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POWER REACTOR PROJECTS IN MEMBER STATES

Information gathered as a result of invitations from Canada, the United Kingdom of Great Britain and Northern Ireland and the United States of America

CONTENTS

63-5729

LIST OF ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safety
AECL	Atomic Energy Canada Ltd.
AEI	Associated Electrical Industries
ANL	Argonne National Laboratory
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing Materials
BNES	British Nuclear Energy Society
BONUS	boiling nuclear superheater power station
BTU	British thermal unit
BWR	boiling-water reactor
CANDU	Douglas Point nuclear power station
cc	cubic centimetres
CEGB	Central Electricity Generating Board
CNRN	Comitato Nazionale per le Ricerche Nucleari
CPPD	Consumers Public Power District
cu yds	cubic yards
DON	D_2O (heavy water) organic natural uranium reactor
EBWR	experimental boiling-water reactor
EGCR	experimental gas-cooled reactor
ELPHR	experimental low-temperature process heat reactor
ERR	Elk River reactor
ESSOR	Essai Orgel
F	Fahrenheit
ft	feet
GCR	gas-cooled reactor
GNEC	General Nuclear Engineering Corporation
h, hr	hour
HNPF	Hallam nuclear power facility
HTGR	high-temperature gas-cooled reactor
HWR	heavy water reactor
ICRP	International Commission on Radiological Protection
in.	inch
k_{eff}	multiplication constant, effective
kV	kilovolt
kW	kilowatt

kWh	kilowatt-hour
kWth	kilowatt thermal
lb	pound
mill	one-thousandth of a dollar
mrem	milliroentgen equivalent man
min	minute
MRC	Medical Research Council
MW	megawatt
MWd/t	thermal megawatt-day/metric ton
MWe	megawatt electrical
MWh	megawatt-hour
MWth	megawatt thermal
n.a.	not applicable
NPD	nuclear power demonstration
NSPC	Northern States Power Company
OMR	organic moderated reactor
ORNL	Oak Ridge National Laboratory
pH	hydrogen-ion concentration
PNPF	Piqua nuclear power facility
PRO	Programme reactor organic
PRWRA	Puerto Rico Water Resources Authority
psia	pound per square inch absolute
psig	pound per square inch gauge
RCPA	Rural Cooperative Power Association
rem	roentgen equivalent man
SGR	sodium graphite reactor
s.o.	sent out
ss	stainless steel
SSPWR	small-size pressurized water reactor
TVA	Tennessee Valley Authority
UC	uranium monocarbide
UK	United Kingdom of Great Gritain and Northern Ireland
UKAEA	United Kingdom Atomic Energy Agency
U-Mo	uranium molybdenum
USAEC	United States Atomic Energy Commission
VBWR	Vallecitos boiling-water reactor
vol	volume
wt%	weight per cent

A. INTRODUCTION

1. As part of its activities in connection with the development of nuclear power, and in response to the resolutions[1/] adopted by the General Conference, the Agency has been undertaking a continuing study of the technology and economics of power reactors, particularly with reference to the needs of the developing countries.

2. It may be recalled that in 1959, during the third regular session of the General Conference, the Government of the United States of America offered to provide the Agency with relevant technical and economic data on several small power reactor projects of its Atomic Energy Commission. The Agency accepted the offer and during the course of the last three years it has been following the development of several power reactor projects in that country.

3. Early in 1962 the Canadian Government made a similar offer regarding NPD and the Douglas Point nuclear power station. During the last regular session of the General Conference the Government of the United Kingdom of Great Britain and Northern Ireland also extended to the Agency the same facilities with regard to the Bradwell Nuclear Power Station.

4. This report summarizes the information gathered on nine power reactor projects in the Member States mentioned above. It should, however, be read in conjunction with two earlier reports[2/] since, on eight of the reactors dealt with, only new and additional data is given.

5. The following table lists the power reactor projects covered by this report and shows their present status.

1/ GC(II)/RES/27, GC(III)/RES/57, GC(IV)/RES/86, GC(V)/RES/106 and GC(VI)/RES/109.

2/ GC(V)/INF/41 and GC(VI)/INF/54.

