Operating Experience at the Ågesta Nuclear Power Station

Edited by S. Sandström
Information Office

AKTIEBOLAGET ATOMENERGI
STOCKHOLM, SWEDEN 1966
ERRATA:

On the preface page there is a misprint in the last paragraph on the first line

506,000
-59,699-
FAILURE OF INCONEL INSPECTION TUBES

The leakage mentioned on p. 21 has been subject to a thorough investigation which is described in our report AE-245. The results are summarized below.

The reactor had been in operation for 250 days (150 days at full temperature) when a leakage of heavy water was detected in September 1964. The leakage was found to be due to cracks in three inspection tubes, located in the periphery of the flat top lid as shown in Fig. 1. The heavy water flows inside the inspection tubes (OD 90 mm, wall thickness 5 mm) made of Inconel 600 and on their outside the tubes are exposed to light water of about 217°C temperature.

In a sample from one of the leaking tubes, one big and three small circumferential cracks were found. The big crack was located on the tube surface facing the centre of the lid, whereas the small cracks were in a crevice region on the opposite side of the tube as shown in Fig. 2.

The tubes had been exposed to 217°C light water, containing 1-4 ppm LiOH (later KOH) but only small amounts of oxygen, chloride and other impurities. Some of the circumferential cracks developed in or at crevices on the outside surface. At these positions constituents dissolved in the water may have concentrated. The crevices are likely to have contained a gas phase, mainly nitrogen. Local boiling in the crevices may also have occurred. Some few cracks were also found outside the crevice region.

Irradiation effects can be neglected. No surface contamination could be detected except for a very minor fluoride content (1 μg/cm²). The failed tubes had been subjected to high stresses, partly remaining from milling, partly induced by welding operations. The possibility that stresses slightly above the 0.2 per cent offset yield strength have occurred at the operating temperature cannot be excluded.
The cracked tube material contained a large amount of carbide particles and other precipitates, both at grain boundaries and in the interior of grains. The particles appeared as stringers in circumferential zones. Zones depleted in precipitates were found along grain boundaries. The failed tube turned out to have an unusually high mechanical strength, likely due to a combination of some kind of ageing process and cold work (1.0 - 1.3 per cent plastic strain).

Laboratory exposures of stressed surplus material in high purity water and in 1 M LiOH at 220°C showed some pitting but no cracking after 6800 h and 5900 h respectively.

Though the encountered failures may have developed because of influence of some few or several of the above-mentioned detrimental factors, the actual cause cannot be stated with certainty. In the literature information is given concerning intercrystalline stress corrosion cracking of Inconel 600 both in caustic solutions and in plain, pure high-temperature water. At this stage it seems most difficult to establish which of these cases is relevant to the Ägesta failure. This is especially true as both these conditions give the same type of intercrystalline cracks.

The possible influence of cold work, i.e. residual stresses, originating from milling, indicate the necessity of more stringent delivery control. Also the probable effect of carbides and other precipitates should be avoided by selecting a low carbon, pure iron-nickel-chromium alloy, preferably vacuum-melted.

The failures encountered show that Inconel 600 under certain conditions may be susceptible to intercrystalline stress corrosion cracking in alkaline high-temperature water. It remains to be shown if a satisfactory performance of Inconel 600 for long operating periods can be obtained under similar conditions in water-cooled reactors by proper manufacturing control and judicious design, omitting high stresses and crevices.
Fig. 1. Pressure vessel lid structure.
Fig. 2.
Location of cracks in the investigated tube.
Sweden’s first nuclear power reactor Ågesta, achieved criticality on July 17, 1963. Full power (65 MWt) was attained on March 20, 1964. Ågesta is a heavy water cooled and moderated pressure vessel reactor used for production of electricity as well as for district heating. The design, assembly and construction etc, of the reactor was described in detail in a staff report by AB Atomenergi, "The Ågesta Nuclear Power Station" edited by B McHugh, which was published in September, 1964. In the book experiences from the commissioning and the first operation of the reactor were reported as well as findings from the extensive reactor physics studies made during this period.

The report now presented is written by members of the operating team at Ågesta since its start. It reflects in general the experiences up to the end of 1965. The Ågesta Log, however, covers the period up to the normal summer stop 1966.

The reactor has hitherto produced 50,600 MWh power of which 48,700 MWh have been electric power. In July 1965 the responsibility for the reactor operation was taken over by the Swedish State Power Board from AB Atomenergi, which company had started the reactor and operated it until the summer break 1965.

Issued in September 1966
Contents

1. SUMMARY DESCRIPTION OF PLANT CIRCUITS 1
   Kristian Kull
   Primary circuits 2
   Major auxiliary systems 3
   Secondary circuits 4
   Tertiary circuits 6
   Cooling systems 7
   Table 1: Principal particulars of Ågesta Power Station 9

2. ÅGESTA LOG 10
   Nils Rydell and Evert Ericsson
   Light water trials 10
   Low-power operation 11
   Full-power operation 12
   General experience 13

3. HEAVY WATER COMPONENTS 19
   Alvar Östman
   Quality standards 19
      Leakproofing 19
      Corrosion 20
      Service 20
      Cleanliness 21
   Operational record of components 21
      Reactor vessel 21
      Steam generators 22
      Valves 23
      Safety valves 25
      Circulating pumps 25
      Control rods 27
      Instrumentation 27
   Future improvements 27
   Conclusion 28

4. SPECIAL OPERATIONAL PROCEDURES 29
   Ingvar Holtz
   Reactor warm-up and cool-down 29
   Warm-up 29
   Nuclear warm-up 33
   Cool-down 36
   Vacuum drying of heavy water systems 36
   Operating instructions 37
5. INSTRUMENTATION AND CONTROL
Ingemar Myrén

Process instrumentation
  Temperature measurement
  Pressure measurement
  Flow measurement
  Level measurement

Monitoring of fuel elements
  Temperature measurement
  Fuel element failure detection

Process control

Nuclear instrumentation and safety controls
  Measuring equipment
  Control equipment
  Interlocking circuits

Conclusions

6. FUEL AND CONTROL ROD HANDLING
Lars Broström

Fuel and control rod assemblies
Refuelling machine
Fuelling operations

7. CHEMISTRY
Erik Lindén

Flushing
Degassing
Injection of gas and the formation of ammonia
Ion exchangers
Primary system sampling arrangements
Corrosion, corrosion products and corrosion product activities
Blanket gas
Reactor lid circuits
Secondary loops
Other systems

8. RADIATION HAZARDS
Bertil Mandahl

Leakage in the primary system
Personal doses
Special maintenance
Effluents

9. ELECTRICAL INSTALLATIONS
Ingvar Wetterholm

Arrangement and working
  Generator
  Switchboards for 30 kV and 6 kV
  Switchboards for 3 kV and 400 V
  Reactor section
  Main circulators and motor generators

Operating experience
  Stand-by power equipment
  Main circulators and motor generators
  Generating plant
10. REACTOR PHYSICS EXPERIMENTS

Göran Apelqvist and Pehr Blomberg

<table>
<thead>
<tr>
<th>Topic</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>Neutron economy</td>
<td>83</td>
</tr>
<tr>
<td>Definitions</td>
<td>83</td>
</tr>
<tr>
<td>Buckling and critical size</td>
<td>84</td>
</tr>
<tr>
<td>Moderator temperature</td>
<td>86</td>
</tr>
<tr>
<td>Moderator level</td>
<td>88</td>
</tr>
<tr>
<td>Burnup</td>
<td>89</td>
</tr>
<tr>
<td>Reactivity control</td>
<td>90</td>
</tr>
<tr>
<td>Power distribution in the core</td>
<td>94</td>
</tr>
<tr>
<td>Dynamic experiments</td>
<td>100</td>
</tr>
<tr>
<td>Special equipment</td>
<td>104</td>
</tr>
<tr>
<td>Results and continued experiements</td>
<td>105</td>
</tr>
</tbody>
</table>

Appendix 1  Gas and ammonia concentrations during operations 1964 - 1965

Appendix 2  Distribution of Fe and Ni and the activity of Mn$^{54}$, Co$^{58}$, Co$^{60}$ and Fe$^{59}$ in the ex-changers

Appendix 3  Dose rate check points in m$_{a}$ in coolant loop
1. SUMMARY DESCRIPTION OF PLANT CIRCUITS

Kristian Kull

The Ågesta Nuclear Power and District Heating Station was built in cooperation between the Swedish Atomic Energy Company, the Electricity Department of the City of Stockholm and the Swedish State Power Board. The reactor was designed for an output of 65 MWt and when operating at full power the station shall supply 10 MWe to the grid and 55 MWt for district heating in Farsta, a Stockholm suburb.

The first operating period at Ågesta terminated on June 10, 1964, after a practically continuous operation for 15 months, including 3 months full power operation. This period may be regarded as the termination of trials with the reactor during which time a comprehensive experimental program in reactor physics had also been carried out.

---

Fig. 1 Simplified flow chart of the Ågesta plant.
The various parts of the plant, Fig. 1 and Table 1, can be divided, in common with the thermal process, into primary, secondary and tertiary circuits with their auxiliary systems.

**Primary circuits**

The primary circuits comprise the heavy water filled high pressure system: the reactor, the four external main coolant circuits which transport heat to the main steam generators, and the pressure control and relief system. The inlet nozzles on the reactor vessel for the main coolant loops are merged in a manifold to which the tapered inlets to the 140 fuel assemblies are also connected. Each fuel assembly consists of an open shroud tube containing 19 rods of uranium dioxide, each with a seal-welded thinwall cladding of Zircaloy-2, Fig. 2 and 18. Each rod consists of four elements, making a total rod length of 3 metres.

The flow of coolant through the reactor is constant (designed for 4 x 250 kg/sec). Since heat generation is greatest at the centre of the core the fuel elements there get a greater flow of cooling water than those at the periphery. The differential is obtained by fitting throttling discs in the assembly inlets. The designed water inlet temperature when on full power (65 MWt) is 205°C (the actual temperature is 207°C) and the temperature rise in the fuel assemblies 15°. From the top of the fuel assemblies the water flows down between the shroud tubes in the moderator space, from which four nozzles lead to the external main coolant circuits. By means of the power regulating system, the temperature of the water in the

---

**Fig 2 Fuel element**
moderator space is kept at 220°C regardless of the power level. At 65 MW output the average fuel loading is only 3.5 W/g U02 (whereas the boiler elements for Marviken are designed for approx. 18 W/g U02). Despite the generous diameter of the rods, 17 mm, the fuel temperature remains low as a result of this low load. At the most heavily loaded point the temperature reaches some 1400°C (the melting point of the U02 being approx. 2800°C).

In the external main coolant circuits the water flow is about 7 m/s which at the time of the reactor’s construction was optimal due to the price of heavy water. After the steam generators the water passes the main circulating pumps of canned motor type. The total pressure drop in the circuits, including the reactor loops, is approx. 2.5 at.

The pressure control and relief system comprises a pressurizer vessel with safety valves, an electric boiler, steam pipework and pipework for the spray injection of water in the boiler and pressurizer. About 1 m³ of the 20 m³ volume of the pressurizer is occupied by water, the rest by steam. Steam generation in the electric boiler maintains the steam buffer in the pressurizer at 240°C, corresponding to a pressure of 33 atm.

**Major auxiliary systems**

The major auxiliary systems, completely filled with heavy water at the same temperature and pressure as in the reactor vessel, comprise the reactor annular cooling circuit, the routine purification system, the control rod drive systems, and certain parts of the fuel element failure detection system. In the routine purification system, however, the water is cooled from 207°C to under 40°C prior to entering the ion exchangers. It is reheated to about 190°C before returning to the main coolant circuits.

The high pressure section of the heavy water analysis system is connected to the cold part of the purification system. The other auxiliary systems are at atmospheric pressure and ambient temperature. They comprise the closed reactor blow-off system, which includes systems for deuterium and hydrogen recombination and for the drying
(by cooling) of blow-off gases; the draining system with its piston pumps for refilling heavy water to the primary circuits during reactor operation; the heavy water storage tanks and filling pumps; and a nitrogen blanket gas circulating system serving these atmospheric-pressure systems.

During operation only the drain tank normally contains heavy water, this to ensure priming of the piston pumps. The level is maintained by the intermittent automatic operation of these pumps. Otherwise the systems are filled with gas at a slight pressure, 0.5 m WG. Atmospheric release of blow-off gases from the reactor or of blanket gas is through the system for active waste gases, in which such gases can be compressed and held in tanks for the necessary decay period before release.

Secondary circuits

The secondary circuits, Fig. 1 and 3, contain light water. Circuits forming part of the main thermal process are adjacent to the primary circuits in the four main steam generators arranged round the reactor.

Fig. 3 Simplified flow diagram for secondary, tertiary and quartic circuits at Ågesta.
Each steam generator contains about 14 tons of light water and the free surface of the water is at about twice the height of the vertical hairpin tubes through which the heavy water passes. The steam pipes from the generator domes are united in the main steam generator room in a single main running to the turbine hall, where the main branches lead to the turbine and the pair of dump condensers.

The feed flow to each steam generator is individually controlled according to both the steam take-off and the water level in the generator. Feed water enters the steam dome above water level. The steam generators were designed to supply 26 tons of steam per hour at a pressure of 14 atg under full load conditions. The steam passes to the turbine in saturated dry state.

The power control system for the reactor is based on a constant flow of heavy water, at constant delivery temperature, from the reactor to the steam generators. This implies that conditions in the steam domes determine the power drawn from the reactor. For example, the load is 20% (13 MWe) at a steam pressure of 21 atg. The temperature of feed water supplied to the steam generators is 105 - 110°C.

The turbine installed at Ågesta, an existing back-pressure unit which has been slightly modified, has the following main particulars:

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Unit</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Steam pressure at throttles</td>
<td>atg</td>
<td>12.5 - 20.5</td>
</tr>
<tr>
<td>Steam temperature, saturated steam</td>
<td>°C</td>
<td>192 - 215</td>
</tr>
<tr>
<td>Max. steam flow at 12.5 atg</td>
<td>t/h</td>
<td>105</td>
</tr>
<tr>
<td>Back pressure in condenser</td>
<td>ata</td>
<td>0.5 - 1.25</td>
</tr>
<tr>
<td>Electrical output</td>
<td>MW</td>
<td>10</td>
</tr>
</tbody>
</table>

The condenser is cooled by the district heating water. Instead of passing through the turbine, the steam may be diverted to two dump condensers, each dimensioned to handle 35 MWe and featuring vertical straight tubes through which the district heating water passes. The load on these condensers is regulated by valves in the steam lines. The level of condensate in the condensers is automatically regulated by the steam pressure, the back-pressure of the water head and flow in the condensate line to the feed water tank.
Another group of secondary circuits is that comprising the reactor warming and cooling system and the reactor-head temperature control system. During regular operation, the latter system has the purpose of maintaining the head temperature equal to that of the reactor vessel flange, which is cooled by the annular cooling circuit.

For warming the reactor and the steam generators heat is supplied from a hot-water circuit via heat exchangers through which water from the steam generators and the reactor-head circuit also circulates. The steam generators transfer the heat to the reactor primary coolant circuits. When expansion causes the water level in the reactor to rise to the pressurizer vessel, however, the electric boiler for this vessel takes over the warming function for the pressure control and relief system. The same circuits are employed for cooling-down, with cooling water replacing hot water in the initial heat exchangers.

The hot water mentioned is supplied by an electric boiler rated for 4 MWe and located in the turbine hall. This boiler has three vertically installed phase electrodes. Between these and the circular earthing electrodes Teflon screens can be raised and lowered to regulate the input effect. The boiler is only partly filled with water. The steam space above the water comprises the pressurizer for the hot-water circuit.

Tertiary circuits

The Farsta district heating system represents the tertiary circuit in main thermal process, cooling the turbine condenser and the two dump condensers. The return water passes, in sequence, the turbine condenser, the pumps (three, one of which is a stand-by), and the dump condensers, which are connected in parallel.

The water in the district heating supply line varies in temperature from 78°C to 115°C, depending on the outdoor temperature, while the return temperature is 55-60°C. The load is varied by flow through infinite speed regulation of the pumps. The maximum flow that can be supplied by the pumps is 2400 tons/h.
Cooling systems

The largest of the four cooling systems at Ågesta (A in Fig. 1) can act as a quartic circuit in the main thermal process, as in the turbine hall the district heating water can be by-passed through four recoolers in which excess heat is removed by a cooling-water circuit. The recoolers are designed to remove 55 MWt with the district heating water at 100°C.

The cooling tower for the quartic circuit which has the same capacity, is of a new type featuring phenol-resin impregnated fillers for the main cooling effect. At full capacity, 55 MWt, 75 tons of fresh water must be supplied to the cooling tower circuit every hour, largely to replace evaporative losses. Making full use of the cooling tower, the turbine and reactor may be run at full power even with the Farsta circuit shut off. The cooling tower is also used during cooling-down of the reactor. All pipes and fittings in the cooling system are of stainless steel.

Another, closed cooling system (B in Fig. 1), comprising four towers with a total capacity of about 1700 kW, cools during normal operation all appropriate heavy water filled components and the reactor-head temperature control circuit. The pressure in this system is maintained constantly at a lower level than that on the heavy water side in the cooled components. When the reactor is cold it is therefore necessary to maintain a partial vacuum in parts of the system. The pipes and fittings are of carbon steel.

An open cooling system (C in Fig. 1), in general similar to the major system mentioned initially but with three small cooling towers of approx. 4500 kW total capacity, cools the light water filled components in the plant. This system is all stainless steel with few exceptions.

All cooling towers are located in the open air on top of the rock outcrop in which the reactor is contained.
An emergency cooling system (D in Fig. 1) with a tank of about 600 m$^3$ capacity is installed within the reactor containment. The temperature of the water in this system is kept at 4°C by a small (40 kW) refrigerating plant. This system is for use in emergencies such as blow-off of reactor decay heat steam through the safety valves, with a possibility of cooling the blow-off line continuously and thus condense the released heavy water steam.

