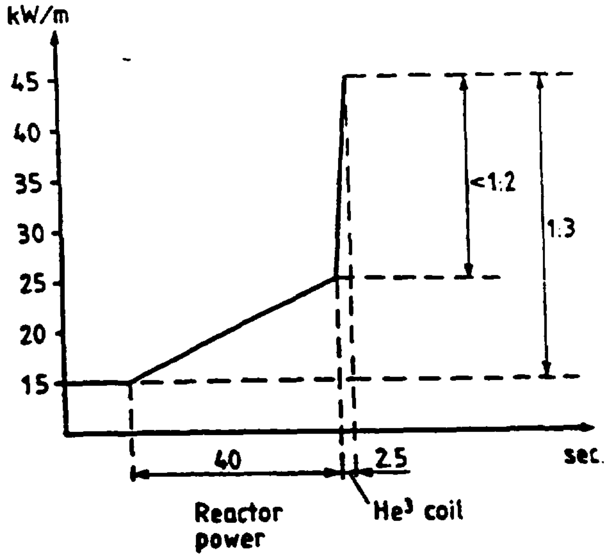

Figure 7
Rod Power vs. Time Diagram
for Double Step Up–Ramping

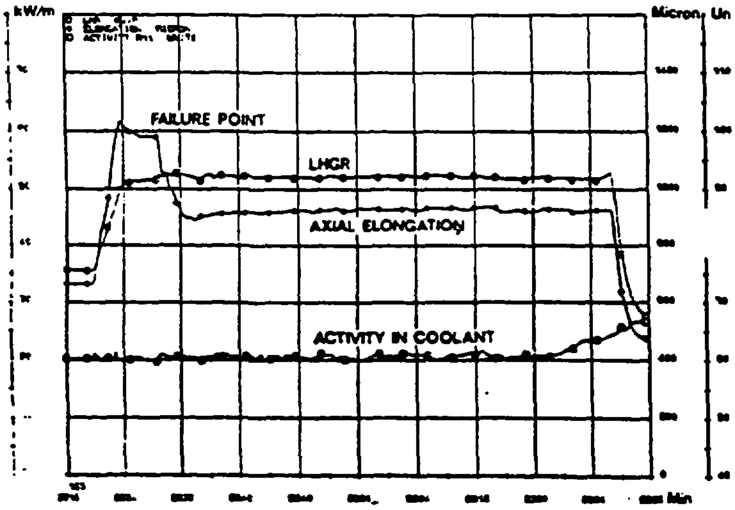

Figure 8
On-Line Measurements
During a Ramp Test

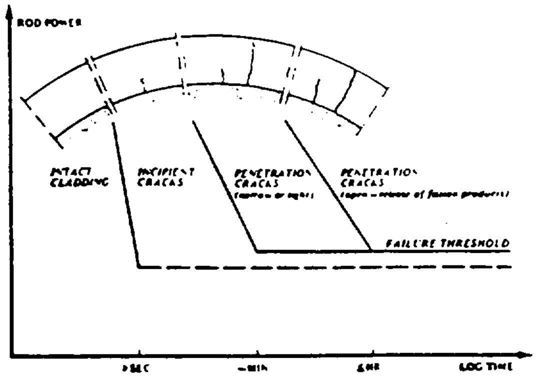

Figure 9
Schematic PCI Failure Progression Diagram

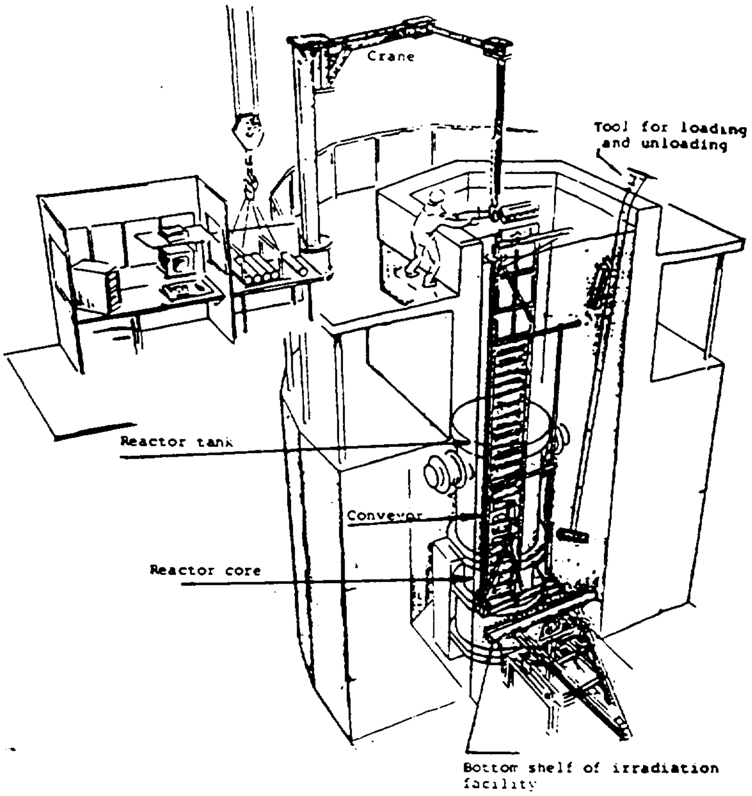