Table 1

Power reactor projects

Reactor	Type	Net output MWe	Location	Actual or expected criticality	Owner/ operator	Remarks
NPD	HWR	19.5	Chalk River, Ontario	April 1962	AECL/Ottawa Hydro Power Commission	Full power reached June 1963
Bradwell	GCR	2x150	Bradwell, UK	Reactor 1: August 1961 Reactor 2: April 1962	CEGB	Full power of reactor 1 reached August 1962; of reactor 2, December 1962
Elk River	BWR, indirect cycle	22	Elk River, Minnesota	November 1962	USAEC/ RCPA	Full power reached September 1963
BONUS	BWR, nuclear superheat	16.3	Punta Higuera, Puerto Rico	January 1964	USAEC/ PRWRA	Full power expected May 1964
Pathfinder	BWR, nuclear superheat	62	Sioux Falls, South Dakota	October 1963	NSPC	Full power expected November 1964
Piqua	OMR	11.4	Piqua, Ohio	June 1963	USAEC/City of Piqua	Full power expected October 1963
Hallam	SGR	75	Hallam, Nebraska	January 1962	USAEC/ CPPD	Full power reached July 1963
Peach Bottom	HTGR	40	Peach Bottom, Pennsylvania	Third quarter 1964	Philadelphia Electric Power Company	Construction on schedule
EGCR	GCR	22	Oak Ridge Tennessee	October 1964	USAEC/TVA	--

B. CANADA

I. THE NUCLEAR POWER DEMONSTRATION PROJECT

General

6. The nuclear power demonstration project completed its first year of power operation in June 1963. This plant is intended to demonstrate base load operation, on-power refuelling, high plant availability, and safety in operation. This small station, with a net output of 19.5 electrical megawatts, provides valuable data applicable to the design, construction and operation of similar but larger nuclear-electric stations, including information on fuel element performance, loss of heavy water, power regulation, response to load demand, response to system disturbances, reliability of components, operating techniques, maintenance methods and cost.

Commissioning

7. The commissioning programme proceeded in four phases:

Phase A - Pre-critical

Phase B - Low power reactor measurements

Phase C - Power run-up

Phase D - Full power tests

8. Phase A. The Phase A work started before construction was completed. During the early part of this period the service systems such as water, compressed air, electricity and ventilation were placed in operation.

9. The primary system which includes all vessels and equipment which are to contain heavy water was pressure tested with air initially, then the entire primary system was filled with helium and tested for leaks. For example, each tube of the D_2O-H_2O heat exchanger was tested, with the maximum permissible leak rate being 10^{-5} cc/sec for a differential pressure of 1 atmosphere. The calandria was also tested on the basis of the same criterion. For the moderator and coolant system as a whole, the permissible leak rate was 1 cu ft/day. When the system had been made as leak-tight as possible, heavy water was added. The system was given a hydrostatic test with heavy water (no light water was ever used in the primary system). This method proved to be satisfactory and the necessary modifications were performed with little difficulty. No cleaning of this system was performed with any fluid except heavy water, the pH of which was

maintained at 10.5. After adding the heavy water, the system was flushed by circulating heavy water at 400°F for several days. This treatment also served to condition the carbon-steel surfaces, by providing a thin oxide coating and particulate matter was removed by filters and screens.

10. The primary system contains three different materials, namely carbon steel, stainless steel, zircaloy. It might be anticipated that corrosion would be a problem. However, such has not been the experience. It has been found that by keeping the pH high and the oxygen content low, the present piping is satisfactory in all applications except small instrument lines. For the latter, stainless steel would be preferable to carbon steel. The initial loading of fuel was performed remotely from the control room, using the automatic fuelling machines. Loading was accomplished beginning 28 February 1962 and being completed 21 March 1962. The start-up apparatus was checked out with a polonium-beryllium neutron source. Phase A ended with the approach to critical 11 April 1962, approximately three and a half months after completion of construction.

11. <u>Phase B</u>. Phase B included zero power measurements of several nuclear parameters and coefficients, such as moderator level at initial clean critical, moderator temperature coefficient, heat transport system temperature coefficient, light water reflector savings, difference in reactivity when moderator level is raised from clean critical height to full tank, reactivity worth of booster rod, and rate of decrease in reactivity upon dumping moderator. The satisfactory performance of the control system was also demonstrated. Some of the results of these measurements are given in the table below.

