At the International Conference on the Peaceful Uses of Atomic Energy, information was given about competing types of reactors which are being developed or built all over the world. The more important of these papers are dealt with in this volume. From it, engineers will be able to study the various lines which are being followed in the struggle for high standards of performance; and by assessing the technological and material requirements of the various systems, their difficulties and their advantages, they will be able to form an idea of the future which lies ahead. In carrying out this study they will remember that a comprehensive nuclear power programme cannot rely on one type of reactor only, and that it will almost certainly be necessary to develop integrated programmes which depend on the use of several types of reactors whose merits are complementary and whose particular advantages fit them to take their part in specific areas of the overall field.

One of the merits of the present publication is, that it brings information together in such a way as to encourage comparison and to facilitate consideration of the way in which integration can be achieved.

1

CANADIAN RESEARCH REACTORS

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Abstract—The heavy-water-moderated reactors, NRX, NRU and Zeep, provide a wide range of reactor research facilities. The NRX reactor has been the major experimental facility since 1947. Studies have been made of the characteristics of this reactor, with particular emphasis on the behaviour of xenon poison in a high neutron flux and the reactivity changes of natural uranium with prolonged irradiation. Unusual occurrences during operation have posed difficult problems of repair. A variety of research investigations have been carried out using the experimental facilities; tests of fuel assemblies for future power reactors under expected operation conditions are being made. The Zeep reactor has proved very useful for lattice studies.

1. INTRODUCTION

THE NRX reactor at Chalk River is a high-flux heavy-water-moderated reactor. Initially designed as a pilot plant for the production of fissile material, NRX was also planned as a major post-war centre for Canadian science. Accordingly, every effort was made to provide suitable arrangements for research. The success of these efforts may be judged by the fact that eight years after it first went into operation NRX continues to be one of the leading research reactors.

The early history has been described by LAURENCE (1947). Following work at Ottawa, the formation in 1942 of a joint U.K.-U.S.-Canadian project at Montreal set the stage for the establishment of the Chalk River Laboratory. When the graphite route to production of plutonium was shown to be feasible this became the main line of development in the U.S.A; the heavy-water route could not compete on the scale required. In 1944 the Montreal project undertook the construction of NRX as a long term contribution.

In the late summer of 1944 plans were made for a low power reactor which could be built in a short time and used for experiments and training as preparation for NRX. Known as the Zero Energy Experimental Pile, or ZEEP, it has been in almost constant use since it became critical in September, 1945.

Concern for the possible obsolescence or failure of NRX emphasized the need for a new research reactor at Chalk River. The diversification in possible types of reactor, which had already taken place, made the choice difficult. Many of the types now available were studied and ruled out, usually because they require enriched material. The final choice was NRU which is under construction. It is a heavy-water-moderated and cooled reactor using natural uranium. In spite of the present availability of enriched uranium, it is unlikely that a substantially different choice would be made today when all the factors are taken into account. NRU will burn natural uranium and produce plutonium, a valuable fuel. In a small enriched reactor with the same flux there would be no production compensating for the rapid consumption of expensive fuel and the restricted volume at high flux would limit its usefulness. For a country with no separation

plants the operation of a high flux reactor on enriched uranium means a continued import of fuel.

This chapter will be devoted largely to NRX with smaller sections on ZEEP and NRU.

2. DESCRIPTION (NRX)

General

The NRX reactor (Fig. 1) is a natural-uranium heavy-water-moderated reactor with light-water cooling of the rod assemblies. The uranium is in the form of long rods. Each rod is sheathed in an aluminium tube having three fins for centering in an outer aluminium sheath. The light water flows through an annulus between the aluminium sheaths, carrying away about 95% of the total heat. These uranium assemblies (Table 1) hang vertically through tubes

Table 1—Dimensions of NRX Rod Assembly

Item	Dimension	Magnitude	
		English	Metric
Uranium	Diameter	1·36 in.	3·46 cm
	Length (approx.)	122 in.	3·10 m
Aluminium sheath	Thickness	0·079 in.	0·20 cm
Outer aluminium sheath	Internal diameter	1·66 in.	4·22 cm
	Thickness	0·040 in.	0·10 cm

Early models had a slightly larger outer sheath. The light water annulus then had 1·5 times the present area.

in the heavy-water vessel, called the "calandria", (Figs. 2 and 3). This is of aluminium and is cylindrical with diameter 8·75 ft. and height 10·5 ft. It is pierced by 198 vertical tubes 2·25-in. diameter for the fuel, control and shutoff rods (Fig. 4). At present eighteen of the calandria tubes, arranged symmetrically in three rings of six, are assigned to shutoff rods. During operation these tubes are empty except for aluminium sheaths. A control rod located about two-thirds of the core radius from the centre takes care of small reactivity drifts by automatic control. A larger tube at the centre, 5·5-in. internal diameter, (the "central thimble") is used for special irradiations.

The calandria is surrounded by a cylindrical graphite reflector and steel and concrete shields. Two thermal columns extend through the shields. Ionisation instruments for control and measurement are located near the outer boundary of the reflector and in a thermal column.

Power and Flux

The present operating power is 40 MW achieved in steps from the original design maximum of 20 MW. The increase was made possible by operating experience and design changes. The limits set on heat transfer in the rods were unduly restrictive and have been modified after recalculation with new data. Addition of a second heat exchanger in the heavy water cooling circuit permitted

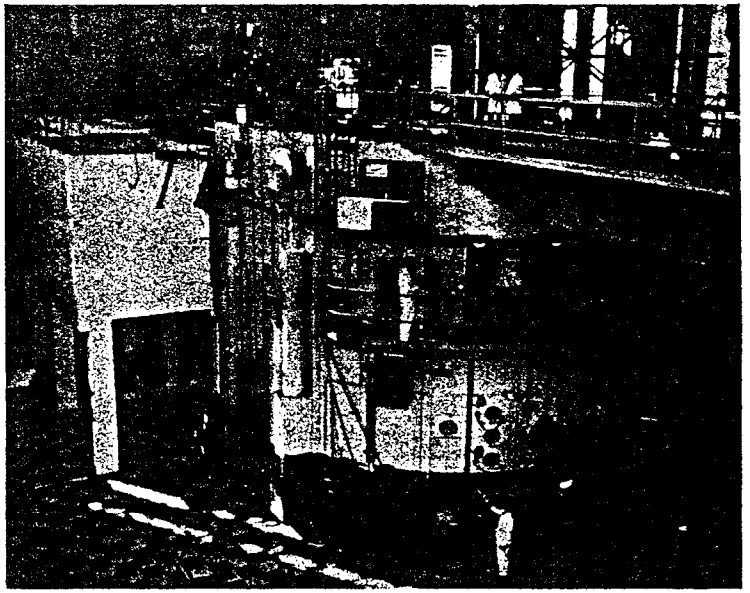

Fig. 1. The NRX reactor. The mid-plane of the calandria is about six feet from the floor and is marked by the boundary between the light and dark paint on the cylindrical concrete shield. To the right are beam exit ports where holes run through the shield and provide neutron beams for experimental purposes. To the left is the lead door (with pulleys and counterweights) which covers the south thermal column.

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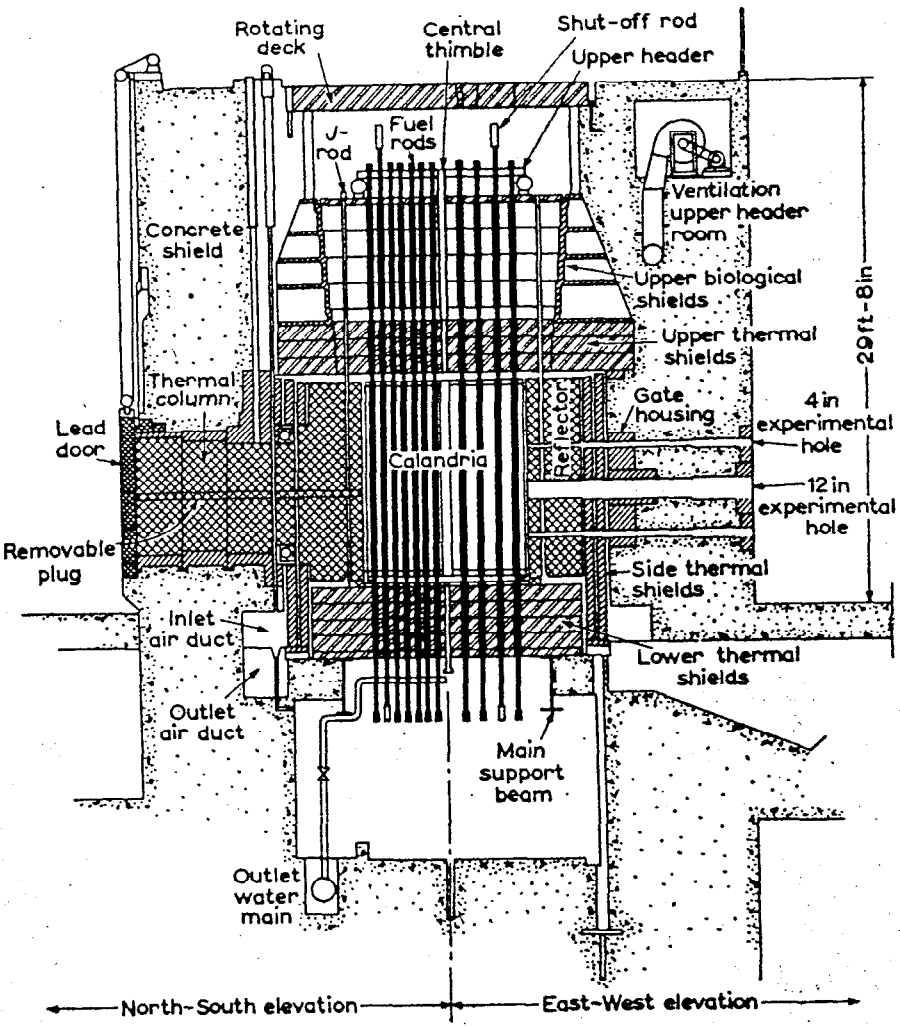

Fig. 2. Approximate vertical sections of the NRX reactor. The 4 inch and 12 inch experimental holes are not in the same vertical plane. Their actual locations can be seen in Fig. 1.

operation at 30 MW. The limit was then imposed by the fear of excessive thermal stresses in the upper steel layer of the lower thermal shield. During restoration following an accident in 1952 top and bottom shields were modified and the increase to 40 MW was made. This maximum power is permitted only with the calandria almost full. Before xenon poison builds up, the reactor operates at reduced heavy-water depth and the power is reduced proportionately

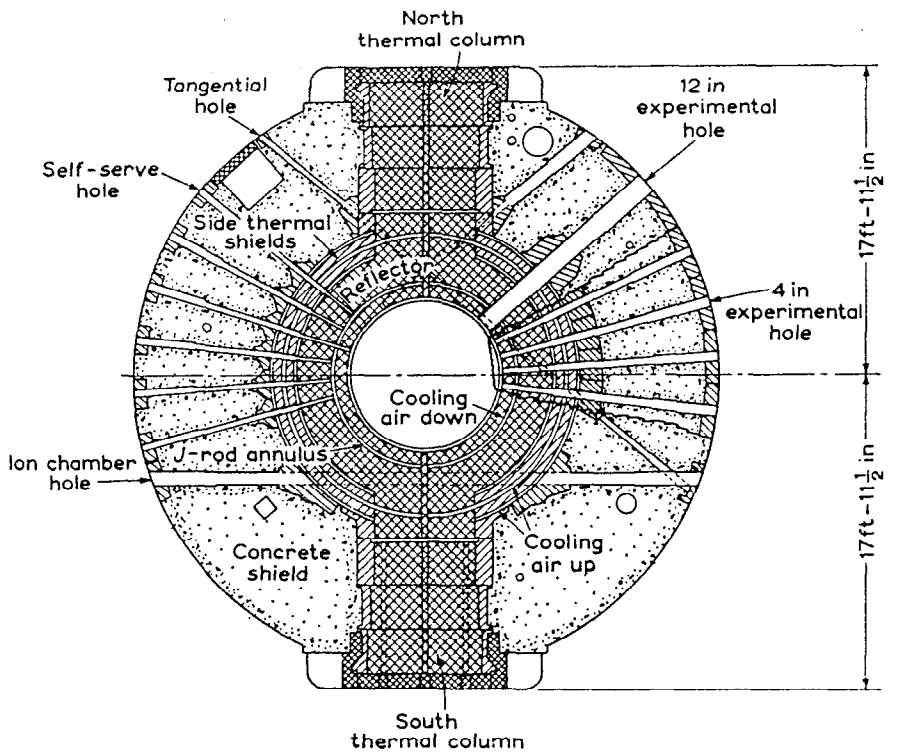

Fig. 3. Approximate horizontal cross section of the NRX reactor at the mid-plane. The cutaway on the right is at the level of the 4 inch experimental holes. The tangential and ion chambers holes shown are not, in fact, at the mid-plane.

to keep the maximum heat flux nearly constant. Fortunately the lack of information on attenuation resulted in a cautious overdesign of the biological shields so that they are adequate for the higher power.

The temperatures in the rods are not known precisely. Calculations for idealised situations show that the $\alpha\beta$ transition temperature, regarded as an operational limit, is not reached. On the other hand, at the high rates of heat transfer encountered, the temperature drop at the uranium-aluminium interface is uncertain. There is no evidence of trouble from such a cause. It has been shown (HADDOW, 1955) that the thermal expansion of the central region of the rod raises the stress in the surrounding metal above the elastic limit. The serious consequences which could be predicted have not occurred.

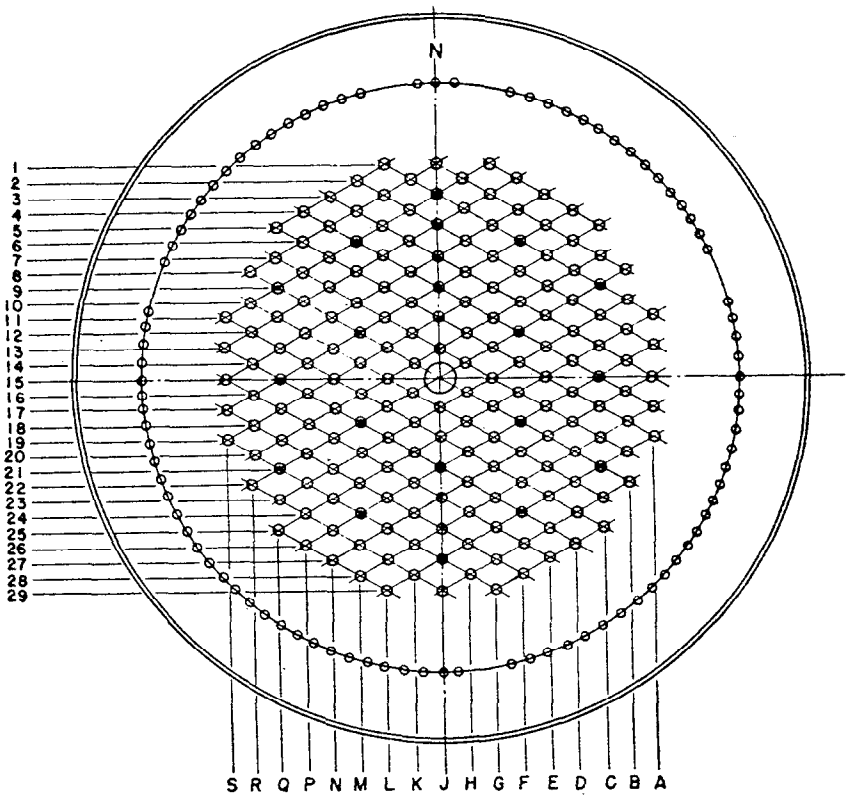

Fig. 4. A schematic drawing of the NRX lattice arrangement showing the numbering system and J-rod positions. The large circle J-15 is the "central thimble". The control rod is at J-5 and shutoff rod positions are marked for example as at J-3.

Some values of the conventional flux at 40 MW are tabulated (Table 2). By conventional flux is meant the quantity nV where n = neutron density in n/cm^3 and $V = 2{\cdot}2 \times 10^5$ cm/sec. (2200 m/sec.). The basis for these values is discussed by HONE, HURST, & WESTCOTT (1955).