Another conceivable emergency is a fracture in the primary system, in which case water from the emergency cooling tank could be sprayed from sprinklers in the spaces nearest the reactor. If the reactor was to lose all its coolant (Maximum Credible Accident, MCA), the cooling water can also be introduced at the top of each fuel assembly by means of the fuel element failure detection system, preventing melting of the fuel rods. The used cooling water is collected and returned to the tank, in which a low temperature can be maintained by putting into service a 1250 kW capacity refrigerating cooler placed outside the containment.
Table 1  **Principal particulars of Ågesta Power Station**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Unit</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power of reactor</td>
<td>MW</td>
<td>65</td>
</tr>
<tr>
<td>Electrical power</td>
<td>MW</td>
<td>10</td>
</tr>
<tr>
<td>Heat supplied to Farsta</td>
<td>MW</td>
<td>55</td>
</tr>
<tr>
<td>Primary system design pressure</td>
<td>atg</td>
<td>33</td>
</tr>
<tr>
<td>Heavy water inlet temperature at reactor</td>
<td>°C</td>
<td>207.5</td>
</tr>
<tr>
<td>Heavy water outlet temperature from reactor</td>
<td>°C</td>
<td>220.0</td>
</tr>
<tr>
<td>Coolant flow through core</td>
<td>kg/s</td>
<td>1180</td>
</tr>
<tr>
<td>&quot; &quot; central fuel assemblies</td>
<td>kg/s</td>
<td>14.4</td>
</tr>
<tr>
<td>&quot; &quot; middle &quot; &quot;</td>
<td>kg/s</td>
<td>8.9</td>
</tr>
<tr>
<td>&quot; &quot; outer &quot; &quot;</td>
<td>kg/s</td>
<td>3.8</td>
</tr>
<tr>
<td>Total fuel charge (UO₂)</td>
<td>t</td>
<td>18.5</td>
</tr>
<tr>
<td>Number of fuel assemblies</td>
<td></td>
<td>140</td>
</tr>
<tr>
<td>Number of working control rods</td>
<td></td>
<td>18</td>
</tr>
<tr>
<td>Reactor vessel inner diameter</td>
<td>mm</td>
<td>4555</td>
</tr>
<tr>
<td>Reactor vessel height</td>
<td>mm</td>
<td>5000</td>
</tr>
<tr>
<td>Reactor vessel wall thickness</td>
<td>mm</td>
<td>70</td>
</tr>
<tr>
<td>Total heavy water charge</td>
<td>t</td>
<td>74</td>
</tr>
<tr>
<td>&quot; &quot; in reactor</td>
<td>t</td>
<td>51</td>
</tr>
<tr>
<td>&quot; &quot; in external circuits</td>
<td>t</td>
<td>23</td>
</tr>
<tr>
<td>Steam pressure in generators, full load</td>
<td>atg</td>
<td>16.2</td>
</tr>
<tr>
<td>&quot; &quot; 20% &quot;</td>
<td>atg</td>
<td>21.0</td>
</tr>
<tr>
<td>Light water contained in each steam generator</td>
<td>t</td>
<td>14</td>
</tr>
<tr>
<td>Secondary steam flow, full load</td>
<td>t/h</td>
<td>102</td>
</tr>
<tr>
<td>Supply temperature in district heating system</td>
<td>°C</td>
<td>78 - 115</td>
</tr>
<tr>
<td>Return</td>
<td>°C</td>
<td>55 - 60</td>
</tr>
</tbody>
</table>
2. ÅGESTA LOG

Nils Rydell and Evert Ericsson

The primary coolant system functioned satisfactorily during the first full-power operating period in spring 1964. Heavy water leakage was estimated to be a mere 25 g/h. But in the autumn of the same year a leak in the reactor lid made a long shutdown for repairs necessary. The repair was completed in the beginning of 1965, and the operation was then resumed. The operation results during the season 1965-66 were quite satisfactory.

light water trials

Commissioning of the reactor plant was commenced in December 1962, before the final completion of erection work. Trials were first carried out with light water in the reactor systems, the following being the dates for the principal events:

November-December 1962, filling of reactor systems with nitrogen, flushing, filling with light water.

December 1962-January 1963, tests of reactor systems using cold water, pressure testing, selected trials of valves, pumps and control equipment.

February-May 1963, tests of systems using heated water: warm-up and operation for periods of 1-2 weeks at progressively higher temperatures. Regular operating temperature 210°C was achieved in the beginning of April.

June 1963, draining and vacuum drying of all heavy water systems.

The rinsing operation showed that the systems were satisfactorily clean after installation. The defects revealed and necessary modifications during commissioning were no greater than could be dealt with immediately, except that a failure in the stator winding of one of the main circulating pump motors required disassembly and return of the motor to the manufacturer for repair.

The repeated heating-up, operation at full temperature and cooling of the reactor systems gave a good insight into the operating characteristics of the various systems, even though light water was used and no power was developed in the core.
The light water trials also confirmed the leakproof condition of the heavy water systems and thus that they could safely be operated with heavy water containing high levels of activity.

After the completion of light water trials the reactor had to be drained and all light water residues dried out before charging with heavy water. In order to ensure "absolute" dryness the vacuum drying method, unique for a plant of this size (about 250 m$^3$ and including some 700 valves), was employed. The results comfortably exceeded requirements.

**Low-power operation**

After the completion of heavy water filling, the reactor was put into low-power operation in July 1963. This period continued until February 1964. During the first three months criticality tests were carried out with the reactor cold, after which the tests continued with the reactor at the regular operating temperature of 210°C.

The salient dates were as follows:

- July 1-10, 1963: heavy water filling.
- July 11-17, 1963: insertion of fuel elements for first criticality with cold reactor.
- July 17, 1963: reactor went critical for the first time with 23 fuel assemblies in the core.
- August-October 1963: criticality tests continued in conjunction with the execution of the cold-reactor physics measurement program. Number of fuel assemblies was increased progressively to the full charge, 140.
- November 1963-February 1964: heating to 210-215°C and criticality tests for hot-reactor physics measurements, as well as continued tests of systems and components. In all, the reactor systems were warmed-up and cooled seven times during this period.

The seven months period of low-power operation was necessary because of the comprehensive program of reactor physics measurements, though this did not prevent simultaneous testing activities so that the adjustment for high-power operation was well advanced for the portion of the plant which had been in use so far. The operating staff had time
to acquire confident familiarity with the supervision of the plant. Further weak points — all of a minor nature — in the heavy water systems were detected and remedied.

The comprehensive monitoring equipment for the reactor was tested thoroughly. The control rods were operated as much during this period as during several years of normal operation, while the refuelling machine was subjected to several intensive periods of use, including the insertion of 140 fuel assemblies and 20 control rod assemblies.

A measurement of fundamental significance was to determine the critical moderator level at temperatures up to the designed operating temperature, 210°C. This implied operation of the reactor with virtually no margin of pressure before gas release and boiling in the heavy water circuits occur. The measurement, which could not be completed (incidents included the unintentional initiation of scram shutdown, in which damage was caused to control rods), emphasized most emphatically that the Ågesta reactor is a power reactor offering little flexibility for experimental needs.

Repeated control tests of all motor operated process valves resulted in an improvement (at the price of extensive checking procedures and adjustments) from 10-15% failure to 6-7%.

**Full-power operation**

Simultaneously with the termination of the low-power reactor physics measurement program early in March 1964, the station steam and district heating systems, including the recooler circuits, were tested as far as possible. However, as it was then not possible to start the turbine all steam must be passed to the dump condensers.

Power was raised successively from 13 to 35 and then to 65 MWt at weekly intervals. On March 9, when power was raised to 13 MW, the first heat was supplied to Farsta and on March 20 the reactor developed full power.
Full-power operation continued thereafter until the close of the planned operating season with only two scrams, involving in all some 24 hours' stoppage. However, as spring advanced heat consumption in Farsta dropped from 40-45 MW in March to 10-15 MW without it being possible to increase the recolder cooling capacity by a corresponding amount. The mean output during this initial operating period was, therefore, only 45 MW. One of the scram shutdowns was due to short-circuiting of the current supply and the other to a testing fault. On one of the occasions the Farsta boiler plant had to assume the heating load for 24 hours.

The turbine became available on May 6, when the generator was synchronized on the local grid for the first time. Electricity generation on a continuous basis started on May 13 and the full output of the turbo-generator set, 10 MW, was achieved on May 22. The reactor was shut-down for the summer on June 10 as scheduled.

The commissioning program was more comprehensive and time consuming than need be expected for future nuclear power stations. The lengthy low-power period was justified in view of the fact that this was the first time a heavy water reactor of this type had ever been available for measurements of core reactivity, control rod effects, power distribution, etc. The commissioning of a familiar type of reactor should not involve more than some four months' effective trials prior to going critical, two months for full core charging and the assessment of control rod effects and reactivity coefficients, and one or two months for running-up to full power.

The heavy water systems functioned satisfactorily during this initial operating period. Any troubles are referable to the secondary and tertiary systems, which had not hitherto been tested under realistic conditions.

General experience

The leak tightness of the heavy water systems is of vital interest for the future commercial attraction of this type of reactor. Experience gained during the first operating season was very encouraging—no noticeable escape of heavy water occurred and
leakage at operating pressure was estimated on the base of tritium measure-
ments in the ventilation air and the light water systems to be approxima-
tely 25 g/h.

However, a substantial leakage of heavy water (about 900 g/h) to the vessel
head was discovered when operation was resumed after the summer shut-down in
1964. This leakage was of such magnitude as to necessitate immediate tra-
cing and repair. Leaks were discovered in as many as three of the eleven
inspection pipes (of Inconel 600) penetrating the lid close to its peri-
iphery. There was an obvious risk that similar leaks would also develop
in the remaining eight pipes. It was therefore decided that all pipes
should be sealed against contact with the light water in the lid by an extra
steel plate welded to the inside of the lid (op. p. 21). The welding area
was barely accessible from outside and the work area on the lid severely
restricted by the piping arrangements. The work therefore proceeded slowly
and was not completed until early next February.

Another heavy water leak was discovered at a terminal entry to the stator
winding of one of the glandless circulators in the primary circuit. This
was due to a corroded soldered joint between the aluminium stem and the
aluminium winding (see fig. 3a). The stator winding is inside the high
pressure primary circuit. Corrosion of the joint had increased the contact
resistance until the stator current melted the nylon insulation, which was
squeezed out by internal pressure, providing a leakage path for the heavy
water.

A similar defect might develop in any of the other cable joints and under
unfavourable circumstances conceivable lead to a major loss of heavy water.
It was therefore decided that all pumps should be rewound with stator win-
dings of copper, where the soldered joint should be fully reliable.

This was carried out during the spring and summer on one pump at a time.
The plant was therefore operated on three pumps for most of the period,
which at first gave problems with the hydraulic control rod system after
the restart of the plant. A series of shut-downs occurred due to individual
control rods falling from their operating position. This could be traced
to gas accumulating in the hydraulic system. A few provisional changes
had to be made in the operating procedures in order to overcome this diffi-
culty.

These incidents delayed continuous operation another month, but from the
beginning of March the plant operated until the summer shut-down in June
without any further trouble.

An enriched fuel element of the Marviken type was test irradiated during
the spring. Another test program initiated during this period concerned nuclear heating of the plant from room temperature. The results of both these tests were very satisfactory. Nuclear heating will be used as the normal starting procedure in the future, provided the necessary amendment to the plant license is granted.

The operating season 1965/66 was on the whole very successful. A number of disturbances occurred initially chiefly in connection with test operations and fuel irradiation experiments. But during the winter season December-March and further on to the summer shut-down the plant operated remarkably smoothly. 316,700 MWh of thermal energy was produced with the reactor on load during 5200 h. Plant availability during the period October-May was 89.2% including all shut-downs, and 97.5% if shut-downs in connection with planned experiments and fuel tests are subtracted.
The sensitivity of the detection system for heavy water leakage gradually increases with the build-up of tritium in the heavy water. The leakage to the air was consistently 50-50 g per day except for one week when it rose to 300 g per day. This was traced to a leaking bellows-sealed valve. The leakage was stopped by tightening of the stuffing-box which backs up the bellows seal. The leakage to the light water circuits is smaller than the present limit of detection, 20 g per day. Even including occasional losses as mentioned above, the total leakage of heavy water in the period 1965-66 has been very small (less than 1% of the inventory) compared to the leakage of 2% per annum or so which is usually assumed in operating costs for heavy water reactors. The operating history of the plant up to June 1966 is summarized in diagrams 1-4.

A great improvement over design data has been noticeable in both the primary and secondary circuits of the thermal process. The main circulating pumps deliver about 20% higher flow, mainly due to the flow resistance on the primary side of the steam generators being less than anticipated. The lowest heavy water temperature at full power is therefore 207.5° instead of the estimated 205°C. Average heat transfer coefficient for the heating surfaces in the steam generators has proved better than expected due to the absence of the anticipated deposits. At full power (102 t/h steam) the steam pressure in the generators is 16.2 atg and at the turbine throttle 15.2 atg, as against the contract figures of 14 and 13 atg respectively.

At a thermal output of 65 MW (102 t/h steam) the turbine was found to have reserve capacity. Only two of the three regulating valves had then opened. The dump condensers can each handle about 40 MW as against the specified capacity of 35 MW. The recooler circuit, however, which was designed for 55 MWT, could not handle more than about 35 MWT. This is because when outdoor temperatures are over 0°C the supply temperature of the district heating water, which is also the primary medium in the recoilers, must be kept at 78-85°C instead of the envisaged 100°C as a result of the terms of the district heating contract.

Data obtained with relation to external primary coolant circuits, steam generators and steam systems show that considerably higher outputs than 65 MW can be absorbed without further modification. Fig. 4. As seen an increase in the temperature rise of the heavy water in the reactor to the designed figure of 15° would mean a reactor output of 78 MWT, corresponding to about 123 t/h steam at about 15 atg.
Diagram 1

1964

- Summer shut down
- Physics measurements
- Station preheating
- Lid leakage discovered. Station cooled down.

March | April | May | June | July | August | Sept.
Diagram 2

October - November 1964

- Lid Repair

1965

- 25.1 - 29.1
  Proof tests of repair work

- 25. - 26.2
  Removal of main circulator
  Plant restart

- 21.3
  Tests with nuclear preheating

- 23.4
  Reinstallation of main circulator

- 22.
  Removal of second main circulator

- May

- June

Calibration of can failure detection system
Diagram 3
Diagram 4.
Studies also reveal that the fuel elements will stand outputs up to 80 MWt without overloading. Taken together, the turbine, the district heating system and the recooler circuit should be able to absorb 85-100 MWt during the cold season.

The regulating characteristics of the reactor are good. It easily adapts to load variations, including crash stoppage of the turbine, when the reactor is cut down to the intermediate output of 13 MW, at which level output reduction is inhibited in conjunction with blow-off to one of the dump condensers through an automatic pressure limiting valve.

Tests with the automatic regulating effect of the control rods interrupted verified that the reactor also has very good self-regulating characteristics in response to load variations, since the temperature of the moderator gives a negative reactivity feedback. In conjunction with the similarly negative reactivity feedback of the fuel temperature, this also results in a strongly stabilizing effect in the event of reactivity disturbance.
A conclusion that may be drawn from work in the control room is that the plant is over-complicated and therefore defies ready comprehension. This is partly because components have been designed and purchased by different companies. Separate drawing offices have worked over the same problems with the results that many opportunities for the simplification of auxiliary systems and the coordination of their functions have been lost, components of comparable significance as regards operational dependability have been contracted to differing standards, and layout drawings, system designations, etc, have not been standardized.

A plant of this size must be designed as a whole, with all those participating in the work under a common project administration and in one location.

While operating results have been satisfactory to date, the interruption suffered during the autumn of 1964 served as a reminder that the plant may yet conceal weaknesses that remain to be discovered. Several years' operation are required before a final judgement may be passed on the quality and suitability of the installations.
3. HEAVY WATER COMPONENTS

Alvar Östman

Inspection of the secondary side of the steam generators revealed that the make-up water pipe had become loose in one type of generator and that bottom plates for make-up water distribution had fallen down in another type. Necessary improvements in the remote operation of valves have been obtained by fitting ball bearings and universal joints in manual linkages and by increasing the operating torque in motor-powered valves. Simpler and sturdier components are needed in nuclear power plants.

The thermal process at the Ågesta power and district heating station has worked well. All design values for flow, pressure, temperature, etc, have been achieved, in some cases with a comfortable margin. Those difficulties which have arisen, and which to a certain extent remain to be solved, are due to shortcomings in components and the technical complexity of certain parts of the installation.

Quality standards

In recent years there has been discussion of the quality standards that should be required in a nuclear power plant. Advocates of lower standards point out the savings to be made through using conventional versions of components, while defendants of high standards maintain that conventional components are not sufficiently leakproof nor sufficiently reliable.

Leakproofing

Radioactivity must not be allowed to escape into the containment atmosphere nor moisture to damage equipment, and furthermore the cost of heavy water is an appreciable item, though this should not be allowed to play too great a role.

Leaks have been prevented by the use of welding wherever possible. Flanged connections are featured only on the reactor vessel, heat exchangers, pumps and certain other components. In the few cases
where couplings have been used, leakage has been observable. In view of this, an all-welded system may be regarded as necessary for reactor applications.

**Corrosion**

Corrosion products passing through the core become radioactive and in turn spread radioactivity to other parts of the installation, making service work difficult or, in the worst case, impossible. Because of corrosion risks an austenitic light welded steel with extra low carbon content has been used. The choice has proved wise. The material has not corroded and of some 40,000 pipe welds only one has been found to leak during both trials and regular operation. Irradiation channels of Inconel have suffered damage on the other hand.

**Service**

Components in a fossil-fuel power station can be supervised continuously, while certain vital parts in a nuclear power station are rendered inaccessible due to core radioactivity. The components must be designed with this in mind. More than this: the number of components and systems must be kept low, and the requirements stipulated in respect of each component must be reasonable.

The main steam generators at Ågesta include shut-off valves for steam and make-up water. These valves are of conventional type, made to good normal standards. This means that the spindle glands must be tightened at regular intervals. On several occasions during the spring of 1964 it looked as though district heating in Farsta must be interrupted to allow servicing of these valves, which a closed-circuit TV in the m.s.g. room had shown were leaking.

Repeated incidents of this kind demonstrate that components needing service must be accessible. No service demands can be tolerated in respect of any component which cannot be rendered accessible. Since this latter demand is virtually impossible to satisfy - American experience with light water reactors confirms that - the only remedy is to make all parts accessible, or expect to have to shut down the reactor for servicing. Obviously power producers very much wish to avoid such shutdowns.
Cleanliness
Possibly the potentially most costly factor in component manufacture is the demand for cleanliness. The advantages of a high standard of cleanliness have been clearly confirmed at Ågesta. In view of the experience now gained there, demands for cleanliness in reactor components should not be reduced.

No great encouragement can be obtained from the record of the conventionally erected components at Ågesta, not as regards leakage nor cleanliness. Leaking past valves and safety valves and choking of small-bore tubes in heat exchangers have been troublesome. The material used in these has mostly been carbon steel.

Operational record of components
Reactor vessel
During the first operating year the reactor vessel gave no trouble. In spring 1964 traces of tritium were observed in the light-water cooled head but the leakage was considered insignificant. During the first warm-up ready for the winter's operations, in September 1964, a high content of tritium was observed in the head. Check analyses revealed that heavy water had leaked to the head. The head was drained and filled with a mixture of nitrogen and helium, after which the Inconel irradiation channels, Fig. 5, were investigated for leaks.