Table 2

<u>The nuclear power demonstration project: Comparison of design and measured parameters</u>

Parameter	Design value	Measured value
Clean critical level	89.3 inches	97.5 inches b/
Moderator temperature coefficient	-0.06 mk/°F a/	-0.04 mk/°F
Heat transport system temp. coeff.	-0.024 mk/°F	-0.02 mk/°F
Light water reflector savings	3 mk	3.6 mk
$\frac{\Delta k}{k}$ Clean critical to full tank	37 mk	39 mk
$\frac{\Delta k}{k}$ for enriched booster	2.5 mk	2.4 mk
Dump rate $\frac{\Delta k}{k}$ in first second	3 mk	5 mk c/

a/ mk = $10^{-3} \frac{\Delta k}{k}$.

b/ Error due to underestimate of depleted uranium - now reconciled.

c/ Conservative value - measured results more favourable.

12. One of the more interesting of the above measurements was the determination of the change in reactivity with increasing moderator level. Initial criticality was achieved when the heavy water level reached 97.5 inches. A small, measured amount of cadmium sulphate was then injected into the heavy water, and the critical height of moderator was again determined. The reactivity worth of cadmium in the NPD lattice was already known, hence each part per million of added cadmium corresponded to a known $\frac{\Delta k}{k}$. By further additions of known amounts of cadmium poison, the relationship between reactivity and moderator level was determined for the full range of moderator levels. After completion of the test, the cadmium was removed from the heavy water by ion exchange resins. Incidentally, although boron might have been used in this test, the removal of boron from the heavy water would have required more ion exchange resin than did cadmium.

13. A discrepancy between the estimated and the actual critical height of the moderator was caused by an incorrect estimate of the effect of depleted uranium. The depleted uranium had been used to reduce the reactivity of the initial clean core and thus to permit early operation at full power. The reasons for this discrepancy have now been fully resolved.

14. When operating at very low power, the reactor was initially placed on manual control. The power level was increased to 0.01% of full power and it was determined that the computer was responding satisfactorily. Then control was switched to automatic where it was permitted to remain for three weeks. Phase B ended on 8 May 1962, lasting approximately one month.

15. Phase C. During Phase C, the reactor was operated at power levels of 6%, 10%, 25%, 50%, 75% and 100% during which tests were performed to establish dynamic performance of the reactor-boiler and turbine-generator. Also measurements were conducted to establish the adequacy of cooling, shielding, recombination, etc. Generally, the power run-up went quite smoothly and no major difficulties were encountered. First electricity was generated 4 June, and full power was reached 28 June 1962.

16. Phase D. During Phase D, the station was modified to eliminate a number of early problems which caused poor performance. The station was declared in-service on 25 September 1962. At this time, the fuelling machines had only been used for off-power fuelling and development work for on-power fuelling was still in progress.

17. Heavy water losses. Heavy water losses are of two kinds, chronic daily losses and spills or non-recurring incidents. The major part of the losses have been chronic daily losses. Recently, the loss has averaged 10-12 pounds of heavy water per day.

18. One weakness identified during commissioning was poor workmanship in the mechanical joints of heavy water instrument tubing. This problem can be avoided by adopting proper quality control methods for mechanical joints as stringent as those required for welded joints. For the CANDU reactor it is planned to eliminate many of the mechanical joints and use welded joints instead. Development work is now in progress to permit joints in stainless steel piping to be welded automatically. In a number of instances in the NPD plant it has been decided to remove valves and fittings, replacing them by an integral length of tubing.

19. No measurable reduction in the isotopic content of the heavy water took place in either the moderator or heat transport system due to down-grading during commissioning. The total inventory of heavy water required was 74 400 kilograms (164 000 lbs) as contrasted with the original estimate of 72 600 kilograms (160 000 lbs). The total loss during the commissioning period of 240 days was 8500 pounds which is considered to be economically acceptable and less serious than that predicted. The major loss during commissioning was caused by chronic vapour loss from numerous small leaks rather than acute spills. The heavy water loss averaged 16 kilograms per day (35 lbs per day) in the early stages of operation, but as of May 1963 the losses averaged 1.5 to 5.5 kilograms of heavy water per day of which about 50% is thought to come from valves.