Table 2—Flux, (nV_{2200}) for 40 MW Operation

Location	Remarks	Maximum Flux
Empty calandria tube	Based on 98 barns for gold	$6{\cdot}8 \times 10^{13}$ neutrons/cm²/sec.
Average across uranium rod at position of maximum flux	Based on σ_f(U) = 4·18 barns and 198 MeV per fission	$2{\cdot}7_6 \times 10^{13}$ neutrons/cm²/sec.
Moderator	Maximum in a normal lattice cell	$5{\cdot}6 \times 10^{13}$ neutrons/cm²/sec.

Experimental Holes

The experimental holes (Figs. 2 and 3) are of two diameters, 12 in. at the calandria for the three holes on the mid-plane, and 4 in. for 12 holes arranged in two sets of six, 2·5 ft. above and below the mid-plane. The gates just beyond the thermal shield have proved to be very useful features, permitting the neutron beams to be cut off for changes to apparatus.

The experimental holes are too crowded for convenient use of the large equipment required for neutron research. Historically the close horizontal spacing arose from a desire to leave an area around the south thermal column free of openings.

Thermal Columns

Each of the two thermal columns is a graphite assembly, roughly 6 ft. square, extending from the reflector to the outside of the shielding where it is covered by a door made of 0·05 in. of cadmium and 9 in. of lead. The outer portion of the graphite consists of three movable sections each 2·5 ft. thick. Openings of various sizes may be left in the graphite and are matched by plugs in the door. A central stringer may be taken out all the way to the calandria giving a hole 4·5 in. diameter at the inner end. Central openings up to 25 in. square may be left in the movable sections, and six smaller holes may be made near the edges.

An iris diaphragm of cadmium is built into each column at the inner end of the removable portion to provide a continuously variable control of flux independent of reactor power.

The flux along the axis is

$$nV = 2{\cdot}3 \times 10^6 \exp(z/32) \text{ per MW for } 50 \leqslant z \leqslant 200 \text{ cm}$$

where z is the distance measured inwards from the door in centimetres. The cadmium ratio in the column is a function of position along the axis, going through a maximum of about 20,000. This is not as high as was expected. The reduction has been ascribed to photo-neutrons generated from the C^{13} by gamma rays of neutron capture in the aluminium of the calandria and in the iron of the column walls.

Some of the ion chambers for reactor control are installed in the south thermal column; their indications follow closely the average output of the reactor and they are less sensitive to detailed shadow effects than ion chambers closer to the calandria.

In front of the south thermal column the floor may be removed to uncover a pit 28 ft. long, 20 ft. wide and 10 ft. deep. One purpose was to provide a clear radius 14 ft. from the centre line of the column. It has proved very useful as a shielded room, and has also been used for decontamination of large pieces of equipment.

Tangential and Ion-Chamber Holes

Several horizontal holes traverse the reactor structure. They pass through the reflector outside the calandria providing access from both ends if required. Five of these, 2·875 in. in diameter, are tangent to the inner face of the reflector. Eight others, 8 in. in diameter and farther from the calandria, are for ion chambers and most of them have chambers installed.

Two holes, 3 in. diameter, are tangent to the inner face of the reflector but run to one face only of the reactor. The diameters are about 9 in. for a distance of 6 ft. from the outside of the reactor.

3. IRRADIATION FACILITIES (NRX)

J-Rod Annulus

Because one of the uses considered for NRX was to produce U^{233}, provision was made for the irradiation of thorium. An annular space 2·5 in. wide (Fig. 3) was left in the reflector outside 9 in. of graphite. A ring of 89 holes through the upper shielding permits rods to be hung in the annulus. These rods, known as J-rods, are cooled by the main air stream and about 1 kW can be dissipated per rod without serious temperature rise. One purpose of the graphite layer was to slow down the fast neutrons and reduce the fast fission in the thorium.

The J-rod positions have been used for a variety of irradiations, particularly where a large bulk of irradiated material is required and maximum flux is not of primary importance. In addition to thorium the rods have contained potassium nitrate to produce C^{14} and K^{40}, lithium to produce tritium, cobalt, other miscellaneous substances, and small loops. Demands for high flux at the experimental holes and thermal columns have interfered with full use of the positions. The flux in the annulus at the mid-plane is $2{\cdot}6 \times 10^{11}$ per MW.

"Self-Serve" Apparatus

The west face of the reactor is largely taken up by devices for irradiating small specimens. These "self-serve" arrangements permit the insertion and removal of the material without affecting reactor operation. The specimen is contained in a cylindrical aluminium capsule which fits inside a superpure aluminium sphere 2·25 in. diameter. The sphere is rolled down an inclined channel through the shield into a socket in a retractable plug which is pushed forward for irradiation. For withdrawal the plug is pulled back and turned over, spilling the sphere into a channel down which it rolls to a flask on the outside of the shield. The capsule is removed from the sphere in a special apparatus and the sphere is re-used after a short interval for decay. There are eighteen units; three of these penetrate to the calandria wall and have positions for five spheres, the remainder penetrate only to the J-rod annulus and have positions for three spheres making a total of 60 positions. A maximum flux of $4{\cdot}3 \times 10^{11}$ per MW is available.

Tray Rods

As originally planned and operated all the lattice sites were occupied by uranium or control or shutoff rods. It soon became apparent that one control rod, instead of the four installed, was ample. Three control rods were removed and the positions made available for irradiations. Since that time the use of lattice positions for isotope production has increased and the mechanism for inserting and removing samples at a lattice position has evolved into what is called a "tray rod." This consists of an aluminium tube with shielding end-plugs. Cylindrical capsules like those used in the self-serve spheres are held in spring clips at intervals of a few inches along the length of the tube which is cut away to provide access at each clip. Capsules can be added or removed individually

when the tray rod has been raised to the correct height in the rod-removal flask. Owing to interference between this flask and the control rod, the tray-rod loading cannot be changed during reactor operation. Alterations to be made shortly in the reactor control system will permit changing the isotope loading with the reactor at power. The tray rods are air-cooled and are used for cobalt and other materials including fissile material in milligram amounts. The maximum flux, $1{\cdot}7 \times 10^{12}$ per MW, is available in tray rods.

Pneumatic Carriers

Two pneumatic devices for rapid insertion and removal of samples from lattice positions are provided. One of these is installed at the radius of maximum flux. The samples, contained in superpure iron capsules, travel from the Research Chemistry building into the reactor and at the end of the irradiation return to any one of three stations in the Chemistry building, all by pneumatic action. The other pneumatic carrier delivers capsules to the top of the reactor; in general no chemical separations are made with the specimens from this carrier.

Neutron Converters

Hollow uranium cylinders in the reactor neutron flux provide an enhanced fast flux in their interior. A number of neutron "converters" or "transformers" based on this principle have been used in experimental holes, the J-rod annulus, and lattice positions. These converters are primarily for the study of fast neutron irradiation of solids. The converters at lattice positions will take samples up to 0·375 inch diameter; much larger samples can be used in converters in experimental holes but the flux is smaller.

4. REACTIVITY AND ITS CONTROL (NRX)

The Heavy-water System

The level of the heavy water in the calandria is controlled by a weir and is adjustable over the full height of the calandria. The weir is contained in the so-called weir box which is connected to the calandria by flexible piping and which can be adjusted in height with a precision of $\pm 0{\cdot}2$ mm.

Heavy water is pumped into the calandria at 5 IGPM from a storage tank, and returns to the storage tank by flowing over the weir. There is a side circulation of the heavy water at the rate of 200 IGPM from the calandria, through heat exchangers and pumps, and back to the calandria but there is no evidence of interference with control by the weir as long as the circulation rate is reasonably steady. A small oscillation has been observed in the neutron flux, of amplitude 0·4% and frequency 0·82 cycles/second. This is attributed to selective response of reactor power to the fundamental radial mode in the surface disturbance caused by the heavy-water circulation.

The position of the weir provides the primary control and measure of reactivity. After the weir box has been moved, some time is required for the heavy water to reach the new level in the calandria. Following a small change in height, readjustment is almost complete after 15 minutes but for high precision 30 minutes must be allowed. To increase the heavy water level rapidly, for instance during the initial filling of the calandria, a pumping rate of 110 IGPM is available.

Dump valves are provided which can empty the calandria at an initial rate of approximately 300 IGPM. The operation of these valves is normally automatic, as part of the protective system.

The temperature of the heavy water is an important item in the reactivity balance. This temperature is not measured in the calandria; instead a recorder-controller is connected to a resistance thermometer element in the line carrying the water from the calandria to the heat exchangers. Reactivity tests during rapid temperature changes have proved that the indications of this instrument represent sufficiently well the effective moderator temperature. The temperature may be read to 0·1°F and is controlled automatically through variation of the light-water supply to the heat exchangers. Under almost steady conditions the temperature remains within ± 1°F and for transient conditions similar constancy can be realized by manual adjustment of the control point.

Control Rod

One water-cooled control rod is installed in the reactor. Manual operation, with a positioner at the control desk, or automatic operation can be selected. The control rod is automatically driven fully into the reactor after a trip signal, and the positioner must be set to zero before the control rod can be raised.

Shutoff Rods

NRX shutoff rods are made of steel tubing packed with boron carbide, each rod being approximately 10 ft. long. In the "up" position the rods are held in the upper shield by electromagnets located in the upper header room. When a series circuit through the magnet coils is broken by a "trip" signal, all the rods are released and are accelerated into the reactor by compressed air, only 0·3 second being required for the first 5 ft. of travel.

The shutoff rods are raised one at a time in a fixed sequence using compressed air for lifting. The maximum removal rate is one rod per thirty seconds. The rate is determined by the travel time of a mechanical valve so constructed that lifting air may be applied to only one rod at a time.

Eighteen shutoff rods are installed in the reactor, the rods being located symmetrically at lattice positions on three circles so that the distribution is reasonably uniform throughout the lattice.

Reactivity

A flux survey showed that the extrapolated height, h, can be written

$$h = \text{Weir-Box Setting} + 22 \text{ cm}$$

and the same extrapolation distance (22 cm) is usually assumed to apply at all settings. The quantity π^2/h^2 and the difference, $\Delta(\pi^2/h^2)$, of two such quantities are useful measures of reactivity. On account of the reflector the relationship between $\Delta\mathscr{H}^2$, the change in Laplacian, and $\Delta(\pi^2/h^2)$ is

$$\Delta\mathscr{H}^2 = 1{\cdot}07\ \Delta(\pi^2/h^2)$$

An alternative measure of reactivity change that is often used is $M^2\Delta(\pi^2/h^2)$, a conventional expression for the change in multiplication, where $M^2 = 312$ cm^2. The quantity $1000\ M^2\Delta(\pi^2/h^2)$ is said to be in "milli-k" (mk).

With a natural uranium lattice and no reactivity load, NRX has a critical weir-box setting of approximately 180 cm. Excess reactivity corresponding to a full calandria, 315 cm, is about 50 mk. Complete loss of the ordinary cooling water from the lattice would contribute an additional 20 mk. During normal operation, the NRX reactor often contains heavy reactivity loads due to the installation of experiments in the central thimble or in normal rod positions. This reactivity load is usually compensated by the addition of enriched uranium or plutonium-aluminium alloys.

The reactivity load of the shutoff rods is approximately 97 mk, inner, middle and outer rods being worth 8·7, 4·7, and 2·7 mk respectively. The control rod is worth 3·1 mk.

Because the compensation of large reactivity changes is by variation of the heavy-water depth simple geometry is preserved. Flux perturbations due to the control rod can be eliminated or kept small, making analysis of reactivity measurements relatively simple.

5. CONTROL AND SAFETY SYSTEM (NRX)

General

The NRX control and safety system must provide protection in the event of possible accidents resulting in an increase of reactivity such as loss of cooling water by boiling, dispersal of enriched fuel in the moderator due to failure of a fuel rod, removal of reactivity loads, or failures of reactivity control devices.

The choice of 18 shutoff rods, controlling 97 mk, has been made on a somewhat arbitrary basis. If we imagine a combination of unlikely circumstances, including no xenon poison, the calandria filled with heavy water, no cooling water and one or two shutoff rods stuck in the "up" position, the need for 18 shutoff rods can be justified. Since the NRX reactor in ordinary operation also contains enriched fuel rods to compensate for heavy experimental loads, the complement of shutoff rods can never be chosen to provide adequate safety in all possible circumstances. In the last analysis correct operational procedure must be maintained.

In an installation as complicated as the NRX reactor there are hundreds of variables which must be monitored, and departure from the normal range requires automatic protective action, usually a "trip" or shutdown of the reactor. For instance the 176 positions available for uranium rods all have flow monitoring instruments, and a low flow instigates a reactor "trip."

The design has undergone many minor and major changes during the 8 years operation of this reactor. The present system resulted from redesign following the accident in December 1952.

Further changes are planned during March 1956. Such changes are made to satisfy two continuing requirements, the desire to incorporate in the reactor advances which match improvements in the philosophy of reactor safety, and the desire to improve the reliability and operational convenience of the system.

Conditional Trip Level

One improvement is based on the concept of a "conditional trip level." If the power is below this level (arbitrarily defined by the neutron flux being less than

1% of the flux at full power) many of the signals which would cause trips under normal operation are automatically converted to alarms. This is permissible because thermal effects are then not significantly influenced by the position of the shutoff rods, due to the heat generated by the fission products. Trips which are important even at low power, for example indications of excessive rate of rise of power, are called "absolute" trips. The shutoff rods may be lifted provided no absolute trips are actuated. This arrangement helps to avoid poison shutdowns as the time-consuming operation of raising shutoff rods can be carried out in parallel with the investigation and correction of conditional trip signals. To ensure that all shutoff rods are up when the reactor is at power, any shutoff rod down is a conditional trip.

Basic Safety Requirements

Four basic requirements have been considered in the design. These are (a) monitoring of power and rate of rise of power (b) rapid shutdown (c) limited rates of increase in the heavy-water depth and of withdrawal of control and shutoff rods (d) provision of a "safety bank" of shutoff rods.

The monitoring of power and rate of rise of power are in most respects conventional. However, the provision of unequivocal rate-of-rise trip signals is complicated by the abrupt reactivity change caused by the lifting of a shutoff rod. This problem has been solved by designing the rate-of-rise trips so that a transient signal of the size normally produced by the lifting of a shutoff rod will not trip the reactor if the power is below the conditional trip level.

Rapid shutdown must be provided because the possibility of a sudden reactivity increase resulting from loss of cooling water is inherent in this type of reactor. With the present shutoff rod design the shutdown action is very rapid. Gravity acceleration of the shutoff rods is considered adequate and will be used for new rods which will soon be installed.

The limitation on rate of change of heavy water level is set by the pumping capacity of 110 IGPM which gives 10 cm rise per minute. The limit on the control rod is set by the driving motor and is such that the control rod can be removed in about 20 seconds. With the present design the limit on the speed of withdrawal of a single shutoff rod cannot be chosen at will, the lifting by compressed air takes two or three seconds and introduces a high rate of change of reactivity, although the reactivity increment is only the effect of one rod. In practice the rate of change of reactivity averaged over several rods is limited by restricting the rate of withdrawal to one rod in 30 seconds. This is a compromise between the increased safety with a lower average rate, and the operational necessity of a relatively rapid start-up to avoid prolonged shut-downs caused by the growth of xenon poison.

The principle of a "Safety Bank" of shutoff rods has been adopted in NRX operating policy to provide some measure of protection against operational errors in loading the reactor. As soon as the reactor shuts down and absolute trips have cleared, operational procedure requires that a "Safety Bank" of four shutoff rods be raised, and this bank of shutoff rods is ready to drop in the event of any absolute trip signal. The operation of the "Safety Bank" will then provide a warning of faulty loading procedures before they have been carried to the point of a reactor runaway with no shutdown protection.

6. MEASUREMENT OF REACTIVITY (NRX)

General

Reactivity measurements in NRX relate primarily to the critical value of weir-box setting and thus to h_c the "critical height" which is found most directly by operating at critical. When a measurement is to be made of the change in critical height due to some operation involving shutdown, for example the substitution of one rod for another, the power should be kept below 50 kW to avoid complications arising from thermal effects and xenon production. After operation at high power there is a strong source of photoneutrons and the reactor is appreciably sub-critical at low power. The weir-box setting for steady low power does not then represent a critical condition. Moreover the behaviour is sluggish and to the extent that the photoneutron sources are decaying the equilibrium is secular. Consequently the direct method is suitable at low power only if the uranium rods are new. It was used when NRX was first started and again in 1954, especially for determination of temperature coefficients.