Fig. 5 Irradiation channel penetration through vessel wall
1 - Inlet tube
2 - Irradiation channel
3 - Flange joint
4 - Vessel lid
5 - Silver seal
6 - Welded sealing shield

The total of 11 irradiation channels were incorporated to allow inspection of the reactor vessel wall, which has not been possible. On the other hand, they have fulfilled their envisaged function as irradiation channels for materials samples. Such channels are usually not included in a power reactor. When it was established that three of the channels were leaking, the risk of future leaks in the other eight was
deemed so great that it was decided to modify all the channels.

Holes were made in the topside of the reactor head and plates were welded round the leaky irradiation channels. The job was completed in January 1965. Fig. 5 shows the arrangement. One channel was cut out from inside the head and a plug welded over the hole. A sample from the extracted channel was submitted for metallurgical examination which showed that leakage from the primary circuit to the reactor-head circuit had occurred through cracks in the wall of the sample channel. These had started from the light water side and were intercristalline, the result of intercristalline stress corrosion. It was not, however, possible to pin the cause on any specific material factor, though certain circumstances suggest alkaline initiated stress corrosion. Further investigations proceed.

The irradiation channels intended to be a safety feature have thus actually reduced the availability of the reactor. Other examples of this can be cited. For example, the fuel element failure detection system and the fuel element temperature measurement system have proved unusable. Both involve small-bore Inconel tubes carrying heavy water through the head, which has further complicated head design.

Steam generators
The four main steam generators are of two different makes, in pairs, and of practically identical design, see Fig. 6. They have exceeded the intended operating data with good margin. The recorded heat transfer coefficient at full load is approx. 4 kW/m²°C or about 50% above the contract figure. The same experience is reported from American pressurized light water reactors. It may be taken to mean that heat transfer is not reduced by dirtying.
The steam pressure at full load has been measured as 16 atg as against the contracted 14 atg. Feed water supply caused trouble at first. Since the inlet is above water level in the steam dome at the top of the generators, steam could rise into the feed pipes under low load conditions and cause condensate knocks. The feed water arrangements have been modified to cure this.

Powerful condensate knocks have also occurred in the dump condensers. Here the power is regulated by valves in the steam lines which reduced the pressure at full load from 16 to 6 atg. When the turbine is running the dump condensers are on stand-by with the control valves on the steam side shut. Steam leakage into the condensers resulted in the knocks mentioned, so that the condensate valves must be closed for top-filling of the condensers. A switch to regulation on the condensate side is being considered.

Inspection of the secondary side of the steam generators revealed that screwed connections had loosened. In one type of generator the downward bent feed pipe had partly come loose, in both units. In the other type the bottom plates for feed water distribution had come loose and fallen down. The dropping of parts in this way has been reported from other reactors – for example, in the boiling light water reactor at Big Rock Point in the US a control rod assembly was jammed by a fallen bolt.

The staying of tube nests in heat exchangers seems to be a difficult problem, even though such tubes have not been observed to suffer vibration at Ågesta. Heavier tubes and sturdier designs should be sought.

Valves
Bellows sealing valves have been employed at Ågesta in pipes up to 50 mm diameter, see Fig. 7. The bellows have suffered damage in a few cases. Valves of this type never leak to the environment as long as the bellows remain intact.
Wedge type valves with two glands and intermediate drainage, as in Fig. 8, have been used in pipes from 80 to 225 mm. Experience of these has been good, except for the glands from which chlorides have been drawn. Leakage through the intermediate drain lines from the wedge type valves has between 0.1 and 0.5 kg/24 h for a total of 20 valves.

The remote operation of valves has proved troublesome. On manually operated valves it has been necessary to introduce ball bearings and universal joints to overcome the great frictional resistance. On motor powered valves the operating torque has been raised as far as possible with regard to material stresses. Nonetheless it was estimated that during the spring of 1964 some 6-7% of attempted valve operations failed. The reason is presumably that the valves
stand open or closed for long periods and are thus more inclined to stick than if they were operated more frequently. Valve operation at regular intervals has therefore been included in operating instructions.

Safety valves
The main safety valves of slow opening back-pressure actuated type are remarkably leakproof. The average leakage per valve, 150 mm pipe connection, has not exceeded 0.1 kg/h during operation at approx. 240°C and 33 atg.

Circulating pumps
The four main circulators are of canned type with wet stators, Fig. 9. They have worked without any trouble during heavy water operations except for a leakage which occurred past the cable leadthrough to one of the pumps. The leak was so serious that the pump had to be disassembled and the seal modified. It was also decided to replace the aluminium winding by copper, a modification which will be applied to the other circulators as opportunity arises. During light water trials a failure occurred on a stator winding which is of aluminium.

Leakage to stators has been observed in two of nine smaller circulating pumps of canned type with annular circulation and dry stators. Disassembly showed that the assembly welding of the circulating cavity was leaking.

Piston pumps for refilling during operation have been run with a reduced stroke in order to minimize knock in the delivery lines. A leaky refuelling standpipe seal, Fig. 10 and 11, flooded a bottom drainage tank through an intermediate drain line this in turn filling the inter-gland space of the piston rod on the pump, Fig. 10 and 12.
The lower section of the piston rod gland failed to retain the water, which seeped along the rod. About 175 kg of D$_2$O was lost.

Fig. 10 Level chart of components which have caused heavy water leakage

Fig. 11 Sealing of refuell channel

Fig. 12 Piston pump (see also fig. 10)
Control rods
The control rod drive assemblies have functioned satisfactorily, despite their complicated design. The 25 control rods fitted in the reactor (only 18 are required at present) have performed a total of over half a million step movements. Shock absorbers were damaged during a test at low water level, due to there being too little water in the control rod assemblies for free-fall buffer purposes.

The canned annular circulation pumps which deliver hydraulic operating pressure have given no trouble.

Instrumentation
Flow and level measurements are a source of difficulty in reactors. Hard-to-solve sealing problems arise if ordinary instruments are used. Venturi tubes with all-welded differential pressure gauges have been employed for flow measurements in the primary coolant circuits. The gauges are sensitive to gas bubbles in the impulse lines. Differential pressure gauges have also been used to record tank levels, temperature disturbances being added those caused by gas bubbles in this case. Gas bubbles cause trouble more particularly in newly filled systems.

A method of measuring flows and temperatures in reactor plants without the use of pipework and associated valves and tank entries is a very urgent requirement. Since heavy water has low electrical conductivity and furthermore is radioactive, several methods that would otherwise be applicable could not be used.

Pressure and temperature measurements have not caused any trouble. The use of surface temperature instruments is attractive in reactor applications.

Future improvements
When Ågesta was designed heavy water cost about SKr 1000/kg and was in short supply. This led to the adoption of troublesome designs in an effort to save heavy water. Now the price has dropped to a quarter and arrangements such as fillers at the bottom of the reactor and in valves to reduce the heavy water inventory would not be necessary. Furthermore, the smaller pipe sizes could be replaced by larger bores.
When the reactor was designed there was insufficient statistical and manufacturing background to justify the choice of a domed head. Nowadays it would be much simpler to carry out nuclear warm-up if the head was domed, and such a head is experimentally planned.

The present fuel element failure detection system could now be replaced by simpler methods using the chemical analysis of samples.

Operating trials have been carried out without the ion exchangers and the results are encouraging. In the future, purification circuits could be designed without coolers and the ion exchangers could be connected-in before and after operating periods.

The Ågesta process has been found to be easily controlled and moderately sluggish.

Installations can be designed from the start for TV inspection, allowing centralized control room instrumentation to be reduced without any ill effects. A great many of the instruments installed at Ågesta could be dispensed with or simplified.

**Conclusion**

Design performances have been realized in virtually every case. Disturbances so far in the primary systems have not been more frequent than in corresponding first-generation reactors in foreign countries. It does not seem as though any appreciable reductions in quality demands are feasible, and conventional engineering standards are out of the question.

The great number of sub-systems and components which must be integrated and the impossibility of continuously supervising components make the plant unduly sensitive to disturbances. Simpler systems with fewer and more rugged and well-tested components would seem to answer for a reactor plant offering greater operational availability.
4. SPECIAL OPERATIONAL PROCEDURES
Ingvar Holtz

Reactor warm-up and cool-down

The most comprehensive measures required in normal reactor operation and systems control are those connected with warm-up to 210°C (working temperature) and cool-down to ambient temperature. Up to October 1965 the reactor had been warmed up and cooled down a total of 29 times, six of these being during light water trials. Six of the warm-ups in 1965 were by the reactor's own heat. These nuclear warm-ups were part of a test program, which has yielded such favourable results that nuclear heating will be employed as routine procedure. Apart from minor troubles in maintaining pressure during the light water trials period, the warm-up and cool-down cycles have been carried through satisfactorily. The introduction of nuclear warm-up has involved minor modifications to the pipework systems.

Warm-up

The reactor and the high pressure systems must be heated to working temperature, approx. 210°C, before power run-up for steam production may be commenced. Furthermore, with the help of the pressure control system the internal pressure must be raised to 33 atg.

The maximum rate of temperature increase has been set at 20°C/h to avoid excessive thermal stresses in the reactor and other heavy components.

The procedure for reactor warm-up and cool-down could be studied under fully realistic conditions during the light water trials period. The thermodynamic differences between light and heavy water are so slight as to be insignificant in this context. (Fig. 13.)
The originally planned method for warming up the reactor and the high-pressure systems involved the use of a 4 MW electric start-up heater located in the turbine building. From this heater hot water is pumped in two parallel flows. One gives its heat to the reactor head and the other to the four steam generators. Heat is then transferred to the reactor from the steam generators by the heavy water, which is circulated by the main circulators running at half-speed. A total of 24 starts have been carried out in this manner so far, largely without any serious difficulties. A number of incidents and observations made during the light water trials period will be recounted in the following.

When the reactor was brought above a temperature of 100°C for the first time circulation was lost, despite a very moderate rate of temperature increase, due to insufficient pressure gradient in the primary coolant circuits. The underlying cause was as follows: During heating the level of water in the reactor rises due to thermal expansion and, providing the reactor contains sufficient water, reaches the underside of the head at a temperature not higher than 100°C. As warm-up continues, the water expands into the pipes connecting the reactor with the pressurizer vessel.

Not until the water enters these pipes can a layer of stationary water be formed and heated by steam in the pressure control system to form a thermal barrier against which the steam can build up pressure in the system. In the case mentioned above, it was found that the water level in the reactor had been about 100 mm too low right from the start of the operation. It could not, therefore, reach the underside of the head in time to form the required barrier and satisfy the increasingly important requirement for pressure gradient. During subsequent warm-ups, starting with the right water level, this problem has not recurred.

The inventory of heavy water at Ågesta has subsequently been increased and it has become possible to keep the reactor quite full even at low temperatures. Thus adequate pressure can be maintained in all operational situations.
The difficulties of maintaining pressure in the primary systems when the heavy water level is low were again felt during attempted reactor physics tests, to determine the lowest moderator level at which the reactor can be critical in the temperature range around 200°C. With the reactor at this temperature the level of the water was experimentally lowered to what proved to be a critical level in more senses than one. The test must be abandoned when the water started to boil, with circulation disturbances in the primary system as a consequence.

An investigation into the pressure and temperature conditions in the primary circuits, at the low level concerned, indicated that the heat input of about 500 kW needed to keep the reactor temperature around 200°C during the requisite test period must inevitably result in boiling in the upper part of the main steam generator tubes or cavitation in the pumps. In either case, circulation ceases.

Boiling inside the tubes in the steam generators occurred at low flow rates. When these were increased cavitation in the pumps occurred due to insufficient suction head, which in turn was a result of the low water level. In this insoluble dilemma, the plans for experimentally determining the critical heavy water level at 200°C must be abandoned.

There were also difficulties at first in controlling the pressure in the 4 MW electric heater used for warming up the reactor. As the load, and therefore the flow of cold water in the heater, increased a layer of hot water remained in the upper part of the boiler vessel and maintained a high pressure, despite the low water temperature in its lower parts. Since the electric power utilized by the heater is controlled by the internal pressure, it was impossible to obtain a satisfactory correlation between load and power. The distinct temperature layering also resulted in considerable local temperature differences in the vessel walls, as evidenced by heavy leaking around the access manhole, for example.
To ensure that the pressure would faithfully follow the water temperature a spray arrangement was installed in the top to break down the hot layer. The feed to the spray nozzles is drawn from the system after the pumps and the arrangement guarantees that the pressure and temperature in the heater will respond to the steam pressure. After these modifications the heater's pressure controlled power regulation worked largely satisfactorily.

During the first warm-ups some of the safety valves in the lower parts of the hot water system blew repeatedly when no justification existed. The tendency of the valves to open was due to the very slight difference — only a few atmospheres — between the system working pressure, at which they should be tightly closed, and the system design pressure, 35 atg, at which they must be fully open. Repeated but unsuccessful attempts were made to adjust the opening pressure of the safety valves within this narrow margin between working pressure and design pressure.

It was finally decided to lower the working pressure of the hot water heater from 27 to 23 atg, and also the pressure in the lower parts of the system by 4 atg. After this no more unmotivated blow-offs have occurred. However, the pressure reduction meant that the working temperature of the heater dropped from 230 to 222°C, which limited the rate of heating the reactor in the temperature range above 200°C.

The electric hot water heater has been used, as stated, to warm up the systems. The heat drawn from the heater is determined by a bypass arrangement. Experience has shown that 12-14°C/h is the highest rate at which the systems may be heated without trouble. If forced heating is attempted at the start of warm-up it can easily happen that more heat is drawn off than the heater is capable of generating, or that — despite the newly-added spray — the increase in power can not keep pace with the increased rate of heat consumption.

In either case the temperature of the water in the heater will fall, the conductivity of the water rises and causes a reduction in the power consumed by the heater, the water temperature drops still more, and so on. This sensitivity is most marked at the start of the warm-up
process. It is due to the great temperature difference obtained between the heater and the rest of the system, and which requires an abnormally high degree of fine adjustment of the three-way valve in the heater by-pass line. If the heater starts to shut itself down in this way, heat draw-off must cease immediately, i.e. the warm-up must be interrupted and the heater allowed to recover its working temperature, 222°C, before warm-up is resumed.

When the reactor and its associated systems are being heated in the temperature range between 100°C and 200°C heat losses to the environment reduce the rate of heating by 2-3/°h. At this stage the heater can comfortably supply 2.5-4 MW, so that a temperature rise rate of 12-14/°h can be maintained.

Once the reactor reaches 200°C or thereabouts the rate of temperature rise falls off due to the reduction in difference between the hot water heater (222°C) and the heated systems. As the heat transfer from the heater dwindles it becomes difficult to reach the ultimate temperature for the reactor, 210°C. But a useful addition to the heating capacity is obtained in the final stage in that the main circulating pumps are switched from half-speed to full-speed.

Nuclear warm-up

Warm-up by means of the electric heater has worked largely satisfactorily. Nonetheless it seemed only natural to investigate the ability of the reactor to provide its own heat for warm-up, in other words, employ nuclear heating. That the station was not initially designed for nuclear heating may be attributed to the safety philosophy accepted then and which is still reflected in the present rules for the operation of the power plant. It was thought unwise to start the reactor without first heating it by auxiliary means, until its reactivity and control characteristics in both hot and cold states had been established by physics tests and other observations. To a certain extent the fear that it would prove difficult to control the heating of the heavy flat reactor head probably played a part. Two years after the reactor had "gone critical" for the first time sufficient knowledge of its nuclear
characteristics and experience of the thermal processes had been
gained to make nuclear warm-up the logical next step in the common
extension of the nuclear physics, thermal engineering and chemistry
test programs.

Thermal contact between the systems involved in warm-up is good
since the flow rates and heat exchanger surfaces are generous.
Thus there are favourable conditions for good heat distribution
in the systems. Even temperature equalization can be expected in
all affected circuits, regardless of the location of the heat input.

During electric warm-up the flat reactor head and the secondary
circuit are heated by parallel flows from the hot water heater.
The temperature of the head then seldom differs by more than a
degree or so from that of the reactor. In nuclear warm-up the major
part of the heat required by the head must follow a devious route
from the primary system to the secondary system, thence to the
heater hot water circuit, and finally via another heat exchanger
to the head circuit and the head itself. If any appreciable flow
of heat is to be induced along this circuitous path a greater
temperature difference than normal must be permitted between the
reactor head flange and the reactor vessel flange. It was decided
that during the tests this temperature difference would not be
permitted to exceed 10°C.

In nuclear warm-up the reactor is run up to power under conditions
of temperature, pressure, flow and levels that in normal operation
would result in a scram. During this type of warm-up, therefore,
the reactor control systems must be adapted to tolerate certain
"abnormal" conditions. For safety it is extremely important that
this tolerance is not extended to subsequent power operation. For
this reason the changeover between these control situations has
been embodied in the key operated power switch. Until this key is
turned the control systems accept the exceptional conditions
referred to above but at the same time two of the ionization
chambers in the scram chain remain effective and will initiate
rapid shutdown if the output exceeds what is normal for warm-up.
Once the power key has been turned after the completion of the
warm-up phase the ordinary scram chain is reinstated and the output
limiting function of the ionization chambers abrogated.
Up to October 1965 the reactor had been warmed up by nuclear means six times. The first two warm-ups were carried out without any modifications to the pipework and with temporary arrangements in the control systems. The electric heater was run for self-heating and pressurization in the hot water circuit. The average rate of temperature rise was approximately 10°C/h. The maximum temperature difference between the reactor vessel and head flanges was 9.5°C.

During the summer shutdown in 1965 the hot water circuit was modified to improve considerably the thermal contact between the reactor and its heavy flat head. At the same time, the changes in the interlocking circuits were made a permanent feature of the control systems.

In the four nuclear warm-ups carried out during the autumn of 1965 the reactor and its ancillary high pressure systems were at times heated at rates up to 20°C/h, the greatest temperature difference between the vessel and head flanges being approximately 8°C, leaving a good margin to the maximum permissible difference of 10°C.

During these warm-ups the hot water heater was completely shut off and served only as a pressurizer vessel in the hot water circuit. Feed water preheating, previously performed in the electric heater prior to the start of the warm-up process, could now also be carried out during warm-up by means of steam bled from the main steam pipe.

One difficulty encountered in controlling the heat output has been the dependency of the ionization chamber channels on reactor temperature. A given ionization chamber reaction represents about three times as great thermal effect when the reactor is cold as it does when the reactor is near its normal operating temperature, around 200°C. As a result, output changes must be carried out with a good deal of care, the generated heat being viewed constantly with an eye on temperature developments in the heated systems.