Operating experience

20. Pumps and seals. The moderator pumps and helium blowers employ multiple shaftseals. The outboard seals are in contact with oil and the inboard seal is of course in contact with heavy water. A labyrinth is provided between the inboard and outboard seals to separate the heavy water and oil seal leakage. The interspace does not effectively separate oil and heavy water leakage as intended and, as a result, oil was being added to the moderator system. Because this oil was converted to acid under irradiation, the ion exchange columns in the moderator system were being consumed at the rate of one per day rather than at the design rate of one per four months. This problem has been temporarily solved by

removing the oil before pumping the collected heavy water back into the system. An improvement is planned to eliminate the oil mixing in the first place. The inboard seal on the NPD pumps is made up of two adjacent sealing rings, one rotating with the shaft and the other stationary in the pump housing. The two faces, one made of stellite and the other graphite, are in intimate contact, rubbing against each other but lubricated and cooled by heavy water which leaks across the sealing surface.

21. In the early operation, the mechanical shaft seals of the primary pumps lasted 200 hours on an average, which was entirely unacceptable. Improved degassing and venting techniques have already increased the life to more than 1500 hours and it is expected that a seal life in excess of 7000 hours will be demonstrated within the next two years.

22. Turbine. A delay of three weeks in reaching full power was experienced due to vibration in the steam turbine. The cause was found to be a faulty bearing, and the difficulty was corrected.

23. Recombiner. The catalytic units for the recombination of deuterium and oxygen produced in the moderator did not operate satisfactorily at first due to condensation of heavy water and wetting of the active surface of the catalyst. Heaters were installed to maintain a dry gas mixture.

24. Gas locking of pumps. Heavy water in the high pressure heat transport system is cooled and depressurized prior to being passed through ion exchange columns for purification. The purified heavy water is then returned to the pressurized system through charging pumps. Helium was found to be gas locking these charging pumps. This was due to the rather considerable evolution of helium upon depressurization, since the solubility of helium in heavy water is about 1600 cc per litre at 450°F and 1000 psia but only 0.1 cc per litre at 180°F and 15 psia. The problem was eliminated by installing degassing facilities.

25. Gas locking was also encountered in the moderator pumps, which can be used to transfer heavy water from the dump tank to the calandria. This problem was caused by entrainment of helium by heavy water as it spilled into the dump tank when the liquid level in the dump tank was low. Corrective action was taken by a modification which spills the heavy water into the end of the dump tank, permitting the helium to disengage prior to reaching the pump suction opening at the centre of the tank.

26. Fuelling machine. The fuelling machine, operated remotely from the control room, was employed to load the entire initial fuel charge. During this loading operation, the primary heavy water was at room temperature and at 100 psi pressure.

27. On 3 December a major problem was encountered during on-power proof testing of the refuelling machines. Two simultaneous leaks developed on one of the machines, one at the point where the machine connects to the reactor pressure tube and the other in one of the hydraulic control lines of the machine itself. Because of the first leak, hot high-pressure heavy water coolant was released into the reactor vault, and because of the second leak the machine could not be moved to rectify the first. Fortunately, the automatic spray of cold heavy water into the reactor vault was actuated in just the fashion intended to reduce vault pressure in such emergencies, and all safety devices and provisions for containment of heavy water performed successfully. Although the spilled heavy water suffered a small downgrading in isotopic purity, operation was resumed at the end of December. The capability of the system for on-power fuelling has not yet been demonstrated but further on-power testing will be scheduled and it is expected that on-power fuelling will be satisfactorily accomplished this year.

Operating characteristics

28. Start-up and shutdown. If the reactor is to be started up after a shutdown of several days, assuming that the system is cold and that the heavy water is in the dump tank, a period of two to two and a quarter hours is required before the turbine picks up load, initially at 2 MWe. Then load may be increased at the rate of 1 MWe per minute, until the generator output reaches 12 MWe (gross). To increase from 12 MWe to full power (22 MWe gross) requires an additional 27 hours, during which time xenon controls the rate of power rise.