Figure 10
Silicon Irradiation Facility

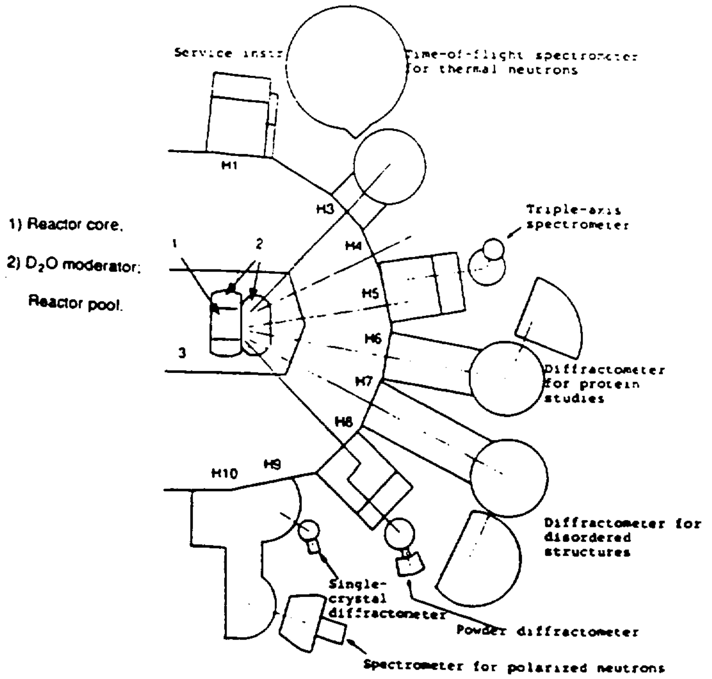

Figure 11
Neutron Scattering Instrument

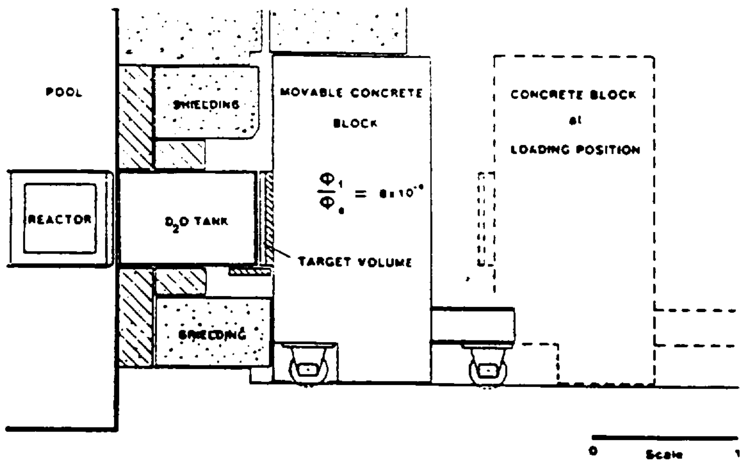

Figure 12
The R2–0 Thermal Neutron Facility for Neutron Capture Radiography

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