Three methods which have been used to find the critical height at low power will be outlined. These are the measurement of reactor period, extrapolation to critical, and variation of the sub-critical flux by a calibrated absorber.

Reactor Period.

The use of dynamic measurements for reactivity determinations is standard practice with graphite reactors, involving the well-known "inhour" formula. Fig. 5 shows the growth of the flux during a period measurement. Without the photoneutrons the early part would be a straight line (the curvature late in the run is produced by thermal effects and sets an upper limit on the useful power range). The lower portion of the curve can be rectified by the addition of a constant, found by trial and error, with the result shown. From such curves periods can be measured even when the source is large. The added constant varies as the source strength and inversely as the excess multiplication. A calibration curve (Fig. 6) obtained under suitable conditions can be used to relate periods determined in this way to excess height. There are difficulties in following a rapidly changing reactivity; for example, the weir must be set at a suitable height well in advance of the period run to permit the moderator to reach its equilibrium level. If a sensitivity of 0·01 cm is desired the period must be measured accurately. In practice periods from 100 to 200 seconds have been used, the accuracy aimed at being 1%. This method was applied in a study of the growth and decay of xenon following shutdown from 1·4 MW. It appears to be unsuitable for use after shutdown from the present full power.

Extrapolation

Instead of finding the critical height from period measurements with excess reactivity, extrapolation may be made from a sub-critical condition. An approach to critical by raising the weir in steps, with sufficiently long intervals for the power to level off at each step, permits an extrapolation to critical as in the common start-up procedure for new reactors. Over a range of many centimetres the graph of $(\text{flux})^{-1}$ vs. π^2/h^2 is generally a straight line which may be extrapolated to meet the π^2/h^2 axis at π^2/h_c^2. Many measurements of this kind have been

made in NRX particularly for determining reactivity effects of individual rods. The method is basically capable of good precision but it is time-consuming as at least two weir settings must be used. The extrapolation itself, even with

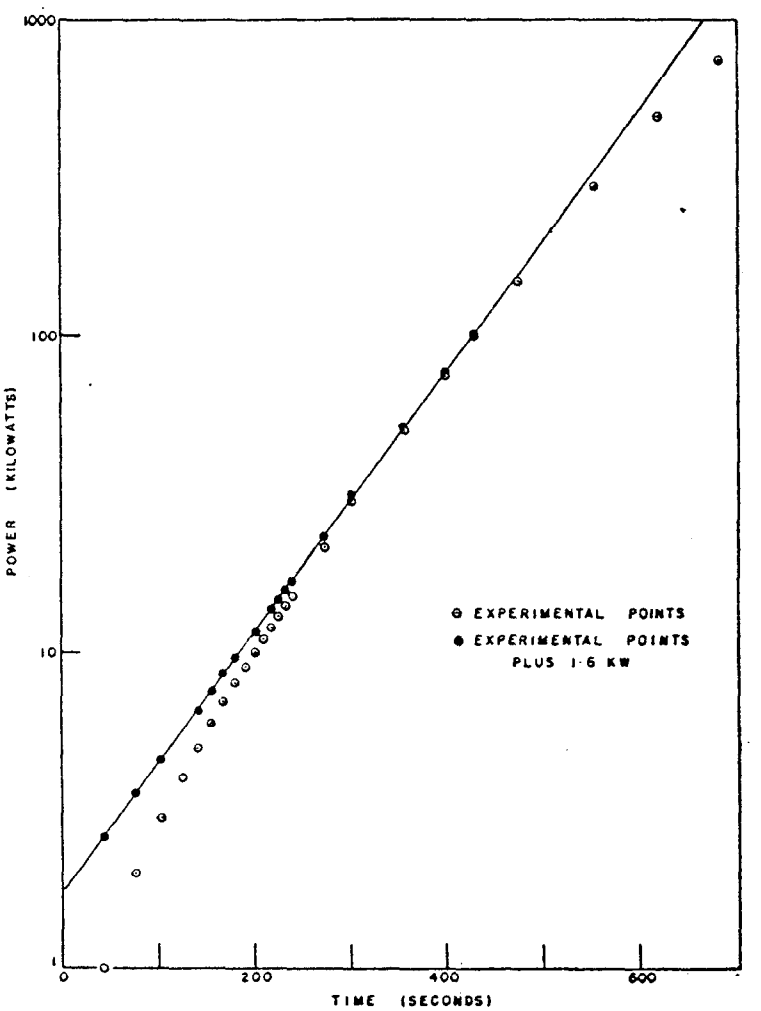

Fig. 5. A typical period run. The curvature at low power is caused by the presence of photoneutron sources and can be removed by the addition of a constant power, in this case 1·6 kW, at all the experimental points as shown. The curvature beyond 100 kW is due to thermal effects. The period obtained from the straight line is 104·5 seconds (doubling time 72·4 seconds)

a source equivalent to 5 or 10 watts, is accurate to $\pm 0{\cdot}02$ cm but drifts in the reactor can introduce errors.

When measurements were attempted with the calandria nearly full the points did not lie on a straight line. The curvature was ascribed to a dependence of

upper extrapolation distance on heavy-water level (a decrease of several centimetres is expected between low level and high level but the manner of decrease is not known). The method was abandoned and replaced by a technique using a standard absorber. This has become the preferred method for sub-critical reactivity measurements.

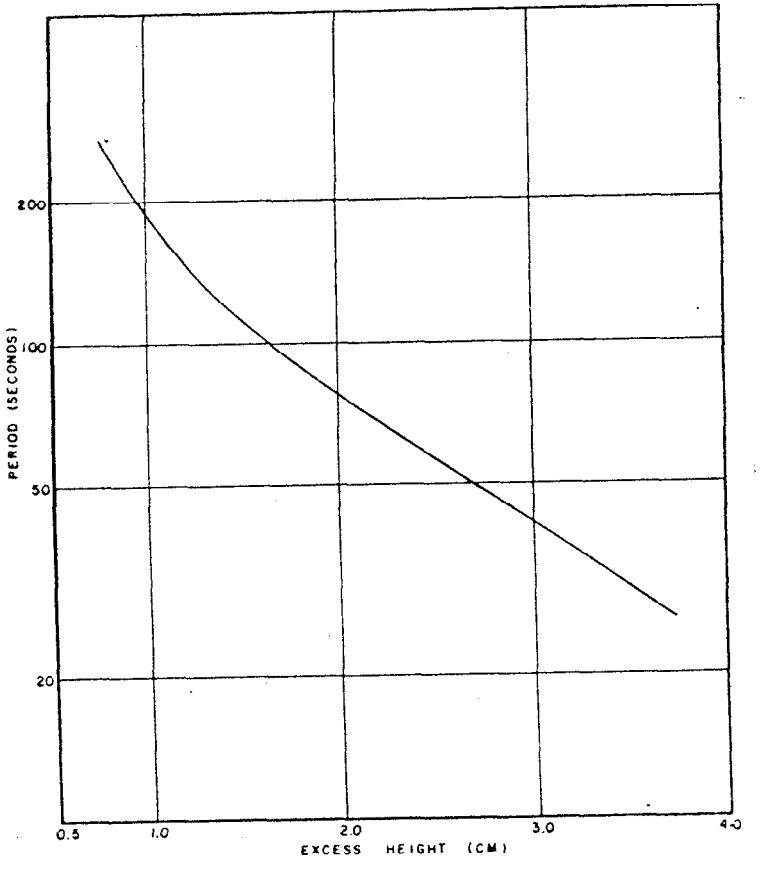

Fig. 6. Reactor period versus excess height of the weir above a critical weir setting of 220 cm.

Standard Absorber

In this method an absorber of known reactivity effect, R, is inserted into the already sub-critical reactor, causing the flux to decrease. The initial flux, D_1, and the final flux, D_2, are measured at a location remote from the absorber. The initial amount below critical is given by $RD_2/(D_1 - D_2)$. Thus without moving the weir the reactivity deficit is obtained. When several weir settings are involved it is important to use an absorber having radial symmetry and extending the full height of the core. Two applications of this technique have been made: a determination of reactivity changes resulting from irradiation of uranium rods, and a study of xenon poison after shutdown from 40 MW. In the first application the light water in three fuel rods symmetrically located was used as the standard absorber. Water was drained from the three rods, and the cooling was maintained by a downward flow of compressed air. Insertion of the absorber consisted of turning off the compressed air and flowing water into the rods from below. The effects to be measured were small and drifts in the reactor were considered to be the major source of error as each operation took several hours. The reactivity effect of the light water was found by measurements at several weir settings. In the second application the absorber was an aluminium rod 3 metres long and 3·75 cm diameter which could be inserted into the central thimble. The same procedure was used for calibration; in this case a direct measurement of the change of critical height was also possible.

In the study of xenon the aluminium standard absorber was operated on a regular schedule for six days; at first on a seven-minute half-cycle, then a ten-minute, and finally a twenty-minute half-cycle. Two similar ion chambers, one with a boron coating, were located side by side in a thermal column and connected in opposition to a recorder with an expanded scale. Gamma ray and activation currents, which were in any case small, balanced each other; the net reading represented the current produced in the boron-coated chamber by the neutron flux.

Errors and Corrections

A number of errors and corrections applicable to the xenon study will be discussed.

The formula $RD_2/(D_1 - D_2)$ assumes equilibrium whereas the cycles were short compared to many of the photoneutron lifetimes. The error due to lack of equilibrium has been calculated. It is less than 5×10^{-5} in k and has been ignored. This error is almost independent of the distance from critical, the sluggish behaviour near critical being compensated by the greater magnitude of the flux.

The insertion of the standard absorber disturbs the radial flux distribution and increases the radial leakage. A simple calculation of this effect for a bare rectangular pile was made with the one-group approximation by finding the second harmonic leakage current produced by insertion of an equivalent absorber along the axis. The result, 0·7%, was applied to the readings with the standard absorber inserted.

At its maximum the xenon poison amounted to more than 30 mk. The resulting flux perturbation caused the reactivity effects of the xenon and the standard absorber to be non-additive. A correction was calculated, again for a simple model of a bare reactor, and applied to that portion of the xenon which grew from iodine after the shutdown. (The distribution of the xenon already in the reactor at the time of shutdown was considered to be almost uniform as a result of burnout and thus not to interfere with the standard absorber.) The maximum correction amounted at most to 1·2 mk and was appreciable only for a few hours near the poison peak.

A few other corrections, although not associated with the standard absorber, will be outlined at this time.

The calculated self-interference of the xenon was very small and was neglected.

In the course of the measurement the weir-box was lowered in five steps from 259 cm, the level at which the iodine was produced, to 194 cm. The measuring flux did not match the poison distribution and with the "statistical weight theorem" a factor was calculated for each weir setting to reduce the measurements to a common basis. By chance the factor was nearly the same at 259 cm as at 194 cm. The largest correction amounted to 2%.

Isotope-producing loads were located in three symmetrical positions H-18, L-15 and H-12 (Fig. 4). The loading was not quite uniform vertically and the change with weir setting was also calculated. This amounted to 0·2 mk between 194 cm and 259 cm.

Because the calculated interference corrections are small, no attempt has been made to improve the accuracy by using a model which would be a better representation of the reactor.

7. CHARACTERISTICS OF NRX

General

The design and operation of high-flux natural uranium heavy-water-moderated reactors requires a good understanding of the various effects which cause reactivity changes. In NRX, major reactivity changes are associated with the growth, radioactive decay and burn-up of Xe^{135} poison, while reactivity changes due to temperature effects and other causes are an order of magnitude less important. Studies have been made to verify the predicted reactivity changes based on knowledge of lattice theory, relevant cross sections and other experimental data, and to search for additional reactivity changes peculiar to high flux operation which might have been undetected in previous lower flux experiments.

Measurements were made during 1947 and 1948, at the time of the initial start-up experiments on NRX, and during February and March, 1954, when the reconstructed reactor was put back into service with a fresh charge of uranium. The measured changes in reactivity of NRX agree reasonably well with predictions based on data from other types of experiments. The measured reactivity changes due to Xe^{135} are about 10% smaller than the predicted changes based on a 6·4% fission yield. The measurements also favour a half life for I^{135} less than the currently accepted value of 6·68 hours. Since no marked discrepancy between calculation and experiment has been observed, some confidence can be placed in calculation of reactivity changes to be expected in higher flux reactors.

Experimental results on the temperature coefficients and so-called "prompt poison" are discussed first; these effects cause relatively small reactivity changes in NRX but their values must be known accurately to allow for their effects in the analysis of the reactivity changes due to Xe^{135} poison.

Temperature Coefficients of NRX

Reactivity changes due to changes of heavy-water-moderator temperature, and changes of rod-assembly temperature, have been measured separately. The experimental results of measurements taken during 1948 are shown in Figs. 7 and 8. During the cooling of the heavy-water moderator, Fig. 7, the

power was kept nearly constant by adjustment of the weir box to compensate for the reactivity change with moderator temperature. The scatter of the points is a good indication of the precision of the measurements. The points with circles are of lower precision since they have required rather large correction (>0·05 cm) for the difference between the actual moderator level and the equilibrium level at the given weir-box reading. The curve has been drawn to

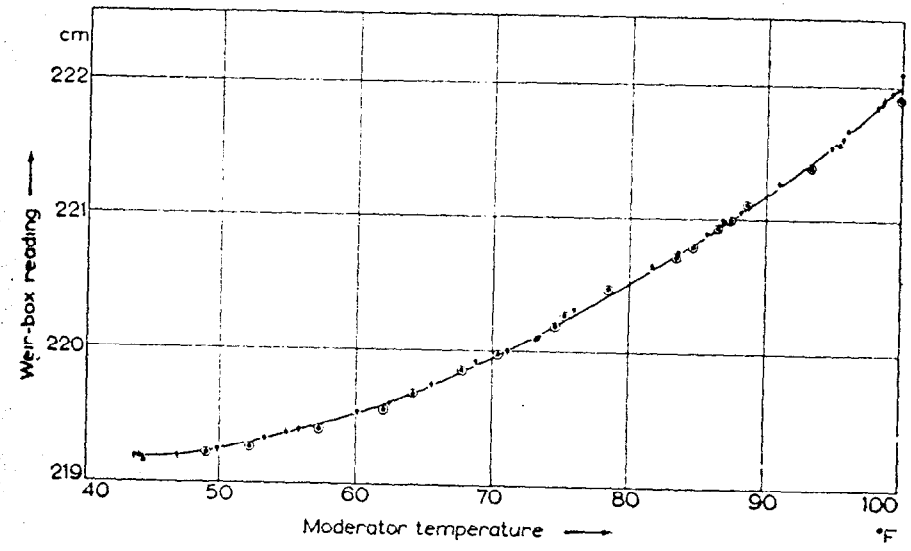

Fig. 7. Variation of critical weir-box reading with heavy-water temperature. The measurements were taken during cooling of the heavy water from 100°F to 44°F. The power was kept nearly constant at a few kW by adjustment of the weir-box.

- reading taken at a power maximum
- reading taken at a power minimum
- ⊙ heavy-water level not in equilibrium, correction of >0·05 cm applied.

give a reasonable fit to the experimental points. The curvature results from the non-linear relation between heavy-water density and temperature. A preliminary experiment over a somewhat smaller range of temperatures gave similar results.

The points in Fig. 8 were obtained in a similar experiment in 1948 during the cooling of the rod-assemblies from 117°F to 37°F. For this experiment the light water coolant circulated in a closed loop with temperature control through a heat exchanger using steam for heating and river water for cooling.

The temperature coefficient of the rod assemblies as measured in this experiment was much larger than had been expected. The difference between experiment and the preliminary calculation has been attributed to the strong moderating properties of the light water in the cooling annulus surrounding each rod. Although the coolant annulus was only 0·1 inch thick, a change in light water temperature caused a corresponding, though smaller, change in the temperature of the thermal neutrons reaching the uranium.