The nuclear warm-ups carried out so far have formed part of a series stipulated by the licensing authority. The results show that the method can be employed with no sacrifice of safety, at
the same time as it means a certain simplification of warm-up procedures and provides a more favourable operating condition for a number of components. Nuclear warm-up will therefore become the standard practice at Ågesta, subject to the final approval of the authority.

If it should be necessary the pumps required during warm-up could very well be powered by the emergency generating equipment at the station. In other words, the station could start up entirely independently with no outside power supplies. In certain situations this ability could endow Ågesta with a significance as emergency power source quite out of proportion to its normally modest contribution to the Swedish grid.

Cool-down
In cooling down the reactor from operating temperature to about 30°C as usually required hitherto, a rate of temperature drop of 20°C/h has proved easily achievable, meaning that cool-down has taken 9 to 10 hours. The maximum permissible cooling down rate has been exploited only on a few occasions in conjunction with irradiation experiments in which it has been important to remove the irradiated object as quickly as possible from the reactor. As a rule cooling down has been at a rate of 10-16°C/h. A low rate is selected if cooling is carried out overnight with plenty of time in hand. But the rate is often determined by the need to terminate cooling down at a specific time in order to suit other scheduled measures.

It has proved possible to satisfy the requirement that the heavy flat head of the reactor should hold a temperature near to that of the reactor, even during cool-down. Generally speaking, the temperature difference between reactor vessel and head flanges has amounted to but one or two degrees during cool-down.

Vacuum drying of heavy water systems
During test run of the reactor light water was used. Before the systems could be filled with heavy water the light water must be completely drained, which was done by vacuum drying in two stages. Fig. 14 shows a pressure time diagram of this.
Operating instructions

During trials, commissioning and operation of the Ågesta reactor experience has been gained which has shown how the systems involved in the thermal process should best be handled for simple and trouble free operation. By and large it has been possible to employ the methods envisaged during designing of the various parts of the process. But in all instances certain additional steps have been introduced, which have made the operating procedures more extensive and time consuming than foreseen. As an example, setting and checking of the systems involved in reactor warm-up require about 7 hours before actual warm-up can be started. As practical experience has been acquired, the station operating instructions have been reappraised. At first they tended simply to increase as new items were added in the light of working experience. But as time has passed, many of the instructions have become so self-evident that it has been possible to simplify in certain respects.

Lessons learned at Ågesta tell us that no matter how well prepared and proven an operating instruction may be, it must never be followed blindly but applied subject to the discretion of the operating staff. The actual situations encountered are of such varied nature that the application of an instruction always requires awareness and good judgment on the part of the duty personnel, since the instruction is inevitably based on an "ideal" situation, unlikely to be realized in practice. Thus not even the most detailed instructions can fully replace an understanding of the process involved but must rather serve as a kind of check list, ensuring that no measures will be overlooked or taken at the wrong moment.
5. INSTRUMENTATION AND CONTROL

Ingemar Myrén

The individual components of the extensive reactor monitoring and control systems have behaved well but taken as a whole there are certain defects due to inadequate liaison between heat and instrumentation engineers in the design stage. The use of both valve and transistor amplifiers has shown that the latter require less maintenance.

The instrumentation and control installation comprises two main parts: process instrumentation for the heat engineering systems and nuclear instrumentation together with controls for reactor power and interlocking circuits. At the time when the installation was designed transistor techniques were not sufficiently advanced to permit complete transistorization. The installation therefore includes both valve and transistor amplifiers.

Process instrumentation

The process instrumentation is largely conventional, comprising transmitters for various parameters, signal converters with current take-off, limit sensing devices and direct reading or recording instruments. The factors distinguishing Ågesta from a purely conventional installation are the presence of radioactivity and the heavy water in the primary circuit.

Because of radiation hazards there is no access whatsoever to the primary circuit during operation. All important measuring circuits, which via limit sensing devices are connected to the reactor interlocking circuits, have therefore been trebled and interconnected so that at least two must indicate a transgression of the permitted limits before a fault signal is initiated. Pressure and differential pressure gauges are installed in accessible spaces and connected to the points of measurement by long tubing. All temperature probes are duplicated, with one as a stand-by, the switching-over facilities being contained in accessible enclosures. The only instruments that have proved impossible to render accessible are flow meters of buoyant force types.
Since all heavy water circuits and the related instrument components are of stainless steel and all-welded, any repairs of transmitters must be performed in situ in order to avoid cutting-out and rewelding. During design much care was given, therefore, to the selection of individual components.

Temperature measurement

Probes for resistance thermometers are of platinum with 100 Ohms resistance. For purely monitoring functions, especially in the various cooling circuits, these are connected directly to crossed coil instruments. Where active control is also required, the resistance probe is connected to an amplifier and subsequent limit sensing devices.

Only a few of the large number of transmitters— a hundred or so— have required replacement. The connecting enclosure should have been made somewhat larger, however, to permit more dependable connection between cable and transmitter. An unfortunate choice of ratio in certain of the crossed coil instruments has meant that these could show two quite different temperatures. After replacement of these instruments no further trouble has been experienced.

The signal converters include an integral adjustable resistance for compensation of line resistance. This item is very difficult to get at and, since it requires resetting when converters are replaced, it has been decided to separate the resistance from the converter and incorporate it in the permanent wiring.

The thermocouples employed are of Cromel-Alumel, in stainless steel enclosures with ceramic insulation. They are used primarily for the measurement of temperatures in the fuel elements and in the reactor head. These transmitters, of which there are about 200, have also behaved satisfactorily.

It may be noted here that certain comparative temperature measurements are required in conjunction with the control of the reactor head temperature. One of the temperatures compared
is measured by a resistance thermometer and the other by a thermocouple, an arrangement which has been found rather unsuitable since the differing non-linearity of the two types makes full proportional adjustment in different temperature ranges impossible.

Pressure measurement
Pressure is measured for the most part by Bourdon tube gauges with signal converters. The instruments have worked well, except in one case where the pipe to the tube in the instrument failed.

Flow measurement
Flow rates are converted to differentials by venturis or orifice plates. The design of the venturi tubes and their location in vertical pipes makes it very difficult to achieve complete degassing. The residual gas causes incorrect flow information, with the errors getting serious at low flow rates. The long small-bore tubes between transmitters and differential pressure gauges are also difficult to degas.

The bouyant-force meters employed are of full-flow type with magnetic transmission of float position to the electromagnetic signal converter system. On several occasions imperfectly designed end stops have resulted in sticking of the floats in several meters, especially at the upper end position. A further trouble is that, at its lower end position, the float acts practically like a non-return valve and impedes drainage. The mechanical device for sensing the position of the float has stuck on several occasions and patient re-adjustment has been required to achieve proper functioning.

Level measurement
Differential pressure gauges are used for level measurement. Since it is difficult to achieve the same temperature gradient in the reference column as in the tank contents the method is temperature sensitive. It can also be difficult to eliminate
apparent level differences due in fact, to pressure changes. Gas pockets in the communicating lines also affect readings. The instruments as such, however, have worked well.

Monitoring of fuel elements

There are two distinct systems for monitoring the fuel elements. In one the temperature of the water leaving the elements is measured, in the other the same water is tested for fission products. Part of the heated water flow is channelled off through the fuel assembly suspension hooks. After passing the thermocouples this water is returned to the suction side of one of the main circulators.

Temperature measurement

The thermocouples are connected to an oscilloscope via a rotating mercury switch. A discriminator is incorporated for alarm purposes. During one rotation the alarm signals are stored in a memory which turns at half the scanning rate. The memory is scanned continuously and, in conjunction with the direct discriminator output, actuates a coincidence circuit for alarm signalling. Thus a signal indicating excessive fuel temperature will be transmitted only if the thermocouple shows the excessive temperature during at least two consecutive scanning cycles.

The mercury switch, comprising a rotating nozzle and a ring of contact pins, has been one of the major problems in this installation. The mercury, which is impelled partly by a small pump and partly by centrifugal force, oxidizes very quickly and causes choking of the nozzle.

Operation could be depended on for not more than three days without special measures. This period has been increased to 40 days after the fitting of a somewhat larger nozzle and the supply of nitrogen to the switch. But despite the reduction in false alarm frequency from ten or so per day to perhaps two or three per annum, the system seems far too sensitive. In the light of present knowledge a less sensitive and slower system with a transistor switching mechanism would surely have been preferred.
Fuel element failure detection

For fuel element failure detection a small portion of the water is drawn off after the thermocouples, through solenoid valves to cooler, a reduction valve and a gas purger. From the latter the water is returned via the drainage tank to the reactor system. Gas contained in the water — normally nitrogen but in the event of a cladding failure also gaseous fission products — is driven out in the purger.

The released gases are forced together with purging nitrogen to a scanning unit consisting of a cylindrical precipitator vessel through which a powered steel tape passes. An electrostatic field between the tape and the vessel wall causes solid decay products (beta active) to be collected on the tape, which is scanned by a beta sensitive counter. The radioactivity thus detected is a measure of the extent of the damage.

The detection system comprises channels for bulk monitoring, a channel for localizing a failure to one of a number of groups of six fuel assemblies and a channel for individual samples. Besides five gas purgers and scanning units, the system contains about 200 solenoid valves. The valves’ measuring equipment and loggers are controlled by a program unit which is largely based on 30-stage stepping selectors.

For a high degree of measurement accuracy it is necessary for all flows in the heat processes to be constant and accurately established. In practice, however, wide variations are encountered due to clogging of reduction valves, thermal imbalance, pressure changes and so forth, and readings must therefore always be assessed in the light of the actual conditions. This places great demands on the recording of flows, temperatures, etc.

A good many problems have been encountered in conjunction with the fuel element failure detection system. Major adjustments have been necessary on the process side, such as to prevent condensation in the scanning units. Less serious adjustments have also been necessary in the programming arrangements to
ensure correct control and recording. This work has been of such an extensive nature, and taken so much time, that the system is still not available for regular use. Simpler methods are likely to be employed in future plants.

**Process control**

A hot water boiler and cooling circuits are provided for the warming-up and cooling-down of the reactor system, including the reactor head circuit. The system is adapted to heating or cooling as required by means of powered valves. Regulators and a powered reference source permit variation of the warm-up or cool-down rate between 4° and 20°C per hour.

Similar regulators are employed for the feedwater supply to the steam generators. In this case they are controlled by level in conjunction with steam and feedwater flows. Again, in the steam line a similar type of regulator is used to control the flow of steam to the dump condensers. During their short period of operation these regulators have functioned satisfactorily over the whole.

To prevent boiling, the reactor is associated with a pressure control system. Pressure is raised by means of a small electric steam generator with a maximum capacity of 400 kW. The water in this boiler is heated to boiling point by 85 electric elements connected in various groups. One group, representing 60 kW, is continuously variable by transductor. The other groups are cut-in or cut-out in fixed steps.

Because it is important that the pressure be maintained within close limits the control equipment has been required to meet stringent specifications. It relies on the difference between the actual pressure and an adjustable fixed reference level. Voltage proportional to the difference is fed into a circuit which gives a basic current, proportional to the pressure difference and its time derivatives. In this way the large time constant of the heating elements is compensated. The control unit has two outputs: one controlling the above mentioned...
transductor and the other a chopper-bar regulator which, in turn, cuts in or out the fixed power groups.

The first set of heating elements enjoyed a very short life. The control equipment requires that all elements be in working shape and makes allowance only in retrospect for lack of power due to malfunctioning elements. It was, therefore, difficult at first to achieve stable control. After the elements had been replaced by an improved version, the equipment has worked perfectly and has had no difficulty in maintaining the pressure to within ±0.2 at. In the later type of heating element the single coil has been replaced by three coils connected in parallel. The amount of insulation (aluminium oxide) is much reduced. The temperature of the coils has hereby been reduced to below 700°C.

Nuclear instrumentation and safety controls

Measuring equipment

The equipment through which the reactor power is monitored and controlled comprises three types of detectors with amplifiers. In the low-power range (up to 0.8 MW) BF₂ counters and fission detectors are used, for higher powers ion chambers are used.

Instrumentation of this kind, designed for pulse detection, is very sensitive to disturbances. In order to avoid these all cables associated with a particular measuring channel are contained in a common highly effective screening sheath. The cable harnesses thus formed are run in 5 mm thick steel pipes. In addition, all relay and contact coils are interference screened by voltage dependent resistors or RC circuits. As a result of these measures the instrumentation is quite unaffected by any of the existing disturbances.

The instrumentation is duplicated in the low-power range and trebled in the high-power range. The connections to the safety system are arranged so that, in the low-power range, a single instrument defect is sufficient to initiate a scram. In the higher powers, on the other hand, two simultaneous defects are
necessary to initiate scram. This provides considerably improved security against malfunctioning. The equipment, which depends entirely on valves, has worked excellently. All malfunctions encountered hitherto have been due to valve failures.

Control equipment

The control equipment utilizes an ion chamber to measure reactor effect and a linear DC amplifier with variable amplitude, giving the actual value. The control rods regulating the reactor power receive an operating impulse when the output voltage from the amplifier deviates from 10 V. The reference value is provided by a servo-controlled potentiometer incorporated in the DC amplifier reconnecting circuit. A change in the reference signal results in a change in amplification, which in turn causes the output signal from the amplifiers to deviate from 10V. Through impulses initiated by the resultant difference in voltage, representing the power deficit of excess, the control rods are moved to restore balance.

The reference signal can be adjusted manually or automatically. In automatic control the servo motor is actuated both directly by the difference between the temperature of the water leaving the reactor and 220°C (as represented by a reference voltage), and by the changes in the inlet and outlet temperatures at the reactor.

The hydraulically operated control rods receive operating impulses from contactor controlled DC solenoid valves. The contactors may be operated manually from the control desk, as well as by the control equipment described above.

Subsequent to minor adjustments intended to improve amplification in certain of the component amplifiers, the equipment has behaved perfectly. Under conditions such as relatively large changes of load the outlet temperature of the coolant is restored very quickly to 220 ± 1°C. Despite the arrangement of an RC circuit across the solenoid valve coils the contactor contacts to such valves were heavily burnt. No such damage was detectable, however, after the RC circuit was replaced by diodes.
The station hydraulics have behaved satisfactorily, but the position indicating mechanism for the control rods, on the other hand, has been less satisfactory. For the control rods this equipment comprises a long potentiometer. For scanning there is a crossed-coil instrument for each control rod in the control room, and in the control desk a servo powered counting mechanism which by means of a changeover switch can be connected to any control rod.

Problems have arisen due to poor contact in the potentiometer, either because of oxidation or because of sweating in the underlying insulation, and due to seizing of the servo, presumably because of overheating.

Ways of dealing with the potentiometer troubles are currently being sought. As regards the servo motor, the input power has been reduced somewhat and a hood will be taken down.

Interlocking circuits

The reactor safety system consists of a number of relay combinations reacting to signals from the nuclear and process instrumentation. The purpose is to prevent reactor start-up or stop the reactor immediately if any of the requirements for safe start or continued operation, as appropriate, is not fulfilled.

The basic philosophy is that operation is prevented as soon as certain readings transgress given limits. A defect in an item of equipment gives similar indications to a limit transgression.

In order to improve operational reliability, the majority of the measuring channels are trebled and connected so that at least two of the three must indicate a transgression before the interlock becomes operative. Experience of this system has been very encouraging. Thanks to the use of 2-of-3 intercouplings, no more than a couple of scrams have occurred during full power operation.
Conclusions

The extensive intermingling of valve and transistor equipped amplifiers puts severe demands on the maintenance staff but has yielded useful experience. It is now possible to state that transistorized equipment requires considerably less maintenance, particularly of the preventive kind.

The individual components of the monitoring system have functioned well. The various transmitters have transmitted their information properly to the metering equipment. Yet despite this, the total picture is far from pleasing. The reason is found in the failure to observe during the design stage the requirement that process engineers and instrument technicians should work in intimate cooperation. Furthermore, for a fully satisfactory result, it is necessary for them to have a keen appreciation of each other's work.

Problems which could have been entirely eliminated include wrongly located temperature probes, inadequately dimensioned lines to differential pressure and pressure meters, unnecessarily complicated control equipment, etc. It is also likely that the number of measuring points could have been reduced.
6. FUEL AND CONTROL ROD HANDLING

Lars Broström

The standpipes through which fuel and control rod assemblies are introduced into the reactor by the refuelling machine are sealed against the reactor pressure by the control rods themselves or by radiation shielding standpipe plugs, which are pressed against their seats by spring washers. A number of these have fractured. Certain difficulties have been encountered in hanging the fuel assemblies in the reactor.

On the underside of the reactor head there are 140 hooks for fuel assemblies and a further 8 positions, non-cooled, which are either empty or used for assemblies containing irradiation samples. The hooks are arranged in groups of four, each group being served by a standpipe, Fig. 15, through which the refuelling machine inserts or withdraws the fuel assemblies.

Of the 37 standpipes, 20 contain control rods — 18 coarse and 2 fine — and since the standpipes are connected with the inside of the reactor they must be sealed against the pressure in the reactor. This is done by means of the control rods, where fitted, or by radiation shielding standpipe plugs. Contact between the seal faces is maintained by a compressed spring washer assembly.

Fig. 15 Assembly job on the reactor vessel lid. The big vertical pipes are charge pipes for fuel elements and control rods. The smaller pipe to the left is an irradiating channel.
Fuel and control rod assemblies

Radiation from fresh fuel elements is very slight and they may be handled without any shielding. The storage for fresh fuel elements is, therefore, a simple installation of small compass.

Spent fuel, on the other hand, generates both radiation and decay heat in such amounts that it must be stored behind radiation shields and cooled continuously while in storage. The storage space for irradiated fuel consists of vertical tanks with standpipes, similar to the reactor standpipes, fitted to their tops. The refuelling machine is connected to these pipes. Each of the six tanks can hold 49 fuel assemblies.

Fuel assemblies with damaged cladding must be stored in a special facility, basically similar to the regular irradiated fuel storage tanks but containing flasks, one for each fuel assembly, to prevent the spreading of radioactive matter that might pass through the cladding.

Two more storage facilities are allocated for control rods and standpipe plugs. Here spare control rod assemblies and other assemblies on their way to or from the reactor are stored, and standpipe sealing gaskets are replaced prior to re-installation in the reactor.

Maintenance of control rod assemblies will be performed at Studsvik, where spent fuel will also be sent for long-term storage. Three different types of transport containers for irradiated materials have been acquired: for one control rod assembly, for one fuel assembly, and for seven fuel assemblies. These are stored under an indexable cover to which the refuelling machine can be connected. The charged transport flask is placed on a special vehicle for forwarding to Studsvik.