29. If the station experiences a trip after having been operating at full power, the plant output can be restored to full power in approximately 25 minutes after the trip has taken place and the fault corrected.

30. To shut down the reactor, the helium pressure differential is reduced to zero by opening helium dump valves thereby dumping the moderator into the dump tank. By this means, reactivity is reduced by 0.005 $\frac{\Delta k}{k}$ within 0.95 seconds after the trip signal (by actual measurement), and reactor power is reduced from full power to 25% of full power within three seconds after the trip signal.

31. Response to changes in load. The NPD station is designed as a base load unit. However, the electrical output can be changed at the design rate of 20% of full power per minute (4 MWe/min) without difficulty. During normal operation the entire station runs automatically, with an electronic computer maintaining a constant steam pressure at the turbine throttle.

32. Response to system disturbances. In the very early stages of operation, the reactor tripped frequently because the reactor regulating system was not adequate to cope with minor disturbances. The major cause of this trouble was traced to a non-linear characteristic in the ion chamber amplifiers. Now the regulating system is giving top performance and provides excellent response and stability. The station can now override most disturbances such as change-over of pumps, changing temperature controller set points, line voltage changes, line frequency changes, on-power emergency stop valve tests, on-power dump valve tests, etc. However, further improvements are required to withstand the following three major disturbances:

(a) Turbine overspeed trip. The turbine can handle a loss of line but trips out on full load rejection of generator;

(b) Reactor overpressure trip. The reactor trips out on both generator rejection and loss of line due to a small transient overpressure in the surge tank; and

(c) Reactor undervoltage trips. On major transmission line faults where the 13.8 kV voltage drops to 10 kV, the reactor trips out on undervoltage due to fail-safe protective relays.

33. Such large disturbances do not occur frequently. Nevertheless, no major difficulty is anticipated in correcting these problems and it is intended to demonstrate the ability of the station to stay on line under such circumstances.

34. Operating and maintenance techniques. In order to open a pipe or disconnect a component for maintenance purposes in a line where no valves have been installed, it has been found convenient to freeze a plug of heavy water inside the pipe. Such plugs can be formed quickly, effectively, and inexpensively by attaching to the outside of the pipe a trough made with plastic sheeting and paper tape, then filling with liquid nitrogen or solid carbon dioxide. During the commissioning of NPD more than 2000 such plugs were utilized on various sizes and kinds of heavy water piping. One lesson learned in this work concerns the formation of two heavy water ice plugs close to each other in the same tube. As the plugs

grow in length, the liquid heavy water trapped between the two plugs is compressed and the tube may rupture. Plugs may be formed even when heavy water is flowing through the pipe. For example, with an initial flow of 1 gallon/min through a 1.5 inch pipe, a plug can be formed in about 20 minutes. As a result of this experience, it is recommended that certain valves might be omitted in future heavy water reactors. This would both save capital cost for the valves and reduce maintenance cost and the possibility for loss of heavy water at valve stem and pipe point.

Safety analysis

35. The consequences of a number of types of accidents which could occur in NPD have been examined, including loss of heavy water coolant, loss of regulation, coolant pump failure, loss of light water from boiler, and loss of pressure control. The most severe accident appears to be the severance of the largest (16 inch diameter) pipe in the heavy water primary coolant circuit, between the pumps and the steam generator. Providing that the protective system, including dousing water, and the emergency injection systems were operative, it is calculated that there would be no fuel ruptures and no release of fission products, that the maximum temperature reached by fuel cladding would be 1650°F. The safety systems provided in the plant would terminate this most severe accident, without any undue hazard.

Operating staff

36. The staffing plan at NPD has been revised slightly as compared with that reported last year. The present operating staff is shown in the table below.