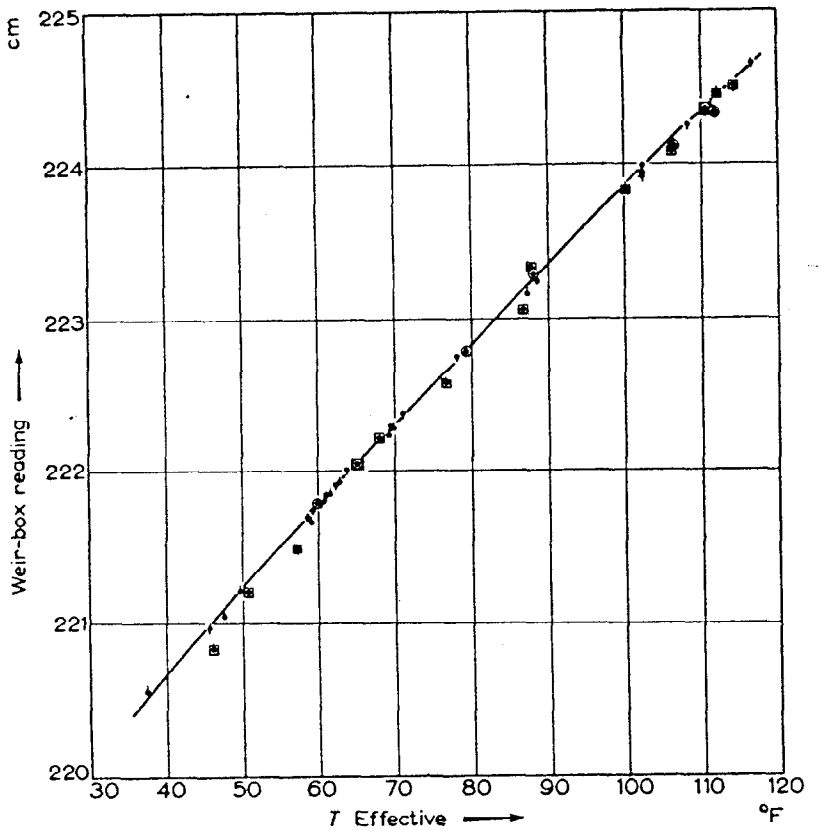

Fig. 8. Variation of critical weir-box reading with rod-assembly temperature. The measurements were taken during cooling of the light water from 117°F to 37°F. The power was kept nearly constant at a few kW by adjustment of the weir-box

- reading taken at a power maximum
- reading taken at a power minimum
- ⊙ heavy-water level not in equilibrium, correction of >0·05 cm applied
- ⊡ temperature uncertain.

Fig. 9 shows measurements of the reactivity change with moderator temperature taken in February 1954. A curve representing a fit to the 1948 measurements is also shown. It can be seen that the moderator temperature coefficients are markedly different, while the only change in the reactor which should affect the result is the use of a 0·071 inch cooling annulus for the rods in this experiment. Although the temperature coefficient of reactivity of the rod assemblies was not measured at this time, on the basis of the above measurements it is expected to be appreciably smaller with the thinner cooling annulus.

A comparison has been made of the overall temperature coefficient of reactivity, the sum of the moderator and rod-assembly temperature coefficients, with that

predicted from the lattice theory (WARD, 1953a). Effects included in the calculations are:

(a) The Eta-Effect—arising from the change in the ratio of the effective cross section of U^{235} to U^{238} with neutron temperature.
(b) The Doppler broadening of the U^{238} resonances.
(c) The change in thermal utilization arising from the flattening of the neutron flux distribution across a lattice cell as the neutron temperature rises.
(d) The changes in L_s^2 and L^2 with temperature.

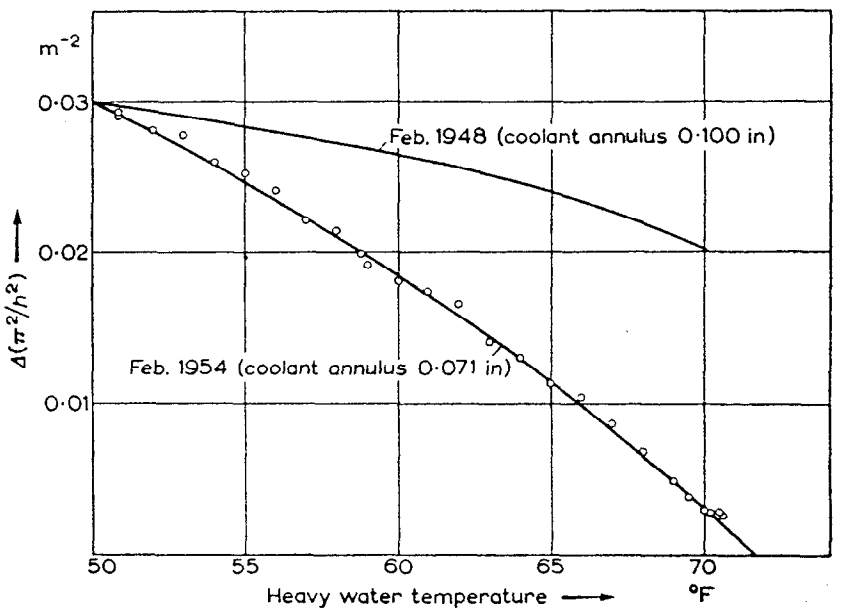

Fig. 9. Change of reactivity with heavy-water temperature.

The calculated temperature coefficient for NRX was

$$10^7 \frac{d}{dT}(\pi^2/h^2) = -2{\cdot}0 + 5{\cdot}9 \times 10^3 \frac{1}{\rho_m}\frac{d\rho_m}{dT} - 1{\cdot}08 \times 10^3 \frac{1}{\rho_w}\frac{d\rho_w}{dT}$$

The experiments give

$$10^7 \frac{d}{dT}(\pi^2/h^2) = -2{\cdot}05 + 5{\cdot}0 \times 10^3 \frac{1}{\rho_m}\frac{d\rho_m}{dT} - 1{\cdot}08 \times 10^3 \frac{1}{\rho_w}\frac{d\rho_w}{dT}$$

where h = critical height (weir-box reading plus 22 cm)
T = temperature (°C)
ρ_m = density of heavy-water moderator
ρ_w = density of light-water coolant

The agreement is much better than is justified by the accuracy of the data used in the calculation. It should be mentioned that the last term in both equations, describing the effect of density changes of the light-water coolant, is semi-empirical in nature, being derived from lattice measurements with and without cooling water. It has been assumed in the analysis of the experimental

measurements. A correction has also been made for the effect of the graphite reflector in calculating the measured reactivity changes with temperature.

Figs. 10 and 11 show the experimental results plotted in a way that enables the density effects to be separated from the effects arising from neutron temperature only. The points plotted were derived from the curves of Figs. 7, 8, and 9, and the differential nature of this plot emphasizes the experimental errors.

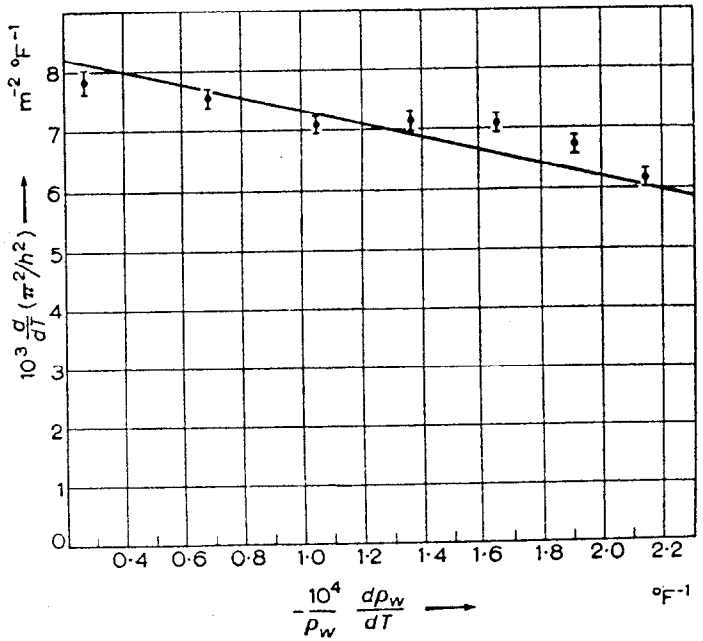

Fig. 10. Temperature coefficient of reactivity for NRX rod assemblies. The slope of the line was calculated from lattice measurements with and without cooling water.

The temperature coefficients of the NRX reactor are small and relatively unimportant in operation. In the design of power reactors, much larger changes of reactivity with temperature are possible, and correct estimates of temperature coefficients of reactivity are required.

Prompt Poison in NRX

The immediate reactivity change accompanying a change of operating power in NRX is called "prompt poison". It is of importance in stabilizing the power. It was measured in 1948 (WARD, 1953b); effects contributing are

(a) The change in mean coolant temperature in the pile with change of power.
(b) The change in Doppler width of the U^{238} resonances, and consequent change in the resonance trap, with change of metal temperature.
(c) Differential expansion of the rod assembly, and the consequent alteration of the thickness of the cooling annulus with the radial temperature gradient.

The measurements were made in the following way: the reactor was operated at a steady power, and then the power was rapidly altered to a different value and held constant. The control rod was normally used to provide the necessary reactivity changes, and the control-rod position and moderator temperature were recorded as functions of time. Results from a typical experiment are shown

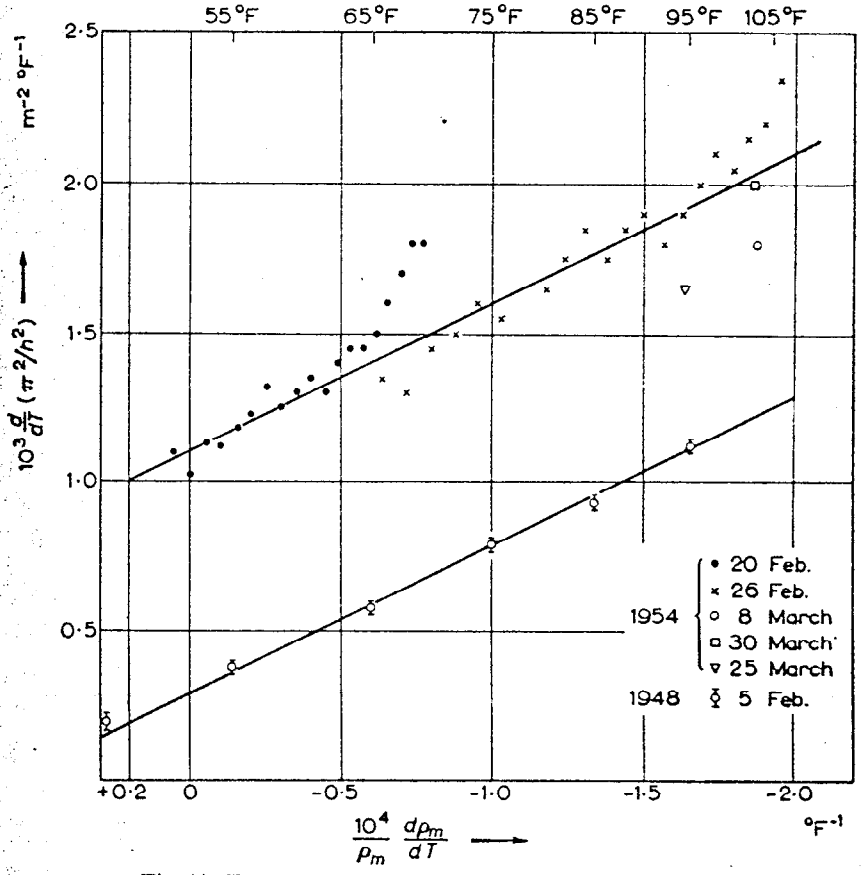

Fig. 11. Temperature coefficient of reactivity for the NRX moderator.

in Fig. 12 (Expt. Feb. 27, 1954). The operating power as a function of time is indicated at the top. The reactivity changes calculated from the control-rod position are shown by the circled points. At the bottom of Fig. 12 are shown the reactivity changes resulting from changes in moderator temperature, delayed neutron sources and changes in xenon poison. The control-rod reactivity changes are corrected for these effects and the remaining change in reactivity with power is indicated by the crosses. The discontinuity at the times when power changes have been made are then interpreted as the changes in reactivity due to the "prompt poison."

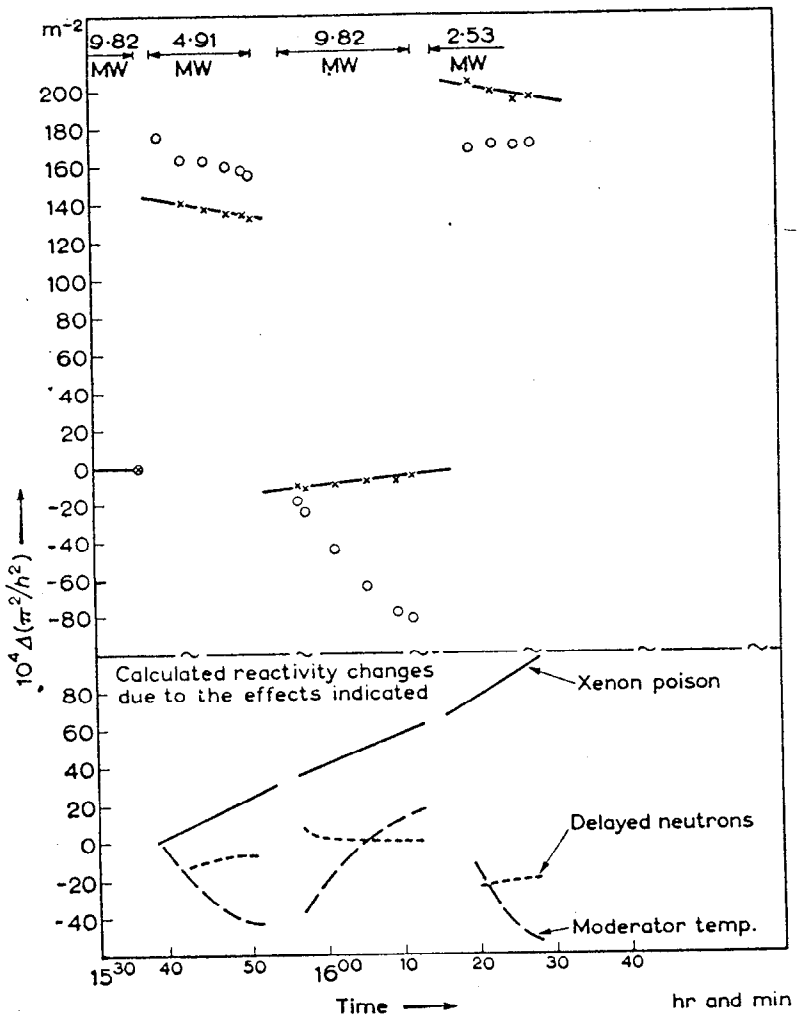

Fig. 12. Prompt poison experiment.

○ reactivity changes as measured by control-rod position
× net reactivity changes after correction for the three effects shown in the lower part of the figure. The discontinuities are attributed to prompt poison.

Fig. 13 shows the collected results of several such measurements plotted against the mean of the powers before and after the change. The results were normalized to a weir-box reading of 250 cm on the assumption that the measured prompt poison per MW is inversely proportional to the weir-box reading. It can be seen that the prompt poison per MW decreases at the higher reactor powers. The total prompt poison at 40 MW corresponds to about 2·2 mk.

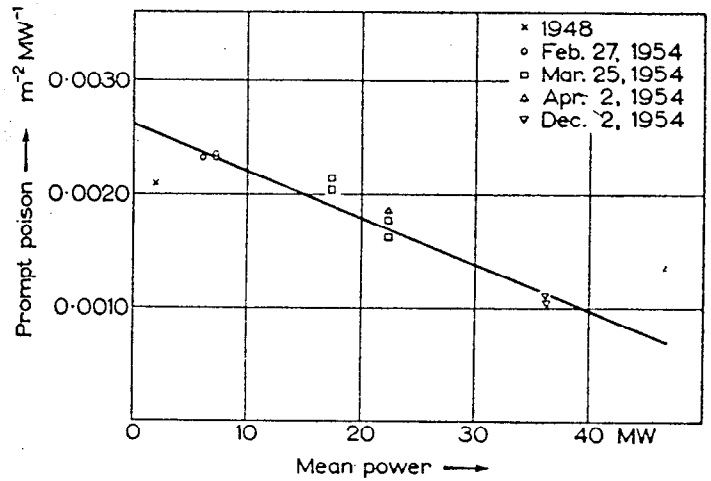

Fig. 13. Collected results of prompt poison experiments normalized to a weir-box setting of 250 cm.