The heat generated in a fuel element depends on its position in the reactor and the flow of coolant must be adapted accordingly. The reactor core is divided into three zones and three different channel gags, inserted at the bottom end of the fuel assemblies, are used. It has been found economical to move the fuel assemblies
between different positions in the reactor, and a manipulator at a special station is employed for the accompanying changes of gags.

Refuelling machine

For moving the fuel and control rod assemblies and plugs between the reactor, storage positions and other points within the reactor containment, the refuelling machine has been placed on a gantry running parallel with the longitudinal axis of the reactor hall. See Fig. 16. A traverse carriage running on the gantry carries the radiation shield, which extends very close to the bottom of the refuelling trough, and inside this the storage vessel with its hooks for 4 fuel assemblies and 2 control rod assemblies or standpipe plugs. An indexable tube with grooves for a telescopic manipulator having a grab at its lower end, is fitted centrally in the vessel. The grab can grip the fitting which terminates the upper end of each fuel assembly and control rod, either directly from above or from a position to one side, whereupon a grab arm is extended and retracted to bring the assembly into the manipulator axis for lifting.
On the refuelling machine (Fig. 17) there are two control consoles, one for carriage operation and the other for the remaining functions. When the machine is to be used, it is first driven to approx. the right position as indicated by a synchro system and then finally aligned relative to a positioning mark visible in a periscope. The nose unit is then lowered and seals off the operating spaces from the environment. The refuelling machine bottom valve may then be opened and the recharging operation commenced.

When a removed fuel assembly is suspended from a hook in the handling vessel its lower end fits in a gas collector, and the assembly can be connected to a nitrogen circuit that incorporates a gas cooler, de-humidifier fans, heater, and control equipment. The gas cooler transfers fuel element decay heat to a water circuit which in turn sinks it in the containment atmosphere.

Before coupling to the reactor the refuelling machine must be warmed in order to avoid heavy water condensation on its cold surfaces. The heater serves this purpose, and can also be used in combination with the de-humidifier when it is necessary to remove moisture from the storage vessel and its contents - especially important if an assembly that is wet with light water is to be inserted in the reactor. The cooling water in the
de-humidifier coils is melted ice, the ice being constantly reformed by a refrigerating machine.

If a defect should occur in the water cooling circuit that passes decay heat from the fuel elements to the atmosphere, light water from a stand-by tank mounted on the refuelling machine can be circulated through the gas cooler. If this system should also fail, the fuel elements can be cooled by spraying the water over them, whether they are hung from the hooks or held by the manipulator. While the machine is coupled to the reactor, spray water must be drawn from a heavy water tank, also on the refuelling machine.

The refuelling machine controls are based on a system of interlocks to prevent the operator from making risky movements. Information is supplied by micro-switches with associated relays, except in the reactor and irradiated storage tanks, where a memory relay circuit is substituted. The information signals do not trigger action automatically, the operator has to initiate each step manually. A number of monitoring instruments have also been included in the console instrumentation.

It may be expected that there will be long intervals of idleness between periods of intensive refuelling activity. A reactor mock-up has therefore been provided to ensure that the personell maintain their familiarity with the machine and that it functions correctly.

**Fuelling operations**

The first fuel assemblies were inserted during the light water trials period and comprised the 36 most central positions in the core. These assemblies remained in place till the trials period was terminated, after which the reactor was completely emptied and the fuel elements transferred to empty storage spaces. During this operation it was found that the assemblies tended to catch on the lower ends of the stand pipes, due to these being inadequately chamfered. Fortunately this problem could be solved by replacing the rings fitted at the lower ends of the assemblies by a modified pattern.

Simultaneously with the hanging of a fuel assembly (Fig. 18) on its hook in the reactor (Fig. 19) its lower end is guided by a cone into a cylindrical receptacle (Fig. 20). Occasionally difficulties arose in the hooking process. The cause was unsuitable angles in the cones and at the bottom of the fuel assemblies, so that the latter tended to bite into the cones and stick. The fuel assemblies were therefore rounded slightly at the bottom edge but later experience shows that this is not wholly satisfactory either.

During reactor physics measurements with an exceptionally low water level
in the reactor a scram occurred with the result that the emergency shock absorbers in 11 of the control rod assemblies were crushed. This damage cannot occur when the water level is normal. By progressive switching the damaged rods could be removed from the reactor, disassembled and fitted with new shock absorbers. No other damage was observed. The irradiation of the control rods was at that time so slight that the work could be performed on site, without special radiation shielding precautions.

A number of the spring washers in the stand pipe seal assemblies have fractured, though not so as to cause leakage at any time. At first, damaged springs only were replaced but later the adjacent ones were also replaced, since the fractured component causes damage that will result in further fractures. The primary stand pipe seals were originally of silver wire, assembled by soldering. When the rings were stretched by the tightening of the seal, relatively great plastic distortion occurred in the region of fusion, resulting in leaks in some cases. New rings of cast type gave considerably better performance.

Molybdenum disulphide has been used successfully as a lubricant in screwed ss/ss connections. The lubricant effect has not proved adequate, however, to prevent tapered surfaces from scoring into each other.

Fig. 18 A new fuel element is lowered into the storage
Fig. 19
Underside of vessel lid with hooks for fuel elements (upper thermal shield of the reactor)

Fig. 20 Guide cones at vessel bottom
Preparatory for full power operation the reactor was charged with 4 fuel elements intended for burnout studies and 2 test assemblies with no fuel material for studies of irradiated cladding materials. The test assemblies are shorter and lighter than the regular fuel assemblies and, quite naturally, the control system was fed with information that was, in this special case, incorrect and certain operations were blocked. By interrupting the interlocks, however, the safety margin, which is considered necessary during fuel handling, is partly compromised.

After three month’s full power operation 6 fuel assemblies were temporarily removed from the reactor for measurement of their gamma activity. These tests were performed with special equipment at the channel gag changing station. The equipment was also found to be suitable for checking the straightness of the fuel assemblies, which in the case of these units was quite satisfactory.

Fueling operations are relatively time consuming. During completion of the full charge an average of 1.1 h was required for the insertion of a fuel assembly, and 1.2 h to replace one fuel assembly by another during the reactor physics tests. The first mentioned task required seven men, the other six.

Finally, emphasis should be placed on the importance of a routine requiring written orders for refuelling operations, written reports of the action taken and observations, and the careful logging of all fuel and control rod assemblies, etc.
7. CHEMISTRY

Erik Lindén

With the exception of the fuel cladding the primary system is made of or lined with stainless steel. The aim of water chemistry control is, therefore, to avoid stress corrosion and other types of local attack, reducing contamination by active corrosion products and avoiding the accumulation of these products, at places as in valves and narrow flow passages, to an extent dangerous to the correct functioning of plant equipment.

To assist herein all surfaces in contact with water have been given a high surface finish and a high standard of purity has been maintained in the light and heavy water used. The heavy water systems are closed circuits and water chemistry control is achieved by the use of ion exchangers in conjunction with chemical additives.

**Flushing**

After completion of plant erection the cleanliness of the surfaces in contact with the water was checked by rinsing the entire system twice with de-ionized water. During the first rinsing a wetting agent was employed. The two outlets at the low points in the system were fitted with filters. A flow through the system of $4 \text{ m}^3/\text{h}$ gave a water velocity of between 0.25 and 2 m/s in the various system circuits, perhaps a low value for effective rinsing in certain instances. The process was continued until a conductivity value of $0.5 \times 10^{-6} \text{ ohm cm}$ was obtained in the water. By that time only 2 g of sludge had been collected in the drainage filters.

**Degassing**

Due to the risk of corrosion in the carbon steel storage tanks the de-ionized heavy and light water used to fill the primary system was degassed by the use of hydrazine in carbon filters. Since the initial filling, the oxygen content of the water has been held at a low value. The quick removal of inleaking air during certain loading operations has been carried out using the water sprays in the pressure control system. This has been done at low temperatures by circulating nitrogen through the pressurizer vessel, and during heating operations by the venting of steam from the vessel.
Injection of gas and the formation of ammonia

During power operation deuterium and nitrogen are added to the coolant water to control the radiolytic processes

\[
2\text{D}_2\text{O} \leftrightarrow 2\text{D}_2 + \text{O}_2
\]

\[
\text{N}_2 + 3\text{D}_2 \leftrightarrow 2\text{ND}_3
\]

so that the oxygen content is limited and a certain ammonia alkalinity is obtained. The addition of deuterium prevents the oxidation of the ammonia to nitric acid. It has been shown that a deuterium concentration of 1 cm³/kg D₂O is sufficient to limit the acid formation to values below the limit of measurement.

It was intended to maintain a certain partial pressure of gas in the pressurizer vessel so that the gas concentration in the water could be held constant, and the synthesis of ammonia could be conveniently studied. However, the leakage of gas through the pressurizer safety valves to the blanket gas system has been greater than expected, the amount of gas in the water has decreased at a rate dependent on the spray water flow. The decrease rate has a half-life of one or two days, and gas is usually added to the water once per day.

Desirable alkalization has been obtained through the formation of ammonia. The equilibrium value of the ammonia concentration has been higher than reported for light water reactors. Based on measurements made during the spring of 1965, the following empirical relationship was found:

\[
(\text{ND}_3) = 0.1 (\text{N}_2)^{0.8} (\text{D}_2)
\]

where \((\text{ND}_3)\) is the concentration of ammonia in ppm and \((\text{N}_2)\) and \((\text{D}_2)\) are the gas concentrations in ml/kg D₂O.

Later operation during the autumn of 1965 indicated that the equilibrium concentration of ammonia was lower than given by the above equation, perhaps due to fuel element testing.
During nuclear warming-up operations with only nitrogen (no deuterium or oxygen) dissolved in the water, no measurable amounts of ammonia or nitric acid were detected.

Appendix 1 gives a record of the gas and ammonia concentrations measured during operations in 1964-1965.

Ion exchangers

Before charging into the equipment, the resins in the purification system ion exchangers were subjected to comprehensive chemical treatment for cleaning and to obtain the desired ion form, followed by the displacement of the light water by heavy water. The resins are made up of a mixture of 40% by volume of strongly acidic cation exchange resin in the D\(^+\) form, and 60% by volume of strongly basic anion exchange resin in the OD\(^-\) form.

The two exchangers which are mounted in parallel have a total resin volume of 500 litres, and 18% of the total heavy water quantity in the reactor passes these each hour. After a month of power operation during the spring of 1964 the exchangers were saturated with ammonia, and the concentration of ammonia in the plant water was increased thereafter up to the radiolytic equilibrium value.

After the period of light water operation the exchangers were found to hold a total of 330 g Fe\(_{\text{Ox}}\), roughly 3% of the total of 10 kg oxide expected from the primary system stainless steel surfaces.

The effectiveness of the exchangers has been investigated by a number of methods: the use of monitoring ion exchangers, water sampling before and after the operational exchangers and by operating for a number of weeks with the operational exchangers shut down.

The monitoring exchangers have a specific volume loading similar to the operational exchangers and are mounted in parallel with them.

The diagram in appendix 2 shows the distribution of iron and nickel and the activity of Mn\(^{54}\), Co\(^{58}\), Co\(^{60}\) and Fe\(^{59}\) axially in the exchangers after use from the commencement of nuclear operations until November 11, 1964. It will be noted that the major activity arises from Mn\(^{54}\).
Water samples taken before and after the operational exchangers have shown that Na$^{24}$ and F$^{-18}$ activity is removed to 98-100%. Results from the removal of corrosion products and fission products are not so conclusive.

The only measurable effect noted on water chemistry during the periods when the exchangers were not in operation (May 1964 and November 1965) was an increase in the amounts of Na$^{24}$ and fission products.

**Primary system sampling arrangements**

The primary system is monitored chemically by in-line instrumentation and by the analysis of samples. The analysis equipment is fed by sampling flows taken from upstream and downstream of the ion exchangers. A separate circuit is used for determining the amounts of undissolved impurities which can be filtered out and for exposing material specimens to hot reactor water. All sampling equipment is mounted in glove boxes.

The in-line instrumentation measures conductivity and oxygen content. The oxygen is estimated from the increase in conductivity obtained after passage through a column filled with thallium chips. Heavy water samples are analysed at Studsvik for D$_2$O content, tritium, ammonia, sodium, chlorides, fission products, corrosion products and gases.

The D$_2$O content of the heavy water, measured by an infra-red technique, has changed very little from 99.825 ± 0.010 % after filling in July 1963 to 99.805 ± 0.010 % in October 1965, which corresponds to the addition of about 15 litres of light water during that time.

Electrical conductivity measurements on the light water and on the heavy water during low power reactor operation indicated values below the limits of measurement (0.1 $10^{-6}$ - 0.2 $10^{-6}$/ohm cm) and during high power operation values between 2 and 12 $10^{-6}$/ohm cm, which corresponded to the ammonia alkalization. Values below 0.2 $10^{-6}$/ohm cm were again reached two months after reactor shutdown in June 1964. pD was only measured occasionally. Based on
the ammonia concentration during high power operation, a pH corresponding to a value of 9.2 - 9.8 at 25°C was obtained. Measurements made with glass and calomel electrodes, originally in light water form, gave a pH roughly 0.5 higher than the corresponding pH calculated for light water.

The amounts of chloride present during light water and heavy water operation have been between 0.05 and 0.20 mg/l.

$^{41}$A, arising from the irradiation of argon, accounted for the major portion of the activity in the samples withdrawn, giving $0.1 - 5 \times 10^7 \text{ diss/sec/kg D}_2\text{O}$. This argon originates partly from the nitrogen supply and partly from the protective gas used during welding operations, which has entered the blanket gas.

The $^{24}\text{Na}$ activity has conformed to the operation of the operational ion exchangers in a remarkable manner. It increased from about $0.13 \times 10^5 \text{ diss/sec/kg}$ to an equilibrium of about $2 \times 10^5 \text{ diss/sec/kg}$ while the exchanger resins were approaching ammonia saturation. Thereafter an increase to about $12 \times 10^5 \text{ diss/sec/kg}$ was noted after shutdown of the exchangers. The sodium concentration based on this activity is less than 0.01 mg/l.

Of fission products only $^{131}J$ could be traced during the spring of 1965 when this activity held constant at about $30 \text{ diss/sec/kg D}_2\text{O}$. Subsequently the fission product activity has increased after the discontinuation of ion exchanger operation. The activity can still be regarded as being very low.

The following table shows some typical activity analyses of water samples.

<table>
<thead>
<tr>
<th>Date</th>
<th>Exchanger resin</th>
<th>Diss/sec/kg D$_2$O</th>
<th>$^{18}F$</th>
<th>$^{41}A$</th>
<th>$^{24}Na$</th>
<th>$^{131}J$</th>
<th>Total corrosion products</th>
</tr>
</thead>
<tbody>
<tr>
<td>16.4.64</td>
<td>Not saturated with ammonia</td>
<td></td>
<td></td>
<td>$4.1 \times 10^5$</td>
<td>$0.13 \times 10^5$</td>
<td></td>
<td>718</td>
</tr>
<tr>
<td>28.4.64</td>
<td>Saturated with ammonia</td>
<td></td>
<td>$3.1 \times 10^5$</td>
<td>$1.8 \times 10^5$</td>
<td></td>
<td>600</td>
<td></td>
</tr>
<tr>
<td>26.5.64</td>
<td>Not in operation</td>
<td></td>
<td></td>
<td>$12. \times 10^5$</td>
<td></td>
<td>1170</td>
<td></td>
</tr>
<tr>
<td>13.5.65</td>
<td>Saturated with ammonia</td>
<td></td>
<td>$2.8 \times 10^5$</td>
<td>$20 \times 10^5$</td>
<td>$3.1 \times 10^5$</td>
<td>19</td>
<td>185</td>
</tr>
<tr>
<td>2.11.65</td>
<td>Not in operation</td>
<td></td>
<td>$3.6 \times 10^5$</td>
<td>$0.8 \times 10^5$</td>
<td>$3.6 \times 10^5$</td>
<td>617</td>
<td>1760</td>
</tr>
</tbody>
</table>
Corrosion, corrosion products and corrosion product activities

General corrosion has been monitored by checking the development of deuterium under high temperature conditions. This is possible since the primary system and the blanket gas system are closed circuits. A further requirement is that no deuterium gas is added, and that the ammonia concentration is low. These conditions have been obtained during short test periods. Deuterium production has on all occasions been in the region of 1 ml/s, which, when converted into the equivalent amount of the dominant magnetite Fe\(_2\)O\(_4\), gives a corrosion of 21 mg/dm\(^2\) month on the system surfaces.

In contrast to this relatively high rate of corrosion is the normal appearance of such corrosion layers as have been observed during maintenance work carried out during plant shutdown.

No local corrosion attack worth mentioning has occurred on the primary system surface but a certain amount has been discovered under the Metaflex seals and other packings and in the leakage drain system. This is in all probability due to a high chloride concentration in the packing materials.

One of the spray pumps was found to be damaged at a number of points. All damage has appeared in the pump cooling system which is connected to the primary system but isolated since the drain line has been closed off. Thus chloride has leaked out of the asbestos material and collected in the circuit.

The activity of the corrosion products has been determined by water sampling, in some cases by filtering or by allowing the water to pass through an analysis ion exchanger. Corrosion products scraped from component surfaces have also been investigated.

Filtering of samples through filters (0.45 µm) indicates that the water normally holds 10-50 mg filterable iron per kg D\(_2\)O. Use of the crud filter in the analysis system gave appreciably lower values than laboratory filtering of water samples.

Activities were determined by the use of a multi-channel analyzer, in certain cases after chemical separation.
Due to the fact that detached corrosion products are transported by the heavy water from the point of formation through the core before being deposited on the surfaces again, activity appears in the corrosion layers even in the outer portions of the heavy water systems.

Activation of the corrosion products has proceeded very rapidly. After some 150 hours the specific activities had reached values which have been more or less maintained since then. Typical values of specific activity are shown in the following table for various samples. The percentage of the individual nuclides are also given.

<table>
<thead>
<tr>
<th>Date</th>
<th>Sample</th>
<th>% total corrosion activity</th>
<th>diss/sec/mg</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Cr$^{51}$</td>
<td>Fe$^{59}$</td>
</tr>
<tr>
<td>20.3.64</td>
<td>In-line filter</td>
<td>39</td>
<td>25</td>
</tr>
<tr>
<td>2.11.65</td>
<td>Water sample</td>
<td>44</td>
<td>11</td>
</tr>
<tr>
<td>18.1 -</td>
<td>Ion exchange</td>
<td>3</td>
<td>3</td>
</tr>
<tr>
<td>9.11.64</td>
<td>resin</td>
<td>36</td>
<td>22</td>
</tr>
<tr>
<td>21.5.65</td>
<td>Pump impeller</td>
<td>39</td>
<td>25</td>
</tr>
</tbody>
</table>

The occurrence in such quantities of Co$^{58}$ and Mn$^{54}$, assumed to have been formed by interaction with fast neutrons, is surprising but in accordance with experience with other water reactors.