Table 3

The nuclear power demonstration project:
Staffing plan

Category	Number of persons
Administration	
Plant superintendent	1
Assistant plant superintendent	1
Operation	
Shift supervisors	5
Operators	22
Radiation inspector	1
Maintenance	
Supervisor	1
Control maintainers	5
Mechanical maintainers	7
Service	6
Technical	
Technical engineer	1
Assistant technical engineers	4
Fuel engineer	1
Chemical unit	2
Radiation protection officer	1
General	
Nuclear training centre - trainees	3
Clerical unit	7
TOTAL	68

Fuel costs

37. Fabrication of fuel for NPD provided the manufacturer, the Canadian General Electric Company, with the opportunity to develop and improve shop techniques and processes. The cost of the fuel elements therefore includes a great deal of development work and is not particularly meaningful. However, the same manufacturer produced complete fuel assemblies for the initial loading of CANDU, which were purchased by AECL at a total (including uranium) price of $29.50 (Canadian) per pound of UO_2 contained in completed fuel bundles. This is US $68.25 per kg

of contained uranium. Assuming a burn-up of 10 000 MWd per metric ton of uranium and an over-all station efficiency of 28.2%, this means that replacement cost of fuel is 1.0 mill per kilowatt-hour. Fuel inventory charges would be an additional 0.05 to 0.10 mill per kWh. Fuel fabricators in Canada consider these costs as realistic and might be willing to offer a ten-year guarantee that fuelling replacement costs over the ten-year period would average no more than 1 mill per kWh.

Selected references

38. A list of selected references concerning the nuclear power demonstration project is given below:

McCONNELL, L.G., Commissioning and Initial Operation of Canada's First Nuclear-Electric Generating Station, Hydro-Electric Power Commission of Ontario, September 1962.

The Heavy Water Power Reactor Handbook 1963, Canadian General Electric Company.

Nuclear Power Demonstration Reactor, AECL 1634, Nuclear Engineering, October 1962.

Each of the following papers presented at the Canadian Nuclear Association Conference, Montreal, Canada, 27-29 May 1963:

McCONNELL, L.G., The Early Operating Experience of NPD.

FULFORD, P.J., Performance of Mechanical Seals for Primary Coolant Pumps.

MOORADIAN, A.J., First NPD Fuel Charge, Present Performance and Basis for Continued Confidence.

ROBERTSON, R.F.S. and RAE, H.K., Canadian Experience with Reactor Moderators and Coolants.

HAYWOOD, L.R., Trends in Design and Costs of Heavy Water Moderated Nuclear Power Plants.

WILLIAMS, N.L., Design and Cost Estimate of a Multiple Unit D_2O Moderated and Cooled Nuclear Power Plant.

C. THE UNITED KINGDOM OF GREAT BRITAIN AND NORTHERN IRELAND

General

39. In the UK, the main emphasis in power reactor development has been on gas-cooled, graphite-moderated reactors. On the basis of the successful design and construction of Calder Hall, the first station to produce nuclear power on a commercial scale, a large programme of commercial nuclear power stations and of further development of the gas-cooled graphite-moderated reactors was planned.

40. Calder Hall began operation at full power in 1956, and its design was the starting point for a series of commercial nuclear power stations announced in February 1955 in the British Government's programme of nuclear power generation. The first of these stations, Berkeley and Bradwell, were commissioned in 1962. All stations in the programme have fuel elements of metallic natural uranium rods, clad in a magnesium alloy known as "magnox". The over-all efficiency of the magnox stations is lower than that of conventional stations due to the limitations of the temperature of the steam cycle imposed by the relatively low ignition point of the magnox cladding of the fuel elements. This fact also limits the development potential of the magnox stations. However the later stations in the power programme already show considerable advance over the first stations.

41. Research and development began at an early stage to advance the technique of the gas-cooled reactors. In August 1957, within a year after the operation of Calder Hall at full power, the UK Atomic Energy Authority initiated a plan for the construction of an advanced reactor of the gas-cooled graphite-moderated type. This reactor, the experimental advanced gas-cooled reactor (AGR) was constructed at Windscale, Cumberland, and achieved its design output in January 1963.

42. The present programme of commercial nuclear power stations in the United Kingdom is shown in the table below.

Table 4

Programme of commercial power stations in the United Kingdom

Station	Location	Year of commissioning first reactor	Net output capacity MWe
Berkeley	Severn Estuary	1962	275
Bradwell	Essex Coast	1962	300
Hinkley Point	Bristol Channel	1963	500
Trawsfynydd	North Wales	1964	500
Dungeness	Kent Coast	1964	550
Hunterston	Ayreshire, Scotland	1964	300
Sizewell	Suffolk Coast	1965	580
Oldbury a/	Severn Estuary	1966	560
Wylfa a/	Anglesey Coast, Wales	1967	1000

a/ Concrete pressure vessel.