Xenon Poison, Experimental Methods, and Theory.

The behaviour of xenon poison in the NRX reactor has been studied in three types of experiment.

(a) The growth to equilibrium level during operation at power levels ~1 MW. In this experiment the weir-box changes are relatively small and the interpretation of the results straightforward. Similar experiments to observe the growth of xenon poison at higher powers have not been analyzed since the complications in the analysis due to gross changes of weir-box level with poison are formidable.

(b) Measurement of the equilibrium levels of xenon poison at various powers. The precision is not high, since for many of the measurements it was not possible to measure the critical height without xenon poison at closely related times.

(c) Reactivity measurements of the transient behaviour of the xenon poison during a shutdown, subsequent to high power operation under near equilibrium conditions.

Calculations of the reactivity change due to Xe^{135} in the NRX reactor (WARD, 1954) have been based on the usual two-group diffusion theory applicable to lattice reactors (GLASSTONE & EDLUND, 1952). The calculated critical heights are based on a model which includes the graphite reflector on the cylindrical

surface only, and an allowance for the vacant positions assigned to the control rod and shutoff rods. In the 1954 experiments three lattice positions contained no uranium rods but were occupied by relatively weakly absorbing samples for the production of isotopes.

The perturbation calculation of the effect of Xe^{135} on the critical height of NRX has been carried out as if the lattice were complete, but assuming that no heat and no xenon is produced at the empty lattice positions reserved for shutoff rods, or at those occupied by the control rod or weak absorbers.

The calculated change in critical height is given by the equation

$$\left(\frac{\pi^2}{h^2}-\frac{\pi^2}{h_0^2}\right)=\frac{1}{R_1}\times\frac{1}{R_2}\times\left(\frac{k}{L_s^2+L^2+2\mathscr{H}^2L_s^2L^2}\right)\frac{\int\left\{\frac{\Delta k}{k}-\left(\frac{\mathscr{H}^2L^2}{1+\mathscr{H}^2L^2}\right)\frac{\Delta L^2}{L^2}\right\}\phi^2\,dV}{\int\phi^2\,dV}$$

where h = critical height (weir-box reading plus 22 cm)

h_0 = critical height without poison (cm)

$R_1 = 1{\cdot}07$ = reflector effect

$R_2 = 1{\cdot}15$ = correction for "vacant" lattice positions

$L_s^2 = 114\ \text{cm}^2$ = slowing down area

$L^2 = 156\ \text{cm}^2$ = thermal diffusion area

$\mathscr{H}^2 = 4{\cdot}07\times10^{-4}\ \text{cm}^{-2}$ = Laplacian or buckling of lattice

$k=(1+\mathscr{H}^2L_s^2)(1+\mathscr{H}^2L^2)$

ϕ = average thermal flux in the uranium

The integrals are taken over the uranium rods in the pile. $\left(\frac{\Delta k}{k}\right)$ and $\left(\frac{\Delta L^2}{L^2}\right)$ are then calculated from the effects of added absorption cross section in the rods using the normal diffusion theory calculations for the lattice cell.

For the NRX lattice

$$\frac{\Delta k}{k}=-0{\cdot}940\,\frac{\Delta\sigma}{\sigma}$$

$$\frac{\Delta L^2}{L^2}=-0{\cdot}528\,\frac{\Delta\sigma}{\sigma}$$

where $\frac{\Delta\sigma}{\sigma}$ is the fractional change in the absorption cross section of the rod (initially uranium only), due to the presence of the added absorber.

With these values, the relation becomes

$$\Delta(\pi^2/h^2)=\left(\frac{\pi^2}{h^2}-\frac{\pi^2}{h_0^2}\right)=-2{\cdot}89\times10^{-3}\times\frac{\int\frac{\Delta\sigma}{\sigma}\phi^2\,dv}{\int\phi^2\,dv}$$

The quantity $\Delta\sigma/\sigma$ was evaluated as follows:

(a) $\phi(r,t)$ was calculated from the record of operating power, weir-box height and the measured radial flux distribution.

(b) The concentration of Xe^{135} was assumed to be governed by the differential equations

$$\frac{dI}{dt}=K\phi-\lambda_I I$$

$$\frac{dX}{dt}=\lambda_I I-(\lambda_x+\sigma_x\phi)X$$

where I = concentration of I^{135} (atoms/cc)

X = concentration of Xe^{135} (atoms/cc)

t = time

λ_I = radioactive decay constant of I^{135}

λ_x = radioactive decay constant of Xe^{135}

σ_x = thermal neutron cross section of Xe^{135}

$K=N_u\sigma_f(1+\delta)Y$

N_u = atoms/cc of uranium

σ_f = thermal fission cross section of uranium

$(1+\delta)$ = total number of fissions per thermal fission

Y = yield per fission of I^{135}

σ_u = thermal absorption cross section of uranium

The value of $\Delta\sigma/\sigma$ is given by $X\sigma_x/N_u\sigma_u$.

The exact computation of $\Delta(\pi^2/h^2)$ as a function of time would be exceedingly tedious, and has not been carried out. The analysis was restricted to approximate solutions which are valid:

(a) At low power, when the fractional change of heavy-water depth with poison is small.

(b) At near equilibrium conditions at all power levels.

(c) For the xenon transient behaviour, after shutdown from operation at near equilibrium conditions.

The calculated reactivity changes were based on the following data:

$\sigma_f/\sigma_u = 0{\cdot}528$ for natural uranium

$\sigma_f = 4{\cdot}18$ barns

Energy per fission = 198 MeV.

Xenon Poison, Observed Build-up

Fig. 14 shows measurements of the change of xenon poison with time when the reactor was operated at low power and there was no xenon poison in the pile initially. Two experiments of this kind were carried out during start-up experiments. The results were complicated by pile trips before the equilibrium level was reached. Also shown in Fig. 14 are calculated points based on two sets of parameters. It should be emphasized that at this power level, the equilibrium

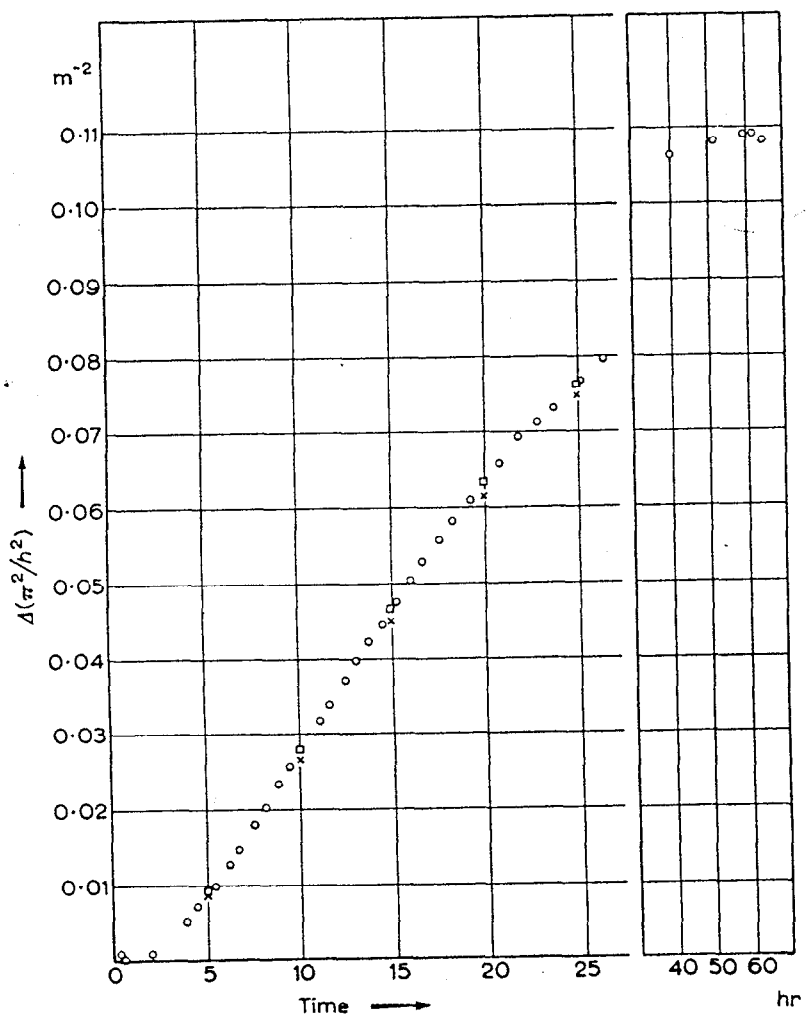

Fig. 14. Growth of Xe^{135} poison at 1·4 megawatts.

○ experimental points
□ calculated for $T_{1/2}(I^{135}) = 6{\cdot}25$ hr.
× calculated for $T_{1/2}(I^{135}) = 6{\cdot}68$ hr.

The calculations assumed that the equilibrium level was $0{\cdot}110\ m^{-2}$ and that burnup reduced the half life of Xe^{135} from 9·20 hr. to $9{\cdot}20 \times 0{\cdot}883$ hr.

xenon poison depends essentially on the product of the xenon yield in fission and the xenon cross section. The calculated time-dependent behaviour depends on the assumed lifetimes of I^{135} and Xe^{135}. At low power the burn-up of Xe^{135} is small and the assumed xenon cross section has relatively little effect on the time dependent behaviour.

In both sets of calculated points no direct production of Xe^{135} from fission has been assumed. The fit of the calculated points to the experimental points is not as good if as little as 3% of the xenon is assumed to be formed directly in fission; this is good experimental evidence that the direct production of Xe^{135} in fission is negligible. It can be seen that the two sets of calculated points both provided a fair fit to the experimental results, but that the set calculated using the value of 6·25 hours for the half life of I^{135} rather than the "accepted" value of 6·68 hours gives a better fit to the experimental data. Recent experiments at Chalk River indicate that the decay scheme may be more complex than assumed and that a somewhat shorter half life may be appropriate.

Measurements of Equilibrium Xenon Poison vs. Operating Power.

Table 3 gives measured values of equilibrium xenon poison at various power levels. For the first four items, the relative power measurements are reliable but there is an uncertainty estimated as 5% in their absolute value. For the remaining entries, the power measurements are reliable (~2%). For items 5 and 6, two values are given to indicate the uncertainty arising from lack of precise knowledge of the unpoisoned critical height. No measurements of this quantity were made at times close to those of the equilibrium poison measurements. The precision of items 7 to 10 is poor for the same reason. Since the 1948 and 1950 results were not very accurate, two measurements, items 11 and 12, were made in February and March 1954, and unpoisoned critical heights were measured at closely related times. These are believed to be reliable.

Table 3—Experimental Results on Equilibrium Xe^{135} Poison

Item No.		Critical Height (WB + 22 cm)	Power (MW)	h/P (metres/MW)	$\Delta(\pi^2/h^2)$ (metres^{-2})
1	1948	250·9	1·4	1·79	0·110
2		258·5	2·8	0·923	0·193
3		268	4·9	0·547	0·293
4		277	7·0	0·396	0·373
5		291	10	0·291	0·480 0·433
6		299	12	0·249	0·531 0·484
7		300·5	10	0·300	0·440
8		312	14	0·223	0·487
9		322	20	0·161	0·516
10	1950	317·2	30	0·106	0·729
11	1954	222	1·16	1·91	0·1035
12	1954	285	40	0·0712	0·712

If the reciprocal equilibrium poison is plotted against the reciprocal power per unit height, the theory predicts that the points should lie within 2% of a straight line. The results plotted in this manner are shown in Fig. 15. The intercept on the ordinate scale is inversely proportional to the yield of Xe^{135} in fission, while the slope of the line is inversely proportional to the product of

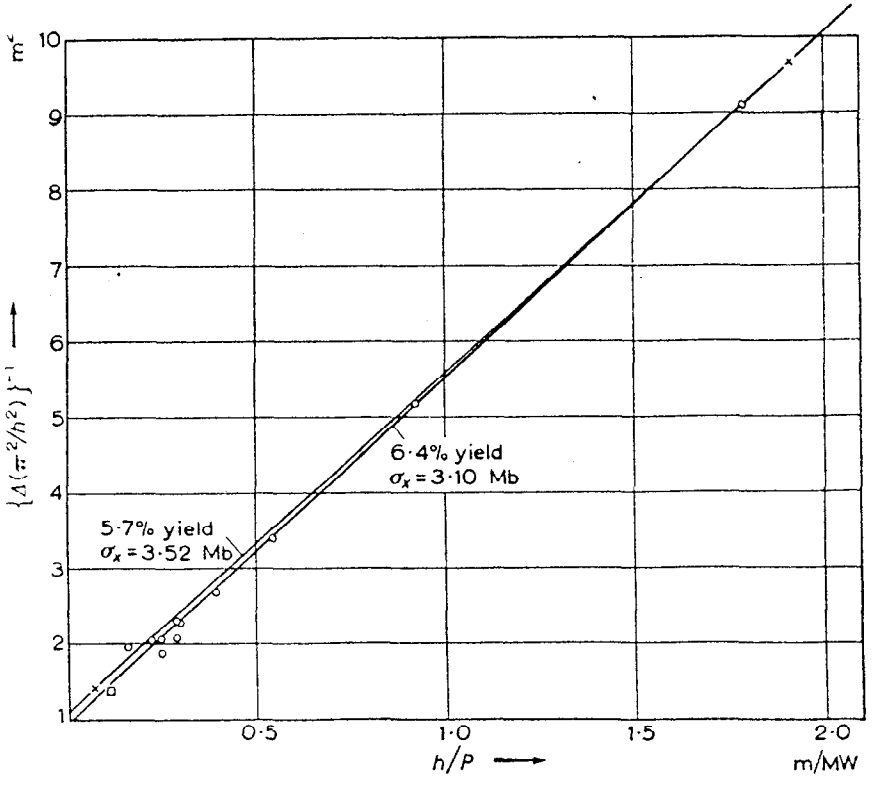

Fig. 15. Equilibrium Xe^{135} poison in NRX
○ 1948 data
□ 1950 data
× 1954 data
The solid lines were calculated for fission yields and xenon cross sections in megabarns as shown.

xenon cross section times xenon yield. The scale factor appearing as a multiplying constant for both the intercept and the slope depends on the precision of our knowledge of the parameters such as L_1^2, L^2 used to calculate the change in (π^2/h^2) resulting from a change in the absorption cross section of the uranium in the reactor. The two straight lines were calculated with differing assumptions as to the effective yield and cross section of Xe^{135}. In this figure the agreement appears good, but the method of plotting does not show clearly the errors of the high power readings.

Fig. 16 is an alternative plot which shows the lack of agreement for the higher power readings. The precision of the 1954 results is believed to be good (±2%), but their interpretation requires an assumed yield of 5·7% and does not agree with the measured yield of 6·4%.

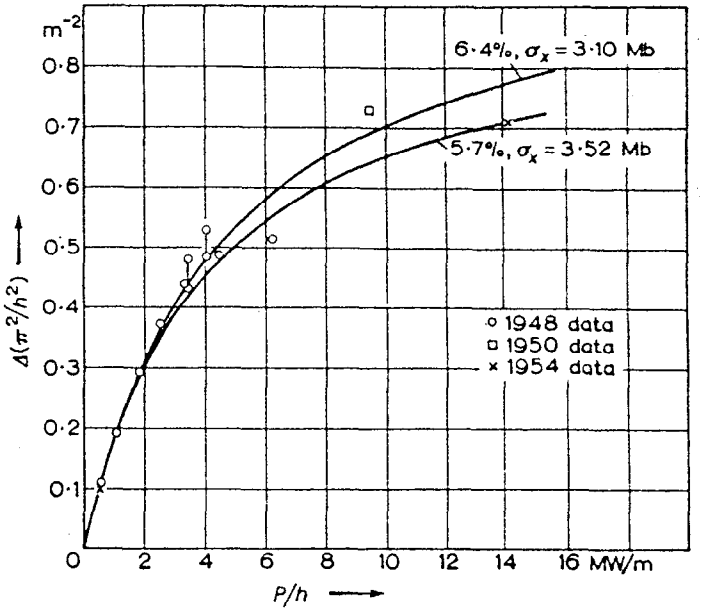

Fig. 16. Equilibrium Xe^{135} poison in NRX
○ 1948 data
□ 1950 data
× 1954 data
(Data of Fig. 15 plotted on reciprocal scales)

Transient Behaviour of Xenon Poison after Shutdown

Experimental results from measurements on the xenon poison transient after shutdown, following near-equilibrium operation at 40 MW, are shown in Fig. 17. The measurements were made using an oscillating standard absorber, as described previously, and with weir-box changes to maintain the reactivity near critical. Throughout this experiment the moderator temperature was kept near 70°F and only minor corrections have been made for temperature changes.