A minor build-up of the long-life Co$^{60}$ (half-life 5.2 years) has been observed in the filter samples.

To monitor contamination of the heavy water wetted surfaces, the dose rates are measured on the coolant system, piping and on the purification system. The largest external surface dose rate on any component two weeks after shutdown in June 1964 was 100 mR/h on the outer portion of the first pipe bend from the reactor in each of the coolant loops (marked 3 in Appendix 3). This dose rate has doubled on shutdown a year later.
Comparable activity occurred in the inlets to the steam generators, and in certain sections of the regenerative heat exchangers in the purification system. Other portions of the systems gave considerably smaller dose rates. For the first three months after shutdown the dose rates decayed with a half life of about 45 days. Since this time the decay has continued at a slower rate.

Blanket gas

Deuterium appears in the blanket gas due to leakage from the pressurizer vessel. On reaching a value of 2% D₂ in the gas, oxygen is added to the recombiner loop which is in parallel with the drying equipment. The heavy water formed from recombiner is here separated out, thus allowing the recovery in the form of D₂O of the major portion of the D₂ gas added.

The blanket gas is checked for deuterium and oxygen by in-line instrumentation. The heavy hydrogen concentration is measured using a thermal conductivity instrument (with the analyzer cells arranged in a wheatstone bridge) and the oxygen is measured in an annular-chamber instrument based on the paramagnetic properties of the gas. The only factor affecting the measurements has been the appearance of helium giving a background to the deuterium measurements due to its informal conductivity. The helium stems from the use of this gas in leakage tests during installation, and afterwards.

In summary it can be said that the coolant system water chemistry control has been largely self-maintaining with the exception of the need to add N₂ and D₂ to the water. The closed nature of the coolant and blanket gas systems has reduced by a very large degree the amount of plant supervision needed.

Reactor lid circuits

The loops, almost entirely of carbon steel, are equipped with a parallel mounted ion exchanger loop containing 50 litres of mixed resins designed for the same duty as in the coolant system. The ion exchangers have been in operation for most of the time.
Make-up water to the loop is taken from the feed water tank.
Lithium hydroxide, and later potassium hydroxide, was added to
give a pH of 10.0 - 10.5. The ion exchangers were saturated with
one or the other hydroxide to maintain the required concentration
in the water. Ammonia alkalisation is now used, however, due to
a fear that stress corrosion cracks in the inspection channels
through the vessel head were caused by precipitation of alkalines.
The exchanger resins had to be replaced at intervals of 2 to 3
months during the first year of operation due to blockage by
solid corrosion products and to increases in the chloride concentra
tions. Since then the frequency of replacement has been
much less. On removal, the resins have contained about 120 to
200 g of magnetite.

Despite the use of only de-ionized water during low power operation
for the make-up of leakage, the chloride concentrations rose, on
one occasion to over 0.5 mg/l. Although not immediately alarming,
it was indicative of the reduced effectiveness of the ion
exchangers. Due to this the anion fraction of the resin mixture
has been given a more thorough chemical treatment than previously
before charging to the vessels, and chlorides are no longer of
concern.

Secondary loops

The question of whether to use conventional pickling was discussed
prior to installation of the steam, feed water and condensate
piping. Due to a number of practical considerations it was finally
decided to install the equipment after being carefully cleaned
and to use shot-blasted piping with a minimum of field welding.
By-pass lines were installed over the steam generators and the
turbine condensor and feed water pumps were used to circulate de-
ionized water containing 150 mg/l hydrazine for nine days at 100°C.

After this treatment the system surfaces, where visible, were seen
to be covered with a thin layer of magnetite. Estimates put the
amount of magnetite sludge collected from drainage points and
from filters at about 1 kg, with an equal amount removed from
the systems up to the summer of 1964 by means of blow-down water
and filtering. Since then a few kilograms of magnetite have been collected from various points. From a heat transfer viewpoint the deposits on the stainless steel steam generator tubing were of negligible effect.

The secondary systems have been operated in a manner similar to conventional boiler practice since the start of high power operation, using high purity water with the addition of only 0.1 mg/l hydrazine. Alkalization with morpholine was used during trial operations but was later discontinued. The ammonia formed has been removed by steaming in the feed water tank.

Other systems

The closed cooling tower system is made of carbon steel with the exception of the heavy water components, and is filled with de-ionized water containing about 50 mg/l hydrazine. No corrosion worth mentioning has occurred despite the fact that it has been difficult on occasions to maintain the hydrazine concentration due to inleakage of air.

Serious corrosion has, however, been observed in one of the open cooling tower systems. Here a mixture of carbon steel, copper and stainless steel materials is used, and the coolant is softened mains water. The tubes in the condenser on the emergency cooling compressor were found to be plugged with rust on dis-assembly in July 1964. Trials with inhibitors are in progress.
8. RADIATION HAZARDS
Bertil Mandahl

The periods of operation described here have been entirely without drama from the radiation hazards aspect. Physics measurements were carried out during the preparatory low-power period. No levels of radiation requiring restrictive measures were observed in the vicinity of the reactor and the primary systems. The safety question arose only in conjunction with the removal of irradiated material and measurements on these but the tests were well prepared and accompanied by only very low or non-detectable personal radiation doses.

High-power operation led to limitation of access to the reactor and primary system areas. In the design stage the spaces within the station had been classified according to the anticipated radiation levels, and radiation shielding and system installations then designed and constructed with a view to the service and access requirements. The design of radiation shielding took into account possible development to twice the present power and the possibility of operation with a number of broken fuel elements. The possible growth of radioactive corrosion products in pipes and components was also taken into account.

Since the systems have not been developed for higher power and no fuel element cladding damage has occurred, it has not been necessary to implement all the scheduled access restrictions. Thus in some cases it has been possible to inspect primary system components with very little restriction.

Leakage in the primary system

Leakage from the primary system and resultant dispersal of radioactive matter in the atmosphere or secondary systems was extremely small during the periods of full-power operation. It has been traced by measurement of the radioactive hydrogen isotope tritium, formed through the neutron bombardment of heavy water. The lower measurable leakage boundary is determined by the sensitivity of the measuring apparatus, the tritium
content of the heavy water and the turnover rate of water in the secondary system. During the first full-power period the amount of tritium in the heavy water increased from approximately 0.6 to 90 Ci/m^3. At the end of the period the leakage detection boundary was about 1 ml/h to the reactor lid circuit, where the water turnover rate is relatively low. Measurements indicated 4-6 g/h. Leakage to the air was measured in collected airborne moisture and calculated to be about 15 g/h. The leakage in the steam generators was so small relative to the water turnover rate that the measurements must be regarded as unreliable. The total estimated heavy water leakage was approximately 25 g/h.

During the second full-power period in the spring of 1965 the tritium contents rose to some 185 Ci/m^3, giving more dependable test results. But there remains a number of doubtful factors — such as disturbances due to chemistry experiments and the relatively very high rate of water turnover in the secondary circuits through the steam generators, meaning that in that instance the leakage corresponding to the lower detection boundary must be assumed. The total heavy water leakage during this period was estimated at about 10 g/h. It should be noted that the leak to the reactor lid system had been selected during the interval between operating periods.

Personal doses

The troublefree operation of the primary system and the low rate of leakage have meant that personal exposure has been extremely slight during the periods concerned. It may be pointed out here that the heavy water contained very little long-life radioactivity other than tritium. Approximately 0.4 mCi/m^3 has been measured after ten days’ decay.

The only personal doses recorded during full-power operation were in conjunction with inspection and work in the main steam generator room, where radiation levels are relatively high. Work was done on conventionally arranged secondary systems including components requiring attention. Doses were kept low (less than 100 mrem per fortnight) by lowering reactor output and limiting
each man's number of exposures. This was a practical illustration of the undesirability of placing components needing servicing in spaces where high levels of radiation may occur.

Except for special maintenance tasks the maintenance periods have mainly involved servicing of the secondary systems. Parts of these are close to the reactor primary circuits, however. Dose rates of 100-200 mR/h, due to collected corrosion products, were recorded at pipe bends and pumps in the main circuits and near to the tube bundles and tube plates in the main steam generators.

The dose rates were no obstructions to work, since either the distance to the working places was sufficiently great or the jobs were quickly performed. Work on the primary system has mainly been limited to small-bore tube systems, e.g. at measuring points and in water chemistry monitoring systems. In those cases where minor leaks occurred despite closed valves, a freezing of the heavy water, to form an ice plug in the pipe, was adopted with success.

**Special maintenance**

Early in September 1964 monitoring of the liquid wastes revealed that heavy water leakage had occurred. A total of some 200 litres had escaped from two piston pumps (for the return of heavy water from the drain system). The repair of these pumps was the first major job involving heavy water in contact with air that had been undertaken at the station. After starting in normal clothing - when it was found that the fitters received low but nonetheless measurable personal tritium doses - the work was mostly continued in protective plastic clothing and face masks, with outside fresh air supply. These protective dresses were perfectly adequate to prevent further tritium effects. The total doses were very low, approximately 25 mR, i.e. less than is normally measurable in external radiation doses.

During the heating of the reactor circuits towards the end of September a relatively serious heavy water leakage to the reactor lid circuit was detected. As soon as temperature and pressure had
been reduced, leak tracing and repair work were started at once. In conjunction with leak tracing, which necessarily involved heavy water in contact with the air, temporary but relatively high levels of airborne tritium activity were recorded. However, no personal exposures worth mentioning occurred.

The actual repair work was carried out on top of the lid and within its system components. The tritium activity here was not measurable after the lid had been totally drained and the reactor systems arranged to prevent further leakage. Dose rates for gamma radiation at the places of work amounted to 2-15 mr/h, mostly from the reactor through the fuel charging standpipes and from corrosion products that had collected in the small-bore tubes of the fuel element failure detection system.

The major part of the work was carried out by personnel from Degerfors Ironworks, makers of the reactor lid. The maximum permissible dose for contractor personnel is 1.5 mr/year and not more than 0.9 mr/3 months. Corresponding permissible doses for radiological workers are 5 and 3 r. In this instance it was possible to conform with the stipulations for contractor personnel, but if similar work should be required after a longer period of operation and perhaps shorter decay periods for fuel elements and corrosion products than the present 5 months, then it might be found that the number of workers required would be unreasonably high.

The external doses recorded during the work amounted to:

<table>
<thead>
<tr>
<th>Doses, mr</th>
<th>No. of persons</th>
</tr>
</thead>
<tbody>
<tr>
<td>100 - 200</td>
<td>14</td>
</tr>
<tr>
<td>200 - 400</td>
<td>3</td>
</tr>
<tr>
<td>400 - 600</td>
<td>6</td>
</tr>
<tr>
<td>600 - 800</td>
<td>1</td>
</tr>
</tbody>
</table>

The doses were mainly suffered by Degerfors and Atomic Energy Company fitters. No internal doses due to tritium or any other isotopes were recorded.
In February 1965 heavy water leakage occurred from one of the main circulating pumps. Most of the leakage, about 70 litres, was recovered for purification and reprocessing, losses being limited to evaporation quantities. The reactor was cold when the leak occurred so that the amount of airborne tritium activity in the premises concerned was only slight, and recovery and clean-up could be carried out without any intake of tritium.

After investigation it was decided that first this pump and later two other main circulating pumps should be taken out of service for rewinding and better sealing. In the periods between the removal and reinstallation of the pumps the reactor would be operated at high power.

When removing the first pump full suits with external fresh air supply were worn in view of possible tritium hazards. But the excellent drainability of the system resulted in very little heavy water spill and, therefore, but slight airborne tritium activity (less than $5 \times 10^{-6} \mu\text{Ci/ml}$) which in conjunction with the short working time meant that the cumbersome suits could be dispensed with. No noteworthy tritium exposures occurred despite the doubling during the interim operating periods of the heavy water tritium content, from 90 to 185 Ci/m³, approximately.

External radiation risks were moderate during both the first mentioned work and the subsequent disassembly of the pumps. Radiation was due to corrosion products, mainly those attaching to pump impellers and diffusers. The first pump had been shut down for 8 months when it was disassembled, and it might have been thought that the dose rates would be much greater when the other pumps were taken out. In fact, the increase was only moderate, from approximately 80 to 180 mR/h, measured in the immediate vicinity of the impeller.

**Effluents**

The first two leakages, mentioned under Special Maintenance, resulted in the passing to the station waste treatment plant of some 26 Ci tritium in a total of 70 m³ water, since in both
cases the heavy water had become so diluted that reprocessing was not economical. The then-applicable Water Rights Court decision allowed an annual release of 24 Ci tritium to Lake Magelungen, with the proviso that the rate must not exceed 0.01 Ci/h diluted in 20 m³ water, though in the present plant 15 m³ of this is taken from the lake.

Up to the end of 1964 some 13 Ci had been discharged from the plant. The rest was still in store at the station. There is no difficulty in storing water quantities of this size, and thus controlling the outlet in accordance with requirements, since the station is provided with a 330 m³ cistern.

A judgment handed down by the Water Rights Court on February 22, 1965, sets new regulations for tritium release, taking into account the low radiotoxicity of this beta emitter compared with many others. The station is permitted to release 2 Ci "other beta emitters" per annum, J₁³¹ being the most critical of these. The present judgment includes tritium among "other beta emitters" but its activity expressed in Curies is to be divided by 500 before addition to the other sources, subject to a maximum release of tritium to the lake of 0.3 Ci/h. This modification will, naturally, have great significance in future operation, since the tritium activity in the primary system will increase continually.

In order to simplify the operation of the waste treatment plant when large volumes of water are drawn off from the recooler tower the new regulations were applied at the end of April 1965, when the stored tritium activity was pumped out. It has not been possible to record the presence in waste water of any of the other radioactive matter mentioned in the judgment, i.e. beta emitters other than tritium, Ca + Sr, and alpha emitters.

It has not been possible to show that any airborne activity had been released except tritium. At the end of the full-power period in 1965 it was estimated that tritium released through the ventilation uptake amounted to less than 0.03 Ci/24 h. The estimated permissible annual amount that may be released this way is 8 MCi, a figure which cannot be reached even theoretically by the Ägesta reactor.
9. ELECTRICAL INSTALLATIONS
   Ingvar Wetterholm

The Ågesta Power Station has been provided with a comprehensive
stand-by power system to ensure reactor functions in the event of
a failure of normal power supplies. However, its complexity makes
this system relatively sensitive and difficult to maintain. The
adoption of the quiescent current principle for reactor monitoring
equipment renders in-operation service difficult. Trials are made
every month to keep stand-by power equipment and the staff at
full efficiency.

Some 10-12 MW of the reactor's total thermal output can be
converted to electric power in the turbo-generator. The electric
power installation has been designed with a view to possible
extension of the reactor plant at a later date, space being
allowed for additional equipment. The turbo-generator was trans-
ferred from an existing power station and adapted to suit the
steam data at Ågesta.

Arrangement and working

The main difference between the electrical installations at Ågesta
and those in a conventional fossil fuel power plant lies in the
stringent demands for operational reliability. These are reflected
most clearly in the extensive stand-by power equipment, Fig. 21.

Generator

The generator, rated for 15 MVA at 6.3 kV and cos $\phi = 0.8$, is
connected to a bus bar from which power is fed to the Stockholm
Electricity Board's 30 kV system via a transformer and to the
local station system at 3 kV and 400 V, similarly via transformers.
When the generator is shut down, power for local needs is obtained
from the 30 kV system, or to a certain extent from a 10 kV line.
Fig. 21 General diagram of electric plant.

Switchboards for 30 kV and 6 kV

The 30 kV and 6 kV switchboards are located adjacent to the turbine hall, outside the underground containment. The 6 kV board is actually designed for 10 kV, since machine voltage may exceed 6 kV after future modifications. Connected to the 6 kV bus bar are the main generator, a 25 MVA 6/30 kV transformer, and two 5 MVA 6/3 kV transformers for station services. Rating of the 25 MVA transformer takes future extensions into account. The transformers are located in bays outside the turbine hall.
Switchboards for 3 kV and 400 V

The 3 kV transformer is located in the so-called link building, which is underground. All power users rated at more than 100 kW are connected to the 3 kV system. Normally the bus bar is divided into a turbine and a reactor section for the sake of a low-resistance earthed 4 MVA hot water boiler which receives power from the reactor section. This section also supplies power to three 400 kW pump motors for district heating circulation, two 130 kW feed water pumps, etc. The switchboard comprises 40 sheet metal enclosed standard cells equipped with air breakers under remote control from the control room.

The 400 V switchboard comprises 24 single and 11 double cells of steel enclosed type. Like the 3 kV board it is divided into turbine and reactor sections, each supplied with power from air insulated 1500 kVA transformers connected to the corresponding sections of the 3 kV board. A third transformer is provided as a stand-by. Emergency power from the 10 kV line is available via another 1500 kVA transformer.

Reactor section

The reactor section may be regarded as comprising three systems: one not subject to any special reliability requirements, one with more stringent requirements, protected by four all-automatic 145 kVA diesel generator sets, and a third for which absolute dependability is necessary, protected by its own 400 V batteries across a 150 kVA motor generator.

Sectioning between the no-break and the diesel-protected systems is achieved automatically through two minimum voltage relays in combination with time delay relays. A frequency relay also causes sectioning at frequencies below 49.5 Hz.

The DC side of the motor generator features a transductor governor for voltage or frequency regulation. Under normal conditions the AC generator operates as a motor and the DC motor as a generator, thus charging the battery. When sectioning occurs the AC side switches automatically to generator operation, then
being driven by the now frequency governed DC motor. The battery is designed for 150 kW during 30 minutes. The charging state is monitored by a special amp-hour meter. An additional 20 kVA generator on the motor generator shaft supplies current to vital control equipment.

Sectioning of the diesel protected system from the non-priority system occurs automatically on an impulse from the minimum-voltage relays in combination with time-delay relays. Auxiliary contacts on the sectioning switch scram the reactor and transmit starting signals to the four diesel generators, which are automatically cut-in and run in parallel. The plant has been designed to allow for full coverage by three of the diesel sets. The sets synchronize in and assume load in 15 to 30 seconds.

A certain portion of the load on the diesel protected system remains switched on during the black-out interval and will, therefore, automatically start again as soon as the diesels supply power. Other power users must be restarted manually. When the diesel protected system is at normal voltage and frequency the no-break system is automatically synchronized and the battery can be recharged.