43. All of these stations, except Hunterston, are owned by the Central Electricity Generating Board (CEGB) which is responsible for generation of electricity in England and Wales. The Hunterston station is owned by the South of Scotland Electricity Board. Each of these stations has two natural uranium-fuelled, carbon dioxide-cooled and graphite-moderated reactors.

44. Nuclear plants represent about 18% of the CEGB's programme of new generating plants for the four years 1961-1964. On the latest load estimates the total generating capacity expected to be in service in the CEGB's area at the end of 1964 is 37 250 MW. The capacity of nuclear plants will at that time be approximately 5% of the total and they will generate at least 6% of the total energy production.

45. In the following paragraphs is described the Bradwell nuclear power station, one of the two commercial stations at present in operation.

II. THE BRADWELL NUCLEAR POWER STATION

General

46. The Bradwell nuclear power station is located on the south-east extremity of the Blackwater estuary and is situated about one and a half miles from the village of Bradwell-on-Sea.

47. The construction site occupies some 75 acres and the area contained by the security fence of the operational station about 20 acres. Access to the site is by road, the nearest rail head being eight miles from the site, and £250 000 has been spent by the Central Electricity Generating Board on road improvements. The site is adjacent to a war-time airfield, the perimeter track of which provides an access road to the power station.

48. The construction of this power station, which is one of the two first nuclear generating plants to be built for the Central Electricity Generating Board, was initiated by the Board's Nuclear Power Branch and responsibility for the project was later passed to the Southern Project Group of the Generating Board. The station was built for the Generating Board by the Nuclear Power Plant Company Limited who, with A.E.I. - John Thompson Nuclear Energy Company Limited, are now partners in the Nuclear Power Group.

49. Access to the site was given on 1 January 1957, the first reactor was put on commercial load on 1 July 1962 and the second reactor on 12 November 1962.

Table 5

The Bradwell nuclear power station: Key programme dates

Item	Reactor 1	Reactor 2
Construction began	January 1957	January 1957
Construction complete	April 1961	November 1961
Pressure vessel tested	October 1959	May 1960
Graphite laying complete	October 1960	March 1961
Fuel loading commenced	August 1961	April 1962
Reactor critical	August 1961	April 1962
Fuel loading complete	October 1961	May 1962
Turbo-alternators on load	July 1962	November 1962
Nominal full power achieved	August 1962	December 1962

Design concept[3/]

50. The design of the Bradwell Power Station was undertaken in May 1955 and completed in its essentials by October 1956. The reasoning behind the design at that stage was therefore coloured largely by the experience gained in the design and construction of the Calder Hall reactors.

51. The first principle of the design was that the conception and operation of the reactor unit should be as simple and straightforward as possible. On this basis, refinements indicated by nuclear or thermodynamic studies were often omitted in the interests of mechanical simplicity. It was also decided to have inside the reactor vessel the absolute minimum of equipment which was not capable of removal for inspection or replacement.

52. To supply base load over a period of months, it was considered essential that the station remain on load during removal or replacement of fuel elements. The charging and discharging equipment was therefore designed to carry out both these operations with the reactor under full power and operating conditions.

53. Plant configuration. The station output itself resulted from a decision to use only turbines with double exhausts. A gross output of 52 MW per machine was considered reasonable for the steam conditions expected and with three machines per reactor the net output was then 150 MW, i.e. 300 MW for the whole station. In fact, the turbo-alternators have been designed to give 5 per cent extra output, and a maximum net station output of 320 MW might be achieved.

54. The layout of reactor gas ducts and boilers was arranged to give the maximum amount of natural circulation of the coolant gas. At full operating temperatures the reactor is designed to run at 10% full output without any assistance at all from the blowers. The blowers are driven by electric motors which take their power supply from variable-frequency turbo-generators fed by low pressure steam, thus permitting operation at any part load down to 20%.