When the weir-box is relatively far from critical, the precision of the measurements is debatable since large corrections have been made for interference between the oscillating absorber and the xenon poison, and for the fact that the ion chambers measure neutron leakage current. When the weir-box is near critical, at the times indicated by circles, and when the xenon poison has nearly disappeared, the precision is good.

Since the fission yield and cross section of Xe^{135} can be considered as fixed by the equilibrium poison experiments, the only parameters adjustable to fit these

results are the half-lives of I^{135} and Xe^{135}. The analysis of the experiment has not been completed. Preliminary analysis has shown that the results are best fitted by assuming a half life of I^{135} appreciably smaller than the accepted value of 6·68 hrs. The shorter half life of I^{135} was also required to fit the build-up of Xe^{135} poison discussed previously.

Extrapolation to the time of shutdown gives a value for the Xe^{135} poison present at shutdown in good agreement with that calculated from the operating

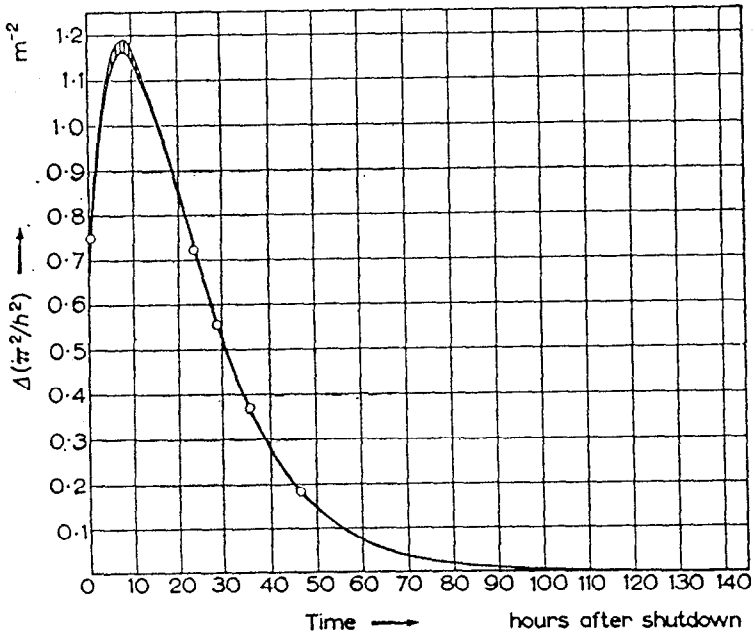

Fig. 17. Poison transient after shutdown from near equilibrium at 40 MW. The double curve indicates the range of experimental uncertainty. ⊙— points for which the precision of measurement is high.

critical height with corrections for prompt poison and moderator temperature. The value of the poison present at shutdown, corrected for the fact that the Xe^{135} is not quite the equilibrium value, has been used to obtain the equilibrium Xe^{135} poison at 40 MW given in Table 3.

Change of Reactivity with Operation.

The loss of reactivity during the first 1000 MW days output of a charge of uranium was studied in 1954 as part of the experimental programme. The reactor was operated under carefully controlled conditions and the loading for isotope production was constant and very nearly symmetrically arranged. The time-table included a 5 day run at 1·16 MW, 25 days at 30 MW, and 12 days at 40 MW (all with minor interruptions), followed by 6 days of subcritical measurements. During the high power operation the principal reactivity effects apart from xenon poison are expected to arise from the growth of Sm^{149}, the

burn-up of U^{235} and its partial replacement by Pu^{239}, and possibly a changing light water content due to deformation of the fuel. This last effect is believed to be small. The net reactivity, expressed as equivalent π^2/h^2 in metres^{-2} is plotted in Fig. 18 versus total energy output. It is the sum of π^2/(Weir-box setting + 22 cm^2) plus corrections for control rod insertion, D_2O temperature variation, and xenon poison. The xenon poison was calculated with the parameters discussed above. Because of uncertainties in calculating critical height when the

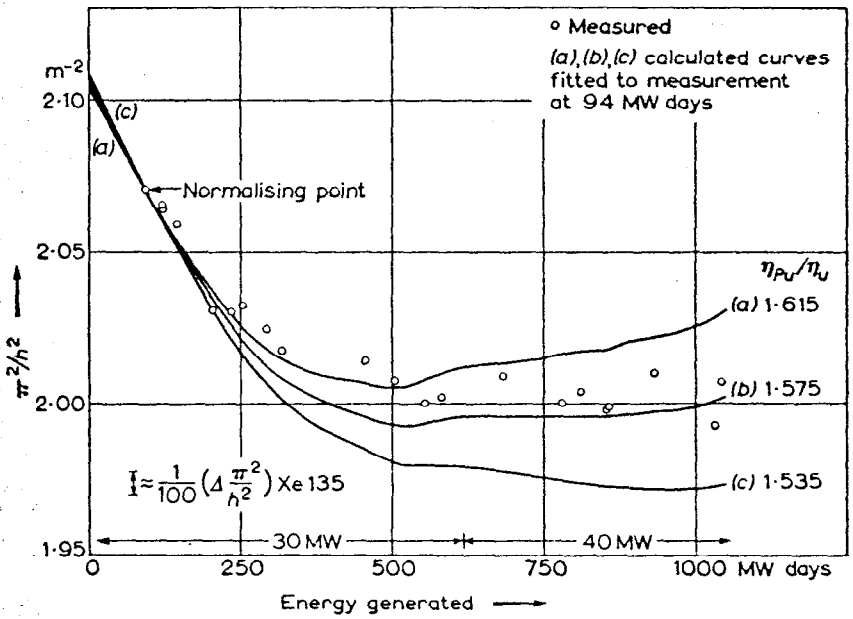

Fig 18. Variation of the reactivity of NRX with total energy output from a charge of uranium. The points have been corrected for changes in moderator temperature, control-rod position, prompt poison, and xenon poison.

poison is changing rapidly, only cases where the xenon was not far from equilibrium are plotted in Fig. 18 and the points have not yet been carried back to the beginning. Another effect occurred which prevents the overall reactivity change from initial to final low power operation being ascribed solely to the effects listed. This was observed as an increase of the flux in the outer part of the calandria relative to the flux near the centre during the first 3 to 4 hours at high power and has been attributed to improved reflecting properties of the graphite, possibly drying out, as a result of heating. There was a concomitant improvement in reactivity. A similar effect, but of smaller magnitude (3 to 4% instead of about 13% flux increase), is observed on each start-up following a shutdown of many hours duration, and is believed to be due to an increased scattering relative to capture with increasing graphite temperature.

Three calculated curves are shown fitted to the point at 94 MWD. The curves are all based on a 1·1% yield of Sm^{149}, a conversion factor of 0·8, and a plutonium cross section of 1125 barns. They differ in the η assumed for Pu^{239}.

For an η of natural uranium equal to 1·3 the curves (a) (b) (c) correspond to η Pu) equal to 2·10, 2·05, 2·00 respectively.

Fig. 19 shows the separate contributions from Sm^{149}, U^{235} and Pu^{239} to the calculated curves of Fig. 18. The details of reactor operation have been taken into account and produce transient effects in samarium and in the three plutonium curves. In addition to its dependence on η the plutonium contribution is proportional to the product of cross section and conversion factor; a lower conversion factor than the assumed 0·8 could be compensated by a larger plutonium cross section. The early loss of reactivity is due largely to the burnout of U^{235}, the plutonium and samarium balancing each other.

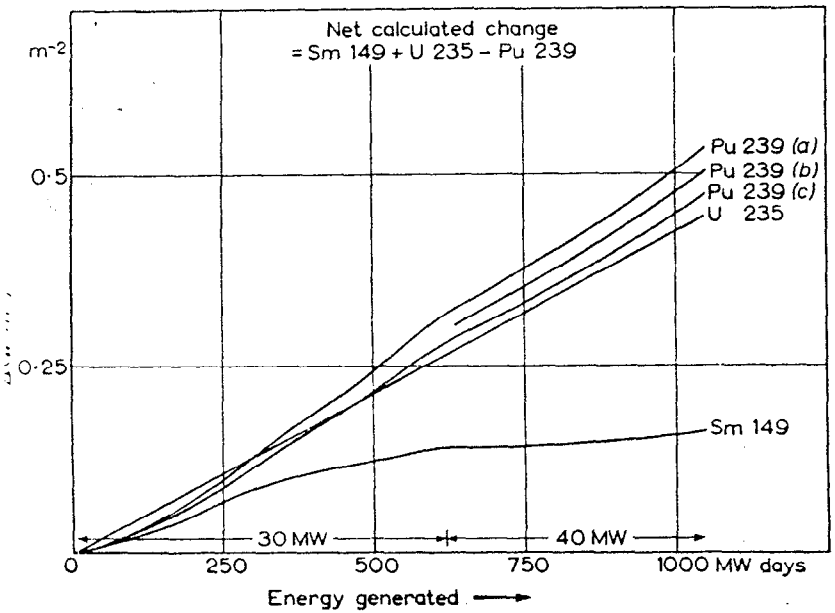

Fig. 19. Components of calculated reactivity change used in Fig. 18.

These experiments provide results for checking the consistency of a set of parameters but do not select a unique set. They are of interest as they have brought to light no unexpected large reactivity losses and the discrepancy between calculation and measurement appears to be in the direction of a smaller decrease of reactivity with irradiation than would be predicted from commonly used values of the parameters, a desirable result for power producing reactors where long irradiation is required.

8. FUEL ELEMENT TESTING IN NRX

A continuing study of the behaviour of the NRX rods has proved very fruitful. Some rods of the first charge of uranium were taken to an irradiation of 4000 MWD/tonne showing clearly the potentially low cost of nuclear fuel. Considerable effort has been expended on a programme of NRX rod development to

identify the various factors which contribute to satisfactory performance under irradiation. Studies of reactivity change with irradiation of NRX rods have given confidence in the data used to predict such effects.

An extensive programme to test fuel element designs for future reactors is being carried out.

Loops

The development of components, particularly fuel elements, for power reactors requires testing under conditions expected in service. To provide these conditions it is often necessary to test the component in a closed circuit or "loop" where flow, temperature, corrosion, gas production and fuel changes can be observed. Each loop has its own system of pumps and coolers which is separate from the reactor system and is often at high pressure. Out-of-pile equipment is complex and occupies much more space than the in-pile section. The provision of shielding for this auxiliary equipment is a major problem. The shielded regions originally available with NRX were the pit in front of the south thermal column and some space inside the upper concrete shield. In 1952 a concrete structure containing two shielded rooms was built in the reactor hall. Further space has recently been made available by moving the helium gasholder from the large concrete-lined pit it had occupied to a location outside the building.

When installed, a loop must be treated as an integral part of the reactor for both operation and safety. Loop failure will seriously reduce reactor operating time. Paring the cost of loop instrumentation can be false economy if it results in reactor damage or unnecessary shutdown, and careful planning is well worthwhile.

The advantages of NRX for loop work are high flux, access to relatively large volumes in the high flux, and considerable excess reactivity, which are not all available in any other research reactor. The reactivity is required because test pieces are often enclosed in stainless steel jackets and these are strong neutron absorbers. Although outclassed now in flux, NRX still retains an important place in development work because the higher flux reactors are restricted in volume.

Many fuel elements have been developed with the aid of NRX. In particular NRU fuel rods have been tested under operating conditions in a loop in the central thimble, and the fuel for NPD (the power demonstration reactor now being designed) will also be tested; one cannot rely on a repetition of the good fortune that resulted from the choice of NRX rods at a time when there were no operational tests and little information.

Fuel Reactivity

Along with interest in the mechanical stability of uranium there has been the need to know the change of reactivity with irradiation. NRX has played a dual role, both as producer of highly irradiated material and as the measuring instrument.

The latest measurements of the loss in reactivity of individual rods were carried out in 1952. Several ranges of irradiation were included, provisionally chosen as (a) 3000 (b) 1800 (c) 900 and (d) 250 megawatt days per tonne maximum

in the rod. The actual values were determined by the material on hand. Group (a) represented the maximum then available, (c) lies near the predicted maximum reactivity when fission products are ignored, (d) is in the region where stable fission products of large cross sections, e.g. Sm^{149}, are near equilibrium, and (b) is a suitable intermediate value. Six rods of group (a), five rods of group (b), and three rods of each group (c) and (d) were measured by comparison with unirradiated rods from the same batch of uranium having the same style of aluminium sheath. Lattice positions L-15, H-12, and H-18 (Fig. 4) were used.

The rods under test were air-cooled to eliminate uncertainties in the volume of light water arising from possible dimensional changes of the uranium. Six shutoff rods were left in the reactor at C-9, C-21, J-27, Q-21, Q-9 and J-3 to simulate xenon poison and to raise the critical height to the usual operating level. In this way the measuring flux was matched to the irradiation flux. The advantages of this condition were considered to outweigh the distortion of radial flux by the shutoff rods. All lattice positions inside the ring of lowered shutoff rods were occupied by uranium rods except for the control rod at J-5 and empty tubes at F-12, M-18, and F-18 (shutoff rods and a special control rod). Measurements were begun 14 days after the pile was shut down. Xenon had by then decayed to a negligible amount and Pu^{239} and Sm^{149} had reached equilibrium.

The standard reactivity change was the effect of the water in the rod assemblies at G-15, K-12 and K-18. These positions are in the same circle as the test positions and are symmetrically located with respect to the thermal column containing the ion chamber used for the measurements. The standard was calibrated by measurements at two neighbouring weir settings. The calibration was carried out at both the high weir (distorted pile) and at low weir (undistorted, all shutoff rods out). The two values obtained are

$$\Delta(\pi^2/h^2) \text{ distorted pile} = 0{\cdot}0414 \text{ metres}^{-2}$$

$$\Delta(\pi^2/h^2) \text{ undistorted pile} = 0{\cdot}0327 \text{ metres}^{-2}$$

The results for the irradiated rods are shown in Table 4 where the irradiation is given in total megawatt days generated by the rod. The maximum irradiation in the rod in MW days/tonne is roughly 30 times this quantity.

Errors are difficult to estimate because many factors are involved. Keeping in mind that the measured effects were in all but one case three times those in the third column of Table 4, these figures may be compared with the magnitude of certain changes required to produce 0·001 metres^{-2}, for example 2°F in moderator temperature, 0·15 cm in depth, 1·3°F in the temperature of the normal rod assemblies. None of these factors is believed to have caused errors of this magnitude. The temperature of the air-cooled irradiated rods is not known but measurements with different air flow rates suggest a maximum error of $0{\cdot}25 \times 10^{-3}$ metres^{-2} per rod from this source.

The net reactivity loss includes losses due to burnup of U^{235}, absorption by fission products, Pu^{240}, etc., and gains due to production of Pu^{239} and Pu^{241}. The net loss is sensitive to cross sections, neutron yields, conversion factor, and the characteristics of the lattice.