**Main circulators and motor generators**

The four pumps for the reactor primary coolant loops are driven by specially designed squirrel cage motors with rotors and stators surrounded by heavy water, which also acts as lubricant in the bearings. Normally the pumps receive current from the 3 kV switchboard across individual air insulated transformers. The correct functioning of these pumps is vital to reactor security, since its operating characteristics are dependent on the maintenance of circulation in the primary coolant loops under all conditions.

To ensure the power supply to the pumps in the event of a black-out each pump has been provided with a motor generator set. This and the pump motor are disconnected from the transformer supply when normal power is disturbed and the motor generator
immediately assumes the supply of current to the pump motor for half-speed running. The prime source for each generator is a lead accumulator rated at 114 Ah for 30 minutes. If stand-by operation continues for a longer period the motor generators are driven by the shaft-coupled induction motor with power from the diesel protected 400 V switchboard.

Operating experience

Stand-by power equipment

In order to check the proper working of the local power supply arrangements, a number of series of overall functional trials has been carried out, in conjunction with simulated black-outs. These trials have shown that the protected electrical systems enjoy an acceptable degree of security, have given the operating staff a certain familiarity with abnormal conditions, and have demonstrated the sequence of events in the various systems under a variety of plant conditions.

During one series, for example, comprising twelve cut-out tests, a total of four diesel start failures was recorded, though on these occasions the station was specially manned.

The performance on unscheduled power failures has not been quite satisfactory. The main reason for this lies in the complexity of the equipment. In order to keep it and the staff at the peak of efficiency, monthly tests are made. A check list of valves, changeover switches, setting devices, etc, has been prepared and is regularly reviewed. Load tests of the batteries, motor generators and diesels show that there are good safety margins at the present levels of loading.

The stand-by power requirements have increased successively during the progress of the project and it has been necessary to install four diesel generator sets, with more auxiliary equipment and complicated starting and synchronizing arrangements as a consequence. The location of the power station, with no normal cooling water supply, also involves auxiliary equipment which must be diesel protected.
From operational and maintenance viewpoints, a single diesel capable of covering the power requirement is clearly to be preferred, in which case the auxiliary equipment could be simplified and quicker load assumption achieved without any problems of synchronization and parallel working. A plant of this kind must also have a stand-by diesel with its own auxiliaries, this being placed in a separate room for fire and servicing reasons. It would preferably also be started automatically at the moment of power failure, ready for emergency use, but would normally be synchronized-in manually when power requirements dictated.

At Ågesta the control equipment for the diesel generators is placed in the diesel room together with the systems for protected and no-break loads. A panel in the control room carries instruments, order acknowledgement signals and certain cut-out switches. For operational reasons it would be better if the stand-by generating plant were controlled from the control room, so that the duty staff could take the appropriate steps immediately after a power disturbance.

With the present arrangement, qualified staff must be detailed to the diesel room where, because of the high noise level, they have difficulties in communicating with the control room. Other arguments in favour of control from the control room include fire risks, high temperatures and some oil mist in the diesel room, simplified training, more dependable supervision in quiet surroundings, and a saving of staff in critical situations.

It is relevant to note that reactor operators, for example, must monitor some 80 functions and evaluate some 60 signals in the event of a black-out and given correct functioning of the stand-by power equipment.

The design of the no-break and diesel protected systems, featuring common drive motors for five other shaft coupled machines, has greatly complicated these systems. The supply of power to the protected components in the station in the event of black-out is dependent not only on the correct manual setting as per check list but also on the correct behaviour of highly sophisticated automatic control equipment.
The majority of the reactor controls feature two-of-three interconnections (quiescent current) to avoid disruption of operations by isolated component defects. The stand-by power equipment consists of entirely conventional components and circuits arranged in chains, in which isolated component defects can have far-reaching consequences. A fault in a quiescent current connected circuit for a generator motor, for example, could in the event of a black-out deprive both the no-break and diesel protected systems of power, as well as the emergency lighting. Battery powered floodlights have, therefore, been installed in the control room, diesel room and other vital locations.

The frequency of external power disturbances is about five per annum but when producing electricity normally the station should be fairly unaffected by external disturbances. Internal troubles, in the stand-by power system itself, have hitherto consisted of a bearing defect, a slip ring defect, corrosion seizing in cooling water pumps, dirt choking in air intakes, and a few component defects.

The generator specification for the protected system supplying electronic, nucleonic and control equipment, etc, did not include any requirement as regards low harmonic content, since the importance of this only became apparent during the operation of certain regulators and control rod indicators. In order to eliminate harmonic disturbances it has been necessary to install a filter, which also eliminates harmonics in the stand-by power supply from the 30 kV system with its connected converter stations. Switching between the ordinary generator and the stand-by transformer has been modified to eliminate breaks but a few degrees difference in phase can result in wild signals in the reactor monitoring equipment.

Since the reactor instrumentation includes sensitive AC transmitters, amplifiers, control devices, limit sensors for quiescent current dependent contact functions in safety circuits, signalling equipment, etc, it will be appreciated that the required standard of power supply exceeds the capabilities of
the present protected system. The presence of quiescent current dependent circuits for both AC and DC introduces further complications. For example, it is virtually impossible during operation to trace earthing faults by the progressive elimination of loads.

In the light of experience at Ågesta, and since it must be regarded as unthinkable to depart from quiescent current circuits in the safety arrangements for reactor plants, simpler and mutually independent supply systems will have to be used in larger nuclear power stations where maximum dependability is required. The transistorization of reactor instrumentation components, together with central battery supply sources, ought to contribute considerably towards greater dependability under power failure conditions.

**Main circulators and motor generators**

During a reactor warm-up difficulties were encountered when starting the main circulators at half speed, whereupon one of the pumps suffered a winding failure. The pump motors have aluminium windings, a survival from the stage when the reactor was planned to have aluminium clad fuel elements. Unfortunately the conductor material was left unchanged when Zircaloy cladding was adopted, the aluminium windings causing not only manufacturing difficulties but also potential problems in the event of winding damage.

The damaged pump motor had a two-part winding, a result of the connecting leads having been too short at the time of manufacture. The motor was returned to the manufacturer and has been rewound with copper.

During the same warm-up process leaks were observed at the cable penetrations on two other main circulators. As a result, all four pumps have been equipped with an improved type of penetration and the windings have been better supported. The primary blame for the starting troubles with these pumps may be attributed to the increase in inertial friction at high temperature, and the
starting equipment for half-speed operation has, therefore, had to be modified.

The requirement for half-speed starting arose during the design stage, due to cavitation in the pump when operating at full speed at pressures less than 25 atm. Previously pumps were started by magnetizing the synchronous motor on switching-on and allowing current to be induced slowly. This equipment has been modified for immediate half-speed starting, the synchronous machine having to be fully excited before the pump is coupled.

In order to permit prolonged running at half-speed the batteries supplying the motor generators have been equipped with Avostat regulated converters for continuous separate excitation of the synchronous machines. During trials it was also found that the brushes on the tachometer generator of the motor generators gave trouble, resulting in one instance in the stopping of a pump during half-speed operation. An improved brush rocker has been fitted.

Pump starting troubles were also caused indirectly by sticking valves, a result of excessively rigorous requirements regarding valve securing arrangements. Subsequent to the easing of these requirements and the improvement of the contact devices the pumps have operated very satisfactorily. Switching of the pumps to operation on stand-by power has also worked well after the running-in stage.

The remarks regarding the location of the stand-by power controls also apply to a certain extent to the pump starters. If they were sited in the control room certain automatic control features such as automatic restart after a power black-out could be eliminated.

Generating plant

Remarks on generator operation are based on only a few weeks' running. Bearing oil and stator winding temperatures have maintained low levels and the set is almost entirely free from vibration. Load shedding tests have been carried out with th
generator at 2, 5 and 8 MW and 0 MVA, and confirm the favourable self-regulating characteristics of the reactor. The grid load has been disconnected in tests at low generator outputs without causing reactor scram. The local power requirement when the generator is running is about 1.5 MW.

There is nothing of importance to report as regards the remainder of the generating plant. A few principles may be enunciated:

- Such comprehensive control panels as those at Ågesta require a special degree of easy surveillance and exactitude.

- From the operating viewpoint, ammeters would be preferable to the acknowledgement switches in the electrical installation schematic presentation, for the controls of the 3 kV motors which are placed in ancillary systems.

- A great many unwarranted signals are given since here (as in the rest of the reactor plant) there is no blocking when the component concerned is non-operative, etc.
10. REACTOR PHYSICS EXPERIMENTS

Göran Apelqvist and Pehr Blomberg

Reactor physics experiments carried out to date at Ågesta show that its excess reactivity when charged with fresh fuel is approximately 2.5% greater than estimated. This means that a higher degree of burnup is attainable and it is, therefore, considered that it should prove possible to run the reactor at full power for more than 15 years, with the two existing fuel charges.

As the first nuclear power station erected in Sweden, the Ågesta reactor is of special significance to Swedish reactor physicists. Design calculations for this and subsequent reactors such as Marviken, have of necessity hitherto been based mainly on experiments in small scale rigs such as the RO and Zebra at Studsvik. Extrapolation to the circumstances obtained in a fullsize power station, with its much larger dimensions, higher power density and changes dependent on burnup, has been a major step. Physics experiments in the Ågesta reactor will contribute to a much better assessment of design calculations for its successor reactors.

The aims of these experiments have not, however, been confined to the supply of material for comparison with theoretical calculations. The introduction of any complicated piece of machinery requires the verification of its operating characteristics, and the severe safety requirements stipulated for nuclear power stations further accentuate the need for a comprehensive program of studies to determine the operating and security characteristics of this reactor.

Preparations for the reactor physics experiments at Ågesta were started as early as two years before the reactor went critical for the first time. Both planning and practical work have been carried out by a group of physicists and other specialists from the Atomic Energy Company, ASEA and the State Power Board.
Neutron economy

Generally speaking, the purpose of reactor physics experiments is to determine certain integrals, such as buckling and reactivity coefficients, and differentials such as the distribution of neutron flux and power within both individual fuel elements and the reactor as a whole, resonance capture of neutrons, neutron energy distribution and so on. The concept of neutron economy, which is intimately related to these integral and differential phenomena, concerns the complicated interplay between neutrons and materials in neutron leakage, resonance capture in fertile materials, parasitic capture in structural materials, coolant, moderator, fission products, etc, and capture — leading to fission — in fissile matter.

The results of the integral experiments give the balance of a reactivity budget but tell very little about the ingoing items. It is, therefore, fairly clear that experiments of both kinds must be carried out to obtain a true picture of the reactor physics sequences and to provide a complete check on the ability of calculation methods to reflect physical realities.

Definitions

Reactivity is expressed as the difference between the numbers of neutrons produced and consumed in a given time, divided by the number produced in that time.

In a critical reactor the neutron density and power are constants. The numbers of neutrons produced and consumed per unit of time are identical. When the reactivity is positive or negative the reactor is super- or subcritical, respectively, and the power is an increasing or diminishing factor. Two types of excess reactivity will be considered: one which may be regarded purely as a comparative coefficient for a postulated reactor (high values of excess reactivity), and one which corresponds to a measurable change-of-power rate (power doubling time or period).

The reactivity coefficient is an expression of the change in reactivity divided by the change in temperature, control rod position, moderator level, etc.
Table 1  Measured and estimated critical moderator levels in the Ågesta reactor, with varying numbers of fuel elements in the core

<table>
<thead>
<tr>
<th>Number of fuel elements</th>
<th>Critical moderator level, cm</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Measured</td>
</tr>
<tr>
<td>32</td>
<td>227.4</td>
</tr>
<tr>
<td>68</td>
<td>178.4</td>
</tr>
<tr>
<td>136</td>
<td>159.6</td>
</tr>
</tbody>
</table>

Critical size usually refers to the minimum size of the reactor (here the lowest moderator level for a given number of fuel elements) at which the multiplication process is self-sufficient in neutrons, i.e. at which the fission process is a self-maintaining chain reaction.

Material buckling is a measure of the deflection of neutron flux distribution in the central parts of the core in a critical reactor.

**Buckling and critical size**

Material buckling is a lattice characteristic, i.e. it applies for a lattice consisting of an infinite number of infinitely long fuel elements. For an exactly critical reactor without reflector or control rods it is possible to obtain a simple relationship between buckling and the geometric dimensions of the reactor.

At Ågesta this is not the case, since the fuel there is always surrounded by an annular column of heavy water, the thickness of which varies with the number of fuel elements. The results of measurements of critical moderator level with varying numbers of fuel elements yield knowledge of both the lattice characteristics and the influence of the reflectors on neutron economy. By measuring buckling — a property which is independent of the reflectors — at the same time it is possible to obtain results which permit the separation of reflector effect from the neutron budget balance.

The critical size (of the critical moderator level) has been determined at a moderator temperature of 20°C with 32, 68 and
136 fuel elements in the core. In these experiments the moderator level has been progressively raised step by step from a subcritical level, while simultaneously recording the neutron flux (by means of neutron detectors inserted into the core), until it has been possible to extrapolate the zero value for inverse neutron flux, which exactly expresses the critical level. The parts of the fuel elements above the moderator surface do not participate in the multiplication process due to the absence of moderator.

Table 1 indicates that agreement between theory and experimental results is weakest when the core is largest, despite the fact that the radial reflector is then thinnest. The axial reflector, on the other hand, is relatively speaking thickest when the critical level is low, i.e. when 136 fuel elements are involved. The difference in level is not easy to explain but is probably largely due to the theoretical treatment of reflectors and fuel junctions.

The material buckling of the lattice has been measured for the three sizes of core mentioned here and a moderator temperature of 20°C and for a core with 136 fuel elements and a moderator temperature of 215°C. From the earlier remarks it will be deduced that identical results might be expected in the first three cases, since the parameter concerned is independent of reflectors and reactor configuration. For practical reasons, the experiment at elevated moderator temperature was carried out with a bank of control rods partly inserted in the core, meaning that measuring accuracy was poorer than if the core had been "pure".
Table 2 Material buckling based on experiments with three sizes of core and at two moderator temperatures

<table>
<thead>
<tr>
<th>Number of fuel elements</th>
<th>Ågesta m(^{-2})</th>
<th>Material buckling Other experiments</th>
<th>Computed m(^{-2})</th>
</tr>
</thead>
<tbody>
<tr>
<td>At 20°C:</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>32</td>
<td>5.17 ± 0.25</td>
<td>-</td>
<td>5.07</td>
</tr>
<tr>
<td>68</td>
<td>4.99 ± 0.04</td>
<td>-</td>
<td>5.07</td>
</tr>
<tr>
<td>136</td>
<td>5.07 ± 0.04</td>
<td>5.06 ± 0.06 a)</td>
<td>5.07</td>
</tr>
<tr>
<td>At 215°C:</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>136</td>
<td>3.41 ± 0.14</td>
<td>3.50 ± 0.17 b)</td>
<td>3.07</td>
</tr>
</tbody>
</table>

a) measured at RO  b) measured at Savannah River Laboratory, USA

The material buckling is measured by fixing copper wires of about the same length as the fuel elements to the outside of ten guide pipes for irradiation in ten different radial positions. The basic tones of the variations in the induced activity along the copper wires and between wires on different fuel elements are adapted to a cosine function in the axial sense and a first order Bessel’s function in the radial sense to obtain the buckling. As shown by Table 2, the variations between the buckling results obtained from the analysis of copper wires irradiated in cores containing 32, 68 and 136 fuel elements at 20°C are fairly small. (It should be observed that the results obtained in the 32 elements core are subject to wide error limits).

The noticeable difference between the measured and calculated critical levels with 136 fuel elements and between the measured and computed buckling at a moderator temperature of 215°C lead to a higher "measured" excess reactivity than that calculated.

Moderator temperature

Knowledge of the moderator temperature coefficient plays an important role in the assessment of the reactor’s dynamical properties. Although it was not thought that the measurements of this at Ågesta would be likely to give results that would materially affect operation or the control system of the power
station, it is most important to obtain results of measurements throughout a wide temperature range, since there is a complete lack of such data for heavy water moderated reactors. In this respect, Ågesta offers facilities which are not available in pure research reactors.

Simply expressed, the temperature coefficient is measured by noting the power doubling time at two fairly close levels of moderator temperature, with other conditions constant. Since a given doubling time corresponds to a certain excess reactivity it is possible to estimate the difference in reactivity which has arisen through the temperature change. The difference in reactivity divided by the difference in temperature gives an adequately accurate measure of the temperature coefficient.

For practical reasons, it is also necessary to have control rods inserted during these measurements. This means that the results are less unequivocal and harder to interpret than if the core had been "pure", see Fig. 22. The measured coefficients are less
negative than the calculated, which qualitatively speaking is in good agreement with the earlier observations regarding the temperature dependence of buckling, since material buckling and reactivity are both measures of the neutron multiplying characteristics of the core.

A factor of practical significance at Ågesta is the tendency of the moderator temperature coefficient, which is negative, to trend increasingly towards positive values with increased burnup. It will never actually become positive at Ågesta, however.

Moderator level

Knowledge of the moderator level coefficient is no more likely than knowledge of the temperature coefficient to be of direct use in the running of the station. The principal aim of these experiments was more to supplement the experimental material on the basis of which the methods of design calculation for heavy water reactors are improved. The moderator level coefficient is obtained by measuring the power doubling time at two nearby and slightly supercritical moderator levels. The difference in reactivity is obtained through the theoretical relationship between doubling time and reactivity, while changes in moderator level are recorded by a specially designed level gauge.

By integration of the measured coefficient with a level dependent factor deduced from elementary reactor physics relationships, from critical level to full tank, a figure for excess reactivity of 9% is obtained for a cold uncontaminated reactor (with a moderator temperature of 20°C and zero power). This is about 1% higher than the design figure for the corresponding condition. When the moderator is heated to 215°C the excess reactivity becomes about 5% and at full power (65 MW) about 2.5%. Although these figures are higher than forecast, and despite increased demands on the control system, it has not been necessary to modify the latter. In fact, the generous margins allowed in design have meant that it has proved possible to reduce the number of control rods required in the reactor.
Burnup

What is directly affected by the higher initial value for excess reactivity is the estimated burnup, or the energy per pound of fuel which can be taken out before the consumption of fissile material and the generation of fission products reduce reactivity so far that further operation becomes impossible without the addition of fresh fuel. To conclude this section, we can present the results of a burnup study based on measured excess reactivity for the reactor in cold state.

During the first years the reactor is assumed to be run at 65 MW output without any redisposition or replacement of fuel, until the reactivity has fallen to approximately 0.1%, see Fig. 23.