The change in $\Delta(\pi^2/h^2)$, due to replacing an unirradiated rod by an irradiated rod at the test position in the undistorted pile, has been calculated (WARD, 1952)

Table 4—Reactivity Losses due to Irradiation

Irradiation Megawatt days per Rod		Measured Reactivity Loss per Rod (Based on 0·0327 metres^{-2} for the standard)	Calculated Reactivity Loss Metres^{-2}		
Individual Rods	Average	Metres^{-2}	Case 1	Case 2	Case 3
9·1					
10·8	10·1	0·0013	0·0026	0·0018	0·0007
10·5					
28·8					
29·6	31·2	0·0013	0·0031	0·0014	−0·0004
35·1					
61·4					
60·3	60·8	0·0031	0·0056	0·0029	−0·0001
—					
67·9					
67·9	67·9	0·0040	0·0065	0·0036	0·0003
67·9					
107·4					
100·8	103 .	0·0072	0·012	0·0084	0·0038
100·8					
107·4					
107·4	107·4	0·0078	0·013	0·0092	0·0045
107·4					

using the perturbation method outlined previously. The statistical weight of the lattice site used as a test position was calculated to be $9{\cdot}4 \times 10^{-3}$ for a normal rod with cooling water. This has been increased by 20% to allow for the increased flux in the uranium when the cooling water is absent. The following relations, which apply to a lattice without cooling water, have been used.

$$\frac{\Delta L^2}{L^2} = -0{\cdot}64\,\frac{\Delta\sigma}{\sigma}$$

$$\frac{\Delta k}{k} = \frac{\Delta\eta}{\eta} + 0{\cdot}036\,\frac{\Delta\sigma}{\sigma}$$

giving the equation

$$\Delta(\pi^2/h^2) = 0{\cdot}40\,\frac{\int(\Delta\eta/\eta + 0{\cdot}077\Delta\sigma/\sigma)\phi^2\,dz}{\int\phi^2\,dz} \text{ metres}^{-2}$$

where the integrals are to be evaluated along the length of the rod in the test position.

Values to be inserted for $\Delta\sigma/\sigma$ and $\Delta\eta/\eta$ are the averages across the rod weighted according to the flux distribution inside the rod. The integrations were

carried out on the assumption that $\Delta\sigma/\sigma$ and $\Delta\eta/\eta$ can be represented by quadratic functions of the irradiation. Tables from which $\Delta\sigma/\sigma$ and $\Delta\eta/\eta$ can be derived as functions of irradiation were available (LEWIS, 1952, 1953a). The constants of Table 5 formed the basis of the calculation which also included

Table 5—Constants for Reactivity Calculations

Nuclide	σ_f (barns)	σ_a (barns)	η		
			Case 1	Case 2	Case 3
U 235	545	645	2·112	2·112	2·112
Pu 239	730	1060	2·00	2·05	2·10
Pu 241	1000	1400	2·35	2·35	2·35
Pu 240		600			
U 236		25			

Conversion factor (Pu^{239} atoms produced per U^{235} atom destroyed) = 0·8.

fission product absorption. It is not possible to get good fits with quadratic expressions and a compromise favouring the long irradiations was made. The resulting calculated reactivity losses are shown in Table 4. These preliminary calculations are presented to show the order of agreement with experiment, and they make evident the effect of small changes in η of Pu^{239}. No conclusions have yet been drawn from these results regarding the correct values of parameters. The conversion factor of NRX is 0·77. It is clear from the agreement of the calculated effects in Case 2 with the measurements that a conversion factor of 0·73 will require changes in the assumed data, for instance, a larger value of η or a larger cross section for Pu^{239}.

9. UNUSUAL OCCURRENCES

General

Many unusual occurrences have marked the history of NRX operation from the first timid approach to critical, through the low power trials to routine operation at powers which have increased from 10 MW to the present 40 MW. Some of the improvements in equipment and operating procedures as a result of accumulated experience have already been discussed. Four unusual occurrences posed difficult problems in repair. Of these the accident of December 1952 is widely known and well documented. The other three, although they entailed shutdown for many days were not of comparable magnitude.

Stuck Rods (Wrinkled Rods)

In October 1949 a rod which required unusual force during removal from the reactor was found to have a compression-type wrinkle, or corrugation, around the outside of the outer sheath. The wrinkle had caught in the upper thermal shield where the hole is 1/8 in. smaller in diameter than the calandria tube. Two weeks later another rod was found to be jammed. A similar wrinkle, large enough to clamp the rod in the calandria, was suspected. The overall length of the rod had decreased while in the reactor. Inspection of the rods in the reactor,

and testing of short rods, showed that four rods were jammed and that three others moved freely for some distance and then stuck, presumably in the upper shield. While the rods stopped by the upper shield could be handled roughly, all operations on the rods jammed in the calandria were done under carefully controlled conditions to prevent undue force being applied to the calandria tube. The calandria tubes are 1/16 in. thick and are sealed into the calandria by rolled joints. It was estimated that not more than 4000 pounds force should be applied to these tubes. After considerable study and testing the jammed rods were freed by slipping an annular saw over the outer sheath and sawing through the corrugation. The top of the corrugation was left as a ring and was easily pulled out after the rod had been removed. Before the saw could be inserted through the shield the upper shielding section of the rod assembly had to be removed. This section consists of an outer tube and an inner plug with the cooling water annulus between. The outer tube was pulled free of the sheath with a "wheel-puller" acting on the inner plug. The plug was then twisted loose, leaving nothing above the uranium section to interfere with the saw.

Examination of the calandria tubes after removal of the stuck rods showed no visible damage other than surface scratches.

The Fusing of Uranium in a Converter

In November 1952 the cooling water for a hollow uranium cylinder (neutron converter) in one of the 4 in. experimental holes was by-passed in such a manner that the safety devices did not operate. The uranium and its aluminium sheathing fused to the aluminium liner of the hole, releasing fission products into the graphite reflector. The activity carried away in the cooling air gave the first indication of trouble.

Removal of the fused mass was accomplished with tools inserted down the length of the experimental hole and through J-rod holes. The reflector was cleaned satisfactorily and the reactor was back in operation within two weeks. The drilling and reaming tools were heavily contaminated with an intensely active dust. Preventing the spread of this dust was one of the important parts of the operation.

The Accident of December 1952

A major accident to NRX occurred on 12 December, 1952, causing extensive internal damage. The radioactivity dispersed from ruptured uranium rods was very great and interfered at every step with the work of restoration. Some of the problems of decontamination and reconstruction have been described (GRAY, 1953; GILBERT, 1954; HATFIELD, 1955) and the course of the accident has been discussed (LEWIS, 1953b; HURST, 1953).

The accident resulted from a combination of mechanical failures, errors, and abnormal operating conditions. Preparations were under way for reactivity measurements with the reactor unpoisoned. The heavy water was at the normal operating level and to compensate for the lack of xenon poison seven shutoff rods had to be down at all times. From a condition with all 12 shutoff rods down (the shutoff rod complement at that time) some were inadvertently raised by the opening of air valves, and unknown to the operating crew, failed to drop after the valve settings were corrected. Consequently as the rods were

being raised a few minutes later to start the reactor it unexpectedly went over critical. Normally the shutoff rods could have been dropped back immediately, but through lack of accelerating air in the heads and other faults some stuck on the way down. The power rose, and might have settled out at a safe level if cooling water had not boiled out of some rod assemblies in which the flow had been reduced for the purposes of the test. The extra reactivity due to loss of water resulted in a surge of power to a high level (~80 MW). The reactivity change resulting from boiling was limited since water re-enters a rod from both ends as soon as the outer sheath has been punctured. The heavy water was pumped to storage tanks and the power fell. About thirteen calandria tubes were punctured and in several a portion near the middle was completely destroyed. A flood of water poured into the basement from the burst rods but the pumping and the in-rush of light water to the calandria reduced the loss of heavy water. Over ninety percent of the heavy water was recovered although somewhat degraded by admixture of light water.

The flooding of the basement produced one of the most serious problems. Water, which was intensely active, accumulated steadily at a rate up to 100,000 gallons per day on account of the requirement for continued cooling of the reactor core. Provision had to be made for its safe disposal. After the basement was drained, the walls, the instruments, and everything else which had been immersed were left thoroughly contaminated. The concrete, owing to its porosity, had absorbed the activity deeply. The whole building had been contaminated by gaseous fission products.

Some weeks were required for unloading the undamaged rods, making preparations for removing the damaged rods, and rough clean-up operations. Another interval of a few weeks was spent on cutting out the damaged rods and further clean-up. Then the calandria was removed and the final decontamination and restoration could be started. The reactor was back in operation in February 1954.

Several items may be put on the profit side to mitigate the considerable cost of the accident. The mere fact of its occurrence gave the event some value as showing that the worst reactor runaway so far recorded could be handled in safety. (On the other hand, general news of the accident appears to have caused decisions to be taken in reactor programmes at other establishments which were unnecessarily restrictive had all the facts been known.) Another intangible profit was the confidence and experience acquired. Problems which would have been considered almost insuperable before this accident are now considered difficult and annoying. Techniques were developed for decontamination, and improved designs for reactor structures and disposal areas were evolved.

The dismantling allowed modification of the thermal shields. The modifications have permitted operation at 40 MW and have transferred the power limitation from the shields to the uranium rods. Other improvements were made in NRX and lessons learned were applied in the construction of NRU.

The Failure of a Plutonium-Aluminium Alloy Rod.

In July, 1955, a plutonium-aluminium alloy rod being used to provide reactivity failed without warning, melting through the calandria tube and injecting some of the alloy into the heavy water. The rod had been in use for

only 16 hours so that the contamination was much less than it would have been from a long-irradiated rod. After the rod was removed the often-discussed process of replacing a calandria tube was successfully undertaken. It was necessary to bore out the upper shields to give sufficient diameter for removal and insertion, and to roll the new tube into the upper and lower tube sheets some ten and twenty feet below the access level.

Before the new tube was inserted an inspection revealed many flakes of alloy on the lower tube sheet of the calandria. Vacuuming removed some of the visible flakes. As only a small area of the tube sheet could be seen there were probably many more flakes out of sight and inaccessible to the suction hose. Flushing with water failed to remove the remaining flakes. All the flakes in the field of view were finally removed by treatment with oleic acid followed by flotation in soda water. The soda water was drained out of the hole in the tube sheet through a filter. This complete removal of the visible flakes gave good reason to hope that the whole tube sheet was left reasonably free from debris except perhaps in a groove around the circumference.

In spite of these efforts the alloy material accounted for at the end of the repair was less than the amount known to have been in the rod. There was considerable worry in case the missing plutonium, presumably in the reactor structure, should produce an unacceptably high activity in the heavy-water and helium systems. Fortunately this has not happened.

The plutonium alloy when in the rod was strongly self-shielded and its dispersal increased the reactivity, causing a sudden increase of power (40 to 50%) for one or two seconds. This emphasizes the need of fast shutoff capacity, particularly in any conditions where such dispersal is a possibility.

10. RESEARCH USE OF THE NRX REACTOR

Research in physics and chemistry at the NRX reactor was begun immediately after the reactor was brought to appreciable power. The main subjects of research have been neutron-capture gamma rays, the fission process, the decay of the neutron, radiation damage, cross sections, radiation chemistry and neutron diffraction. A short review will be given without extensive references. Complete lists of publications are available on request from A.E.C.L. In general only work done at the reactor has been included in the following discussion. This criterion excludes biological and chemical work done with tracers and with pile-produced sources of radiation.

Neutron-Capture Gamma Rays

One of the most fruitful fields has been the measurement of energies and intensities of gamma rays emitted in the de-excitation of the compound nuclei formed by neutron capture. The use of a pair spectrometer has permitted the high-energy components to be measured with precision. A comprehensive study covering most of the periodic table has yielded results interpreted in terms of nuclear shell structure.

Earlier, work with a thin lens magnetic spectrometer led to a revision of the mass of the neutron. The lifetime of the excited state of Li^7 formed in the breakup of the compound nucleus B^{11} was deduced from the variation in width of the

gamma ray line as a function of slowing-down time of the recoiling Li^7 nucleus. The result was $0{\cdot}75 \pm 0{\cdot}25 \times 10^{-13}$ sec.

The Fission Process

The ionization produced by each member of fission fragment pairs has been studied in detail. A separate investigation of the charged fragments showed that in a small fraction of fissions a third charged particle is emitted. Angular correlation of the fission fragments and prompt neutrons has been measured.

Decay of the Neutron

The beta decay of the neutron has been established. The life-time has been shown to be $12{\cdot}8 \pm 2{\cdot}5$ minutes and the beta spectrum to be simple. The angular correlation between the directions of the beta particle and proton is being studied.

Radiation Damage

The transfer of kinetic energy from the neutron to the atoms in certain solids alters the macroscopic properties. The irradiations are done inside neutron converters where the fast flux is high. Substances investigated where changes are especially marked are graphite and the order-disorder alloys Cu_3Au and CuAu.

Cross Sections

Many cross sections have been measured. A high flux was essential for some of these researches, for example, the determination of capture cross sections of unstable nuclides (Ba^{139}, Co^{60*} and Au^{198}), the measurement of the fission cross section of Pu^{239} and U^{233} as a function of energy with a crystal spectrometer, and the study of resonant scattering from Sm, Gd, Cd, In, and Rh.

Radiation Chemistry

A clear understanding of the effects of irradiation on the decomposition and corrosion properties of water is of great importance in water-moderated power reactors. Corrosion by water at high temperatures and pressures and its interaction with fuel materials are being investigated under reactor conditions in loops and autoclaves. During the early operation of NRX it was first clearly demonstrated that radiation decomposition of the heavy-water moderator can be made negligible by control of purity and pH with ion-exchange resins (ROBERTSON, 1955).

Neutron Diffraction

Monoenergetic neutrons, selected by crystal reflection from the neutron spectrum leaving an experimental hole, have been used in studies of diffraction by gases, liquids, and solids. In particular the structures of several magnetic solids have been determined.

A study of the exchange of energy between slow neutrons and the normal modes of liquids and solids is expected to give much information about the frequency spectrum and other properties of the scatterer. Spectral analysis of the scattered neutrons is being done with a second crystal and is yielding interesting results.

Programme on a Highly Irradiated Rod

An extensive series of measurements has been made in a study of a highly-irradiated rod in which the metal near the centre received an irradiation slightly above 3000 MWD/tonne. Slugs cut from this rod, each 12 inches long, have been tested for reactivity change (SPINRAD, CARTER, and EGGLER, 1955). Discs were cut from the rod between each of the slugs, and subjected to chemical and metallurgical analyses. The radial distribution of plutonium isotopes has been measured for several of these discs. Measurements of the depletion of the U^{235} content of the discs are being made. It is expected that careful study of the data will help to clarify our understanding of the changes consequent on long irradiation of uranium.

11. ZEEP

General

The name "ZEEP" is taken from the initial letters of "Zero Energy Experimental Pile". Zeep is a natural uranium, heavy-water-moderated reactor with a graphite reflector and no provision for cooling. The first reactor built outside the U.S.A., it began operation Sept. 5, 1945. The objectives in building Zeep were to gain experience in the operation of a heavy-water reactor, to provide a good neutron source for experiments in neutron physics, and to provide a facility for the study of the reactivity of various lattice arrangements.

With the present shielding, the operating power is restricted by considerations of the permissible health tolerance. Eight hour a day operation is limited to 3·5 watts, but operation at 30 watts or higher for short intervals is possible. The maximum thermal neutron flux with a normal lattice is about 10^7 $n/cm^2/sec/watt$.

The Zeep reactor has proved to be a versatile and valuable tool for many studies in reactor and neutron physics. The relatively poor shielding has limited the power to a level below that desired for some experiments. On the other hand the low integrated power has enabled the fuel elements to be easily handled without any serious radiation problems. The steady demand for the use of ZEEP experimental facilities will ensure the continuing operation of this reactor.

Design and Construction

The design and construction of Zeep, from its conception in the summer of 1944, to criticality in September 1945, were dictated by the desire to build a low-energy reactor as quickly as possible. The cylindrical aluminium tank is 6·75 ft. in diameter and 8·5 ft. in depth, with a capacity of about 10 tons of heavy water. It is suspended from a shallow square box of structural steel which is supported on four steel pillars, the tank flange resting on a rubber gasket around the hole in the base of the square box. Mechanical controls for the operation of control and shutoff elements are housed in the square box, along the sides. A number of parallel steel beams bridge the top of the tank, the ends being supported on two steel I-beams. The uranium rods are suspended between the parallel bridge beams by gimbals which allow the rods to take up vertical positions. Variation of the spacing of the rods and of the spacing of the bridge

beams is used to alter the lattice spacing and arrangement. The uranium metal for standard Zeep fuel elements is in the form of short cylindrical rods, 1·28-in. diameter and 6-in. long. They are loaded into aluminium tubes, 1·375-in. outside diameter and 0·040-in. wall thickness; the lower ends of the tubes are sealed with welded plugs. The number of slugs is chosen to make up the desired length of uranium rod.

The reactor has a graphite reflector 3 ft. thick at the side and approximately 2·5 ft. thick below the tank. Water tanks surround the reactor, providing a shield thickness of approximately 3 ft. Passageways 3 ft. wide have been left between the graphite assembly and the water tanks.

A small basement under the reactor contains a stainless steel storage tank, the plumbing system, and equipment for drying the reactor vessel with hot air. The water vapour is recovered in cold traps.

The shutoff rods are steel tubes, 12 in. long, which are plated with cadmium of minimum thickness 0·010 inch and sealed between two aluminium tubes. They are arranged in two groups, or banks, with four rods in each bank. The four rods in a bank are suspended from steel wires wound on pulleys attached to a common shaft on the underside of a beam across the top of the tank. To raise or lower the shutoff rods the shaft is turned by a mechanical linkage operated from the control desk. For quick shutdown action electromagnetic clutches release the shutoff-rod banks, and electromagnetic braking is applied when the rods are near the end of their travel.

Four control plates are suspended by wires in the gap between the tank and the graphite reflector. Three are operated together for coarse control, and the fourth independently for fine control. The plates are made of cadmium-plated steel, slightly curved to conform to the curvature of the tank. The three plates are 20 in. × 12 in., the fine control plate is 20 in. × 4·5 in. The plates are moved by mechanical linkage from hand-wheels at the control desk, with high and low speed selsyns for position indication.

A reversible stainless steel gear-pump is used to raise or lower the heavy-water level in the reactor. The maximum pumping rate is 6 IGPM corresponding to 1·2 cm/min., or 1·2 mk/min. with a lattice near optimum spacing. A manually operated dump valve is located near the control desk.

A manometer and cathetometer were provided in the original installation for measurement of the heavy-water level. An electrically operated probe has been added to improve the precision of measurement of small changes of level.

Boron trifluoride ionization chambers are placed at several points in the graphite reflector to measure the reactor power and to provide signals for the automatic protective circuits. At very low power, including the shutdown condition, the neutron flux is monitored with a boron trifluoride counter.

Experimental Programme

From September 1945 until NRX commenced operation in July 1947, Zeep was the focus for work on neutron and reactor physics. Preparations for NRX operation included ion-chamber tests, irradiation of water to give data for predicting long-lived activities in the NRX coolant, and a test of a sheath-failure monitor. The "swing" or "pile oscillator" method was used for testing samples of graphite and uranium and also for measuring η of U^{233}. Throughout

this interval the lattice arrangement was unchanged. The Laplacian of the lattice was found from the flux distribution, and analysis of the reactor dynamics gave photo-neutron periods. This phase of Zeep operation ended with the transfer of the heavy water to NRX.

In 1950 additional supplies of heavy water became available and Zeep was recommissioned. The prime objective was the study of the reactivity of various lattice arrangements, one purpose for which Zeep was originally designed. Interest was centred at first on the choice of a suitable lattice for NRU. The study of various lattices has been of major interest up to the present time.

Recently the Zeep programme has included the study of enriched lattices, the enrichment being introduced in the form of plutonium-aluminium alloy elements. Studies are also being made of the behaviour of close packed lattices of uranium and heavy water. These experiments are part of a programme for studying non-uniform lattice arrangements and their possible use in the design of power reactors.

Lattice Measurements in Zeep

The following table contains some results of experiments with Zeep rods (1·28-in. diameter) in an hexagonal lattice.

Table 6

Spacing (inches)	Moderator/ Metal Ratio	$\mathscr{H}^2$ (metres^{-2}) (20°C)	G_0	f_u	$f_m \times 10^2$	$f_{al} \times 10^2$	L^2 (cm^2) (20°C)
4·75	14·04	7·37	(1·21)	0·9893	0·484	0·582	64·0
5·50	19·21	8·18	1·225	0·9872	0·687	0·593	89·4
6·81	30·07	7·88					
7·44	36·10	7·32					
7·75	39·28	6·98	1·22	0·9796	1·451	0·585	187·0
8·10	43·01	6·83	1·22	0·9772	1·689	0·589	210·0
8·66	49·33	6·22	1·245	0·9739	2·022	0·592	255·0

where

$\mathscr{H}^2$ = Laplacian or buckling

G_0 = Ratio of thermal flux at the uranium surface to average flux in the uranium

f_u, f_m, f_{al} = fractional thermal neutron absorption in the uranium, moderator and aluminium.

L^2 = thermal diffusion area = $\Sigma L_i^2 f_i$ with

$L_m^2 = 1{\cdot}26 \times 10^4$ cm^2

$L_u^2 = 1{\cdot}4$ cm^2

$L_{al}^2 = 4{\cdot}32 \times 10^2$ cm^2

The following 2500 m/sec data were used in the analysis:

$$N\sigma_{\text{mod}} = 0{\cdot}716 \times 10^{-4}\ \text{cm}^{-1}$$
$$N\sigma_{al} = 1{\cdot}28 \times 10^{-2}\ \text{cm}^{-1}$$
$$N\sigma_{u} = 0{\cdot}355\ \text{cm}^{-1}$$

The D_2O had an isotopic purity of 99·8%. Flux distributions were found with manganese foils (10% nickel) placed midway between rods on a principal diagonal. Best values of $\mathscr{H}^2$ were determined by least squares fitting of the experimental data. Corrections were made for the reactivity load of the foils and foil holders, and the results were corrected to a standard temperature of 20°C.

Values of G_0 were measured using 0·022-in. diameter manganese wire in a 0·094-in. diametral hole. To test the effect of streaming down this hole, some measurements were repeated with a smaller hole, approximately 0·025-in., and no change was detected within the errors of measurement. The manganese activities have not been corrected for epithermal activation, but the contribution due to neutrons above the cadmium cut-off is about 1·5% at the surface of the rod. The values of G_0 given in the table are approximately 5% low as compared with corresponding values from North American Aviation Company (COHEN, 1955) and this discrepancy is being investigated.

12. THE NRU REACTOR

NRU is a 200 megawatt heavy-water moderated and cooled reactor scheduled for completion in 1956. The reactor will be fueled with natural uranium and, although it will produce appreciable amounts of plutonium, it has been primarily designed to improve and extend the experimental facilities already available in NRX.

The reactor vessel is 11·5 ft. in diameter, 12 ft. high, and will contain approximately 86,000 lb. of heavy water, with a further 44,000 lb. of heavy water in the external system. The fuel elements are located on a 7·75 inch hexagonal lattice, and a maximum fuel charge will consist of 203 fuel rods and 16 dual-purpose shutoff and control rods. In practice several fuel element locations will be used for experimental facilities. Each fuel element contains 120 lb. of uranium in the form of five flat strips 10 ft. long, sheathed in aluminium and contained in an aluminium tube 2·39-in. internal diameter with 0·049-in. wall thickness.

The side reflector is 1 ft. of ordinary water in an aluminium alloy tank. One foot of steel thermal shielding, and 9·5 ft. of concrete comprise the side shield. One foot layers of heavy water in the main reactor vessel, above and below the uranium fuel elements, serve as top and bottom reflectors.

The dual-purpose shutoff and control rods are made of cadmium tubes sheathed in stainless steel, and are approximately 2·5 inches outside diameter. They will be attached to the drive mechanism by a magnet, and will drop freely into the reactor for an emergency shutdown.

One thermal column is provided, with dimensions approximately 8 feet square at the reactor end, 10·5 ft. square at the outer end, and 12 ft. long. Two staggered rows of bismuth rods installed across the inner face of the column reduce the γ-ray flux into the column and lower the heat production in the column. The

column is equipped with a neutron shutter located 2·5 ft. from the inner face. Major experimental facilities are transverse holes, 12 inches and 6 inches in diameter, tangential to the inner face of the column, with direct access from either side of the reactor. Similar transverse holes are also provided outside the shutter in regions of lower (and purer) thermal flux.

The side reflector of the reactor is pierced with numerous experimental holes; two sizes are used with nominal diameters of 12 and 6 inches. Some of the holes look at the side of the reactor vessel. Others face re-entrant cans in the side of the reactor vessel; the flux level at the inner end of the cans should be close to 10^{14} $n/\text{cm}^2/\text{sec}$. Four of the holes face the opposite ends of two tubes which extend all the way across the reactor vessel. These through-tubes, arranged one above the other, are elliptical in cross section with principal diameters of 4·125 and 2·875 inches. The upper tube is at the reactor mid-plane, and is empty right through the reactor while the other tube, three feet lower, has a D_2O target at its mid-point. Where the through-tubes extend through the reactor the lattice is split, adjacent rows of rods being set two inches further apart than normal.

It is possible to install special experimental facilities at any of the lattice positions. Three lattice positions in the reactor are provided with holes through the bottom shield permitting the installation of assemblies which pass through the reactor from top to bottom. They should prove particularly useful for "loop" experiments to test fuel assemblies for future power reactors. Arrangements have been made to house the associated pumps and plumbing for such loops in well shielded experimental rooms conveniently located. Special irradiation facilities will be provided at some lattice positions, each capable of holding a number of samples contained in capsules. Compressed air will be used both as a coolant and as a means of transporting the capsules between in-pile locations and experimental research laboratories.

APPENDIX

REACTOR DETAILS (GREENWOOD, 1955)

NRX REACTOR

Power: 40 MW Start-up: July 22, 1947

Maximum thermal neutron flux: $6{\cdot}8 \times 10^{13}$ n cm^{-2} sec^{-1}

Construction Details

Fuel Elements: Each rod assembly contains a 1·36-in. × 10½-ft. uranium rod clad with 0·08-in. 1S aluminium and surrounded by a water-cooling annulus 0·07 in. thick. Uranium weight 120 lb.

Fuel Core: Right cylindrical aluminium tank in the form of a calandria 8¾ ft. × 10½ ft. high. Tubes have 2·25-in. ID to form an air-cooling annulus around each rod assembly. Approximately 175 fuel rods arrayed in a hexagonal lattice with 199 positions in 22 rings; lattice constant 6-13/16 in.

Calandria contains up to 3300 gallons (18 tons) of heavy water. Total uranium load 10·5 tons.

Reflector: 9-in. graphite, 2½-in. gap for thorium rods, followed by a further 2 ft. of graphite. Total weight of graphite 58 tons.
Shielding: Two 6-in. cast iron thermal shields separated by an air-cooling channel, followed by 7 to 8 ft. of concrete. Three water-cooled cast-iron shields on top and four on bottom.
Overall Size: 34 ft. diameter × 34 ft. high.
Coolant: Up to 3500 Imp. gal/min of river water, mostly to the fuel rods. Air flow of 70,000 lb./hr. for shields and J-rods.
Temperature: Coolant water temperature rise 40°C. Heavy water maintained at a fixed temperature less than 120°F by circulating it to a light-water heat exchanger at 250 gal/min. Reflector not allowed to exceed 300°F.
Control: 18 shutoff rods of boron carbide powder in steel tubes; one control rod of cadmium slugs in a steel tube. All operate in vacant fuel-rod positions.
Experimental Facilities: Two thermal columns approximately 6 ft. square; 18 "self-serve" irradiation ports providing a total of 60 sample sites; twelve 4-in. and three 12-in. experimental holes with inner gates; 5½ in. × 10 ft. central thimble, with flux greater than 10^{13} *n* cm^{-2} sec^{-1} over entire volume.
Estimated Cost: $10 million.
Remarks: Fertile material: thorium (as oxycarbonate pellets) rods in an annular ring (see Reflector). Space for 90 rods at 4-in. separation. First reactor to be provided with such a facility.
Reactor rebuilt after power surge of Dec. 12, 1952.
Originally designed for 10 MW.

ZEEP REACTOR

Power: 10 watts Start-up: Sept. 5, 1945.
Thermal neutron flux : 10^{8} *n* cm^{-2} sec^{-1}

Construction Details

Fuel Elements: Uranium slugs 6 in. long, 1·28 in. diameter, stacked in 9-ft. 2S aluminium tubes and suspended from movable beams by a gimbal arrangement. Elements of other design can be installed for particular experiments.
Fuel Core: Cylindrical aluminium tank of 8½-ft. depth and 6¾-ft. diameter. Lattice arrangement and heavy-water depth are adjustable to suit the particular experiment.
Reflector: 3 ft. graphite on sides ; 2½ ft. on bottom.
Shielding: 3 ft. light water on sides. Concrete lid can be installed on top. An access passage 3 ft. wide is provided between reflector and shielding tanks.
Overall Size: Approximately 25 ft. × 15 ft. high.
Coolant: None.
Temperatures: Room temperature.
Control: Cadmium-coated stainless-steel plates jacketed in aluminium, operating in the vertical gap between core and reflector. Three are 20 in. by 12 in., ganged for coarse control; one is 20 in. by 4½ in., for fine control.
Two sets of four shutoff rods, each 1¼ in. by 1 ft. long, constructed of materials as above.

Experimental Facilities: See under Fuel Core. Holes in the lid facilitate flux measurements.
Estimated Cost: Approximately $200,000

NRU REACTOR

Power: 200 MW Start-up: 1956
Maximum thermal neutron flux: 3 × 10^{14} *n* cm^{-2} sec^{-1}

Construction Details

Fuel Elements: Approximately 200 natural uranium rods, sheathed in aluminium.
Fuel Core: Aluminium tank, height 12 ft., diameter 11·5 ft., with a heavy water capacity of 43 tons. Hexagonal fuel lattice with 7¾ in. spacing.
Reflector: One ft. of light water contained in an annular aluminium tank of ID 12·5 ft., OD 14·7 ft., and depth 11·3 ft.
Shielding: One ft. of water-cooled cast steel. 9·5 ft. of concrete, specific gravity 3·4.
Cooling: Heavy water flows upward within each fuel element tube and overflows into the moderator at the top. Total flow is about 22,000 Imp. gal/min of which 10% enters moderator tank directly and not through the fuel element. Eight heat exchangers are cooled by river water at 20,000 Imperial gal/min.
Air is exhausted from the reactor structure at 12,000 $ft.^3$/min, mostly from the thermal column.
Reflector water is also circulated to a heat exchanger.
Temperatures: Heavy water enters at 120°F and leaves tank at 167°F.
Experimental Facilities: One 8 ft. × 8 ft. thermal column with an inner shutter of cadmium, steel, lead and boral and a gamma filter of staggered bismuth rods.
About 25 horizontal experimental holes, two of which pass through the lattice.
Pneumatic facilities depending on pile load.
Extensive facilities for loops.
Control: Combined control and shutoff rods of cadmium tubing sheathed in stainless steel. About 16, all in fuel rod positions. Operation is completely automatic, including start-up and shutdown.
Emergency dumping of heavy water is also possible.
Remarks: Thorium rods can be installed in approximately 40 positions in an annulus between core and reflector tank where they are surrounded by CO_2 and cooled by light water.

REFERENCES

COHEN, E. R. (1955) Geneva Conference Paper, No. P/605.
GILBERT, F. W. (1954) *Chem. Eng. Progress* **50**, 267.
GLASSTONE, S. & EDLUND, M. C. (1952) *The Elements of Nuclear Reactor Theory*, D. Van Nostrand and Co. Inc., New York.
GRAY, J. L. (1953) *The Engineering Journal*, October, 1953.
GREENWOOD, J. W. (1955) A.E.C.L. Report No. CRR-590.
HADDOW, J. B. (1955) A.E.C.L. Report No. ED-28.
HATFIELD, G. W. (1955) *Mechanical Engineering* **4**, p. 124.

HONE, D. W., HURST, D. G. & WESTCOTT, C. H. (1955) AECL Report No. NDC-2.
HURST, D. G. (1953) A.E.C.L. Report No. GPI-14.
LAURENCE, G. C. (1947) *Bulletin of the Atomic Scientists*, November 1947.
LEWIS, W. B. (1952) A.E.C.L. Report No. DR-21.
LEWIS, W. B. (1953a) A.E.C.L. Report No. DR-28.
LEWIS, W. B. (1953b) A.E.C.L. Report No. DR-32.
ROBERTSON, R. F. S. (1955) Geneva Conference Paper, No. P/7.
SPINRAD, B. I., CARTER, J. C., & EGGLER, C. (1955) Geneva Conference Paper, No. P/835.
WARD. A. G. (1952) A.E.C.L. Report No. CRNE-510
WARD, A. G. (1953a) A.E.C.L. Report No. CRNE-551.
WARD, A. G. (1953b) A.E.C.L. Report No. CRNE-550.
WARD, A. G. (1954) A.E.C.L. Report No. CRNE-555.