When this happens, the peripheral and central fuel assemblies would be interchanged, which yields a clear gain in reactivity, and the reactor would run again until the reactivity once more drops to 0.1% or so. Not until then, after about seven years
(operating for about 5000 h/annum), would any fuel be replaced by fresh. In practice, interchanging and replacement can be carried out during the summer shutdowns.

In the assumed procedure 16 burnt up fuel elements would be removed from a peripheral zone, the other assemblies moved progressively outwards towards the periphery and the vacated space at the centre filled with fresh elements. This procedure, which yields a certain gain in reactivity, would be repeated each time the reactivity dropped to about 0.1% and would be completed when all the first charge had been replaced (change No. 9). This is expected to take about 17 years, on the basis of 5000 h/annum. The mean burnup of the fuel in the reactor when interchanged and at the first and last replacement operations will be 79, 113 and 130 GWh/t, respectively, and the mean burnup of the removed fuel approximately 160 GWh/t.

It should be noted that the relatively high burnup is obtained at the cost of a fairly unavourable power distribution, i.e. a much higher power at the centre than at the periphery. A more normal fuel cycling program would be to charge fresh fuel at the periphery and move it in progressively towards the centre, taking burnt up fuel from the central zone.

In practice, the theoretical state described above will never exist in the Ågesta reactor. The power distribution is affected by control rods, which must be inserted to absorb excess reactivity and furthermore a number of test elements, with properties differing from the standard Ågesta type, will be located in the core during the next few years to investigate their behaviour under reactor conditions.

**Reactivity control**

Dependable control of the reactivity of the reactor is a primary safety requirement. A very conservative assessment of the design calculations was made in the project stage and this resulted in the incorporation of a relatively large number of control rods in the core.
An interesting aspect of experiments at Ågesta is that this represented the first opportunity to study the effects of a large number of control rods in a reactor core. Both the reactor geometry and the design of the control rods are basically simple and enhance the attraction of control rod experiments for comparison with theoretical calculations. The control rods are evenly distributed in the core pattern, are located at the lattice cell boundaries and have not required the introduction of permanent structural materials in the core proper.

The aim of the experiments has been to determine the effect of the control rods on the reactivity of the reactor under various conditions and the disturbances caused in core power distribution. Other problems, such as burnup of the neutron absorbing material in the rods, induced power development and activity in the engaged control rods, have not yet been studied experimentally at Ågesta. Apart from the direct support given to theoretical methods of computing the reactivity effects of the control rods, the experiments also had a safety aspect, viz. a careful check on the reactivity margin available when the reactor is shut down, i.e. when the vessel has a full charge of fuel elements and cold heavy water and all available control rods are inserted.

Another important aim of these experiments has been to establish, through observations of absolute effects, a normal for the reactivity effects of the fine control rods. This is required in dynamic experiments and for the study of burnup effects in the reactor. Various experimental methods of determining the reactivity effects of the control rods have been employed.
Reactivity effects of control rods

<table>
<thead>
<tr>
<th>No. of control rods in core a)</th>
<th>Distance of rods from centre of core a) cm</th>
<th>Reactivity effect of rods Calc. according to crit. moder. level technique</th>
<th>Determined by pulse technique b) %</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0</td>
<td>1.30</td>
<td>1.25</td>
</tr>
<tr>
<td>4</td>
<td>54</td>
<td>4.40</td>
<td>5.29</td>
</tr>
<tr>
<td>1 + 4 = 5</td>
<td>0, 54</td>
<td>4.92</td>
<td>5.23</td>
</tr>
<tr>
<td>4 + 4 = 8</td>
<td>54, 76.4</td>
<td>7.75</td>
<td>9.9</td>
</tr>
<tr>
<td>1 + 4 + 4 = 9</td>
<td>0, 54, 76.4</td>
<td>7.99</td>
<td>10.1</td>
</tr>
<tr>
<td>2 + 4 + 4 = 10</td>
<td>76.4, 108, 120.7</td>
<td>6.64</td>
<td></td>
</tr>
</tbody>
</table>

a) A group of 4 rods is placed symmetrically in the 4 quadrants.
b) Measured at another relative insertion depth.

The critical moderator level was determined for a "pure" core $H_{kr}$ and for cores in which control rods were introduced $H_{ks}$. Using experimentally determined coefficients of moderator level $\Delta \rho/\Delta H$ the reactivity value $\rho_s$ for the control rods is obtained from the equation

$$\rho_s = \int_{H_{kr}}^{H_{ks}} \frac{\Delta \rho}{\Delta H} (H) \cdot dH$$

Large minus reactivity values were measured directly by neutron pulse techniques. A neutron pulse generator placed in the core injected brief (5 μs) neutron pulses. The following decay of the generated prompt neutrons was measured by core detectors and chronologically analysed in multichannel equipment. The exponential decay constant $\alpha$ then gives a measure of the minus reactivity according to the equation

$$\rho_s = \beta - \alpha \cdot \tau$$

where $\beta$ is the total proportion of delayed neutrons and $\tau$ the life of prompt neutrons.
The theoretically familiar relationship between reactivity and power modulations at high frequency (and also at low frequencies for low powers) has been utilized for calibration of the reactivity effect of the fine control rods. As a matter of fact, this is the only method available for direct reactivity measurements at high reactor powers.

The relative reactivity effect of the control rods has been measured by redistributing the control rod groups while maintaining criticality. This method does not allow accurate measurement of the reactivity effect but is valuable for comparison with design models. The control rod measurements at Ågesta have resulted in a large stock of experimental data, which should prove of very great importance in future theoretical comparisons. See Table 3.

The measured reactivity effects of the control rods differ considerably from the computed, which are 10-20% lower. Although the excess reactivity of the reactor in the shut down condition is about 1% higher than previously estimated, the number of control rods required for dependable shut-down is fewer than expected. Only 18 of the 27 coarse control rods originally incorporated in the reactor are now required for normal operation. When irradiating enriched sample elements, 21 coarse rods are used to ensure adequate shut-down reactivity.

The relative reactivity effects of different control rod groups can best be shown in graph form, Fig. 24. The change in reactivity is virtually proportional to the change in temperature and the temperature scale can, therefore, serve as a rough-and-ready reactivity scale. A temperature change of 200°C corresponds to about 3.5% change in reactivity. Fig. 24 therefore gives a good picture of the differential and total reactivity effects of the different control rod groups at various depths of insertion.
The excess reactivity at room temperature is about 9% but the results of tests with 16 control rods show that even this number gives a shut-down margin of more than 4%.

Fig. 24 shows that 16 rods at 20°C moderator temperature have a margin of about 50 cm from the bottom position when the reactor reaches criticality.

The reason for the poor agreement between theory and the practical reactivity experiments is not fully explained. Development work on the theoretical models continues and therefore no comprehensive comparison with the experimental results has yet been started. But these investigations are of considerable importance for the better assessment of conditions in other reactors, such as Marviken.

Theoretical calculations do not satisfactorily predict the relative effect of control rod movement in the axial sense, either. Apart from the fact that the differential reactivity effect is underestimated to the same degree as the integral values, there is a considerable shift upwards of the axial weight function for the control rod reactivity effect.

Power distribution in the core

Knowledge of the power distribution in normal operation and the ability to predict changes due to the introduction of sample elements, control rod movements, raising of the total power and
other actions, are necessary if a nuclear power station is not to be operated with an unreasonably wide safety margin. It is, of course, impossible to carry out experiments covering every operational situation that might occur, so that parallel with the experiments, and on the basis of their results, it is necessary to evolve methods of calculation that can to some extent replace experiments. The situation at Ågesta has been favourable in that it has been possible to measure power distribution in circumstances that closely approximate to normal operating conditions, which is clearly not feasible in research reactors.

It is usual in reactor physics to make a distinction between macroscopic power distribution referring to the reactor as a whole, and microscopic, referring to individual fuel elements. This distinction is observed in principle here, even though certain of the results presented give the gross distribution, for obvious reasons. This applies to the measurement of power distribution axially along the fuel elements, if the power appreciation in the vicinity of the fuel junctions is not presented separately. Also presented as microscopic distributions are the radial distribution across a fuel bundle and the radial distribution in an individual fuel rod.

The methods employed in determining the power distribution pattern in the Ågesta core comprise analysis of recorded coolant temperature rises during passage through the fuel channels, in-core irradiation of metal foils and copper wires as employed for buckling experiments, and measurement of variations in gamma activity along irradiated fuel elements.

By measuring the rise in coolant temperature during passage through the fuel channels it is possible to determine the power development in each fuel element. The Ågesta reactor is equipped for the measuring and recording of this temperature rise. An oscilloscope screen is used to present 2 x 70 pulses the heights of which are proportional to the difference between the inlet and outlet temperatures in the 140 corresponding coolant channels. The screen is photographed and the pulse heights are measured.
Calibration measurements give a base for the assessment of the true temperature rise represented by the photographed pulse height. With the flow of coolant through the individual channels known, it is then possible to calculate the power development.

This method yields no information about the variations in power production along a channel. What is obtained is a total power figure for each fuel element, with no possibility of determining the microscopic distribution. See Fig. 25 and 26. The influence of the control rods on the power distribution in the reactor as a whole and near the control rods in particular is clearly apparent from Fig. 26.

The material buckling was determined by the irradiation of copper wires and the analysis of variations in induced activity at points across the reactor. The power distribution in the core can be studied by similar activity analyses, with the difference that whereas the measurement of activity variations to determine buckling are confined to the central part of the core this is not the case in the study of power distribution. The method depends on the facts that the induced activity is virtually proportional to the thermal neutron flux on the outside of the fuel element and that this neutron flux is proportional to the power developed in the fuel element. (This latter postulation is only approximately correct in reactors containing burnt up fuel).
Fig. 26 Variation of power between fuel elements at various distances from core centre.

A - 4 control rods in position 124 cm up and 108 cm from centre
B - 4 control rods in position 92 cm up and 120.8 cm from centre
C - 1 regulating rod in pos. 20 cm up and 120.8 cm from centre

Hitherto measurements have been made only at zero power. The number of fissions for a given time and volume is then so small as to cause no measurable heat generation, while the neutron flux nonetheless is high enough to give perfectly measurable activity levels in the copper wires. This means, among other things, that the control rods are inserted further into the core than when operating at high power, when the fuel is poisoned by Xenon, a highly neutron-absorbent fission product. The results are no less significant on this account.
Work is now progressing on the improvement of certain methods of computing in which the results of this kind of experiment play an important part, Fig. 27.

Fig. 27 Activity of copper wires after irradiation during reactor control by a bank of 16 control rods. Moderator temp. 215°C (To the right a section of a core quadrant).
All the curves reveal irregularities in distribution in the form of peaks superimposed on a fairly regular (macroscopic) distribution. These originate from the junction regions, where there is no fissile material, and cause an unintended raising of the power in the vicinity. The peaking at the lower part of the curves is due to the bottom reflector.

Gamma radiation scanning of irradiated fuel elements is a relatively new method of determining power and energy distribution in fuel assemblies. The results are obtained without needing to disturb reactor operation, since the measurements are made after removal of the elements from the core. This can involve a disadvantage, however, in that the knowledge of power distribution is retrospective, and may come too late in the worst cases. The main advantage of the method lies in the opportunities it offers to determine fuel burnup by non-destructive methods. By measuring gamma radiation through the disintegration of a certain isotope (say Co$^{144}$) it is possible to determine the quantity of this isotope and thereby the burnup of the fuel. This presumes, of course, that calibration measurements are carried out with known isotope quantities.

In the work at Ågesta the refuelling machine has been used to arrange for irradiated elements to pass a collimator at constant speed. The gamma radiation on the other side of the collimator is detected by a gamma scintillator the signals from which are amplified, integrated during suitable time intervals and recorded through a signal converter that provides digital information in punched tape form.

Gamma radiation is a measure of what must be regarded as a synthesis of the total extracted energy and power during the period immediately prior to the removal of the fuel assembly, since the quantity measured is total gamma radiation and not the radiation from a known isotope of given half-life. However, in view of the relatively brief irradiation to which the Ågesta elements have been subjected it may be taken that energy and power distribution are approximately proportional.
Separate studies of microscopic power distribution have been carried out in conjunction with the comprehensive program for the measurement of neutron capture and the neutron spectrum, together with its variation with burnup. In these studies, thin foils placed in a purpose-made fuel element were irradiated. Activity measurements were carried out in an accurately calibrated gamma scanning station. Foils used comprised Cu, Au, In, Ir, Lu, Mo, W, natural U, $^{235}$U, and Pu. See Fig. 28.

![Figure 28](image-url)

**Fig. 28** Activity of metal foils after irradiation at different distances from a fuel element's centre. Moderator temp. $215^\circ$C.

The information required to determine power distribution is obtained from the curve for $^{235}$U activity. The effect of junctions on power distribution has been studied with the aid of a series of $^{235}$U foils located between U pads in a number of typical rods in the bundle.

The power distribution in a specific fuel rod has been studied through the detailed analysis of irradiated foils covering the full cross-section area of the fuel rod. The results show a distinct microstructure in the fuel, especially in the peripheral rods of the bundle.

**Dynamical experiments**

From a purely operational view, the direct investigations of the dynamical properties of the Ågesta reactor are hardly of any intrinsic value, the reactor being so notably autostabilized. But dynamical studies are justified for many other reasons.
Firstly, it is of interest to check the theoretical models for the reactor's dynamical behaviour, and then there is a desire to acquire experience of measuring methods developed for the purpose, and of the relatively complicated instrumentation required for experiments of the kind.

In the future, more advanced, power reactors the need for accurate theoretical models and study facilities will be much greater than at Ågesta and studies at the latter will help to pave the way. Furthermore, an interesting aspect of the experiments at Ågesta is the way in which the dynamical behaviour of the reactor changes during prolonged operation, i.e. the way in which burnup and usage affect the reactor parameters.

Dynamical experiments at Ågesta have naturally been divided into three stages: Studies of methods and instrumentation techniques, final measurements in the newly started fresh reactor, and continued progressive studies of dynamical properties as burnup advances. Only the first parts of this program have been carried out so far.

Methods and instrumentation for noise measurement have been evolved and preliminarily applied to reactor studies. The object of noise studies is to obtain information supplementing that obtained through controlled disturbances. In addition this more passive technique is intended to achieve an automatic monitoring of the stability limits of the reactor system and of possible internal disturbances.

The most accurate method of measuring dynamical properties is to introduce known disturbances into a variable and simultaneously measure the effect of this on other variables. The relations between response and disturbance signals in the system give the transfer functions of the working system.

Reactivity disturbances have been introduced into the reactor system by means of program controlled regulating rods, coolant flow disturbances — achieved by reducing one of the four main circulators to half-speed — and load disturbances through changes
in the steam valve settings. The ordinary instrumentation of the reactor has been employed for the measurement of response signals, including nuclear power (ionization chamber current), temperature rise in coolant channels, temperatures before and after the main steam generators, and the steam pressure and flow on the light water side.

Measuring has been highly automated in order to permit the simultaneous recording of both disturbance signals and several response signals. Recording is instantaneous on data tapes which, subject to minor editing, can be used directly as input information for detailed analysis in an automatic calculator.

Various ways of introducing the disturbances have been tried. The simplest procedure is the step method and this has been employed for all three input variables. A greater flexibility is enjoyed as regards the reactivity generator (the regulating rods) and trapezoid, BDG and Jensen disturbances have therefore been investigated. The name BDG originates from the initials of the proposers of the method, involving a periodic binary signal with the same amplitude as the basic tone and all overtones up to a certain boundary frequency, while the Jensen system represents a pulse chain comprising a given number of dominating frequencies. Disturbance signals are generated by a program unit which employs tape readers and punched tapes containing the disturbance function concerned.

Trapezoid disturbances give the most accurate results at the highest possible frequency (with the inertia \( \omega \) of the regulating rods as a limiting factor) and this method is therefore always used for calibrating the reactivity effect of the regulating rods. Calibration is possible at high frequency since all effect feedback functions are negligible. The only feedback is then from the internal neutron kinetic processes and the transfer functions for these are well verified both theoretically and experimentally. In this case \( \Delta \rho \) is obtained from the expression

\[
\Delta \rho s = \frac{1}{T}, \frac{\Delta P}{P}
\]
where $\Delta P/P$ is the measured power modulation and $T$ the absolute value of the transfer function at the frequency concerned.

It must be regarded as a fortunate circumstance that the operating mechanism for the regulating rods allows oscillation at a sufficiently high frequency (0.1-0.5 cps). The method is the only one available for reactivity measurements during normal operation and is likely to be frequently employed during future studies of burnup powers and dynamical properties.

Results of the dynamical experiments carried out at Ägesta to date permit the conclusions that trapezoid disturbances are perfectly suitable for accurate reactivity calibration, that BDG disturbances represent the best method for dynamical experiments using disturbance techniques, and that the noise studies have yielded promising results but are not yet fully evolved. Resonances in nuclear power have been observed at 3.8 and 13 cps but the real reason for these remains to be clarified.

Theoretical analogy calculations for the dynamical properties of the Ägesta reactor were prepared prior to the experiments. The model proposed seems generally to have functioned satisfactorily. See Fig. 29.

Fig. 29 Amplitude curves for the transfer function's reactivity to nuclear power.

- Calculated curves
- Measured curves
Special equipment

In some cases the physical experiments to be carried out at Ågesta required the installation of special equipment. Apart from the temperature probes in the outlets of the coolant channels, there are normally no internal measuring devices in the reactor core. For experimental purposes, therefore, special measuring devices were temporarily introduced, either in pressure-tight probe tubes located in temporarily vacated control rod positions or together with the fuel elements, to which were affixed prior to their insertion wires, foil and other test apparatus.

The first measurements were taken at very low nuclear power and sensitive neutron counter chambers were installed in probe tubes for this purpose. An advantage of the arrangement was the elimination of the need for any special neutron source during the first start of the reactor, since the spontaneous fissions in the uranium fuel, representing a natural though very weak neutron source, could be employed. Accurate measurements of the moderator temperature were obtained using resistance thermometers in the same probes.

The moderator level was measured by a purpose-designed differential pressure gauge using a bubble-air flow through a number of tubes inserted to various depths in the reactor vessel. This simple but dependable measuring device gave absolute measurements of the moderator level to an accuracy of closer than 2 mm at all the temperatures concerned.

The neutron pulse generator for reactivity measurements in subcritical states, mentioned earlier, was installed in the core in a special probe tube. This compact generator, which worked with a pulsed deuterium-ion flow accelerated towards a tritium-clad plate, does not work at temperatures above 70°C. A thin annular cooler was therefore introduced between the generator and the tube wall, allowing measurements to be carried out without difficulty even at temperatures up to 215°C.
Pilot ion exchanger (see page 58)
Main coolant loop. Dose rate check points. (See page 62)