55. A spherical pressure vessel was chosen in preference to a cylinder owing to savings deriving directly from the higher gas pressure possible with a given thickness of steel plate.

3/ The important design features of the Bradwell power station are given in the Annex.

56. Reactor dimensions. The height of the core was determined more by the geometry of the layout inside a spherical vessel than by thermodynamic or nuclear considerations. A fuel element was available with potentially low friction factor, and so channel pressure drop was no problem. Height was, however, limited on account of the space required for operation of the charging mechanism inside the spherical vessel and also by the wish to restrict the height of the reactor building, which is determined by the length of control rods and the height of charge machine into which they have to be withdrawn. The core height was eventually chosen for economy to suit an integral number of the longest graphite bricks available.

57. Considerable attention was paid to the conflict in use of reactivity for flux flattening or for longer fuel irradiation. Therefore, for any given core diameter a definite optimum (low) rating was established where the thermodynamic disadvantages of low rating were matched by the economic savings at long fuel life. On this basis the core diameter was finally chosen to be the smallest whose optimum rating did not take the uranium centre temperature above a nominal limiting value of 570°C.

58. The fuel rod diameter was chosen to be the 1.155 in. adopted in the Calder Hall reactors, although high ratings at first suggested using 1 in. or less. Channel diameter was selected primarily to give an acceptable channel pressure drop.

59. Fuel element parameters and choice of reactor gas temperatures. The fundamental temperature limit applied to reactors of graphite-moderated magnox-clad CO_2-cooled type contained in a steel pressure vessel is that in the event of the worst credible accident the temperature of the channels will not exceed the ignition temperature of magnox. The fuel element heat transfer surface chosen was adopted for the first time in the design of Bradwell. A study of it led to the selection of reactor gas outlet temperature. Bearing in mind the easing of the Wigner growth and the energy storage problem with the increase in temperature, a balance was struck between the capital cost and generating cost and 180°C was chosen for the inlet temperature.

60. Over the range of 1 to 2 psi, the pressure drop across the boilers appeared to have little effect on the capital or generating cost. A mean value of 1.5 psi pressure drop was chosen as this fitted in with the size and proportion of boiler that could conveniently be manufactured in the works, transported and lifted into position. The temperature of feed water was chosen at 190°C giving maximum efficiency for optimum number of stages of feed heating.

61. <u>Control</u>. The core and standpipes have been designed to minimize any movement which might prevent control rod insertion. Control rods are moved in and out of the reactor by motor-driven chains. The reactor is controlled by the variation of reactivity caused by the insertion or withdrawal of ferro boron sintered absorber rods which are divided into the following groups:

(a) 80 bulk control rods, controlling 6.5% reactivity dispersed over the core, which can be moved in one or two groups. 30 of the bulk rods are fitted with secondary shutdown devices which can unlatch in the event of a rapid depressurization accident, allowing these rods to fall freely under gravity into the core;

(b) 11 safety rods controlling up to 1% reactivity spread over the unflattened region, normally moving in a group; and

(c) 28 "grey" sector rods, distributed over the core controlling ± 0.5% reactivity about the mean position under normal operating conditions. These rods control instabilities arising from positive temperature coefficient effects. The rod movements are governed by channel gas outlet temperature, and the rods normally work partially inserted into the core.

62. The design of the rods is such that with the sector rods and safety rods withdrawn from the core, the reactor may be shut down on the bulk rods alone. The rate of release of reactivity is such that if individual groups of rods are motored out at their maximum speeds, no transient will occur which cannot be held by the safety protection provided.

63. Reactivity balance. The following is an account of reactivity for three conditions intended to demonstrate that there is always sufficient absorber available to shut down and hold down the reactor. These conditions are:

(1) Hot, unirradiated, poisoned;

(2) Reactivity conditions corresponding to equilibrium fuel cycle based on a mean discharge irradiation of 3000 MWd/t subject to the limitation of maximum fuel element dwell time in the reactor of 5 years; and

(3) At the peak of the reactivity build-up curve.

64. The values quoted in the following table are design values calculated before the commencement of the commissioning programme.

Table 6

The Bradwell nuclear power station: Design values

Item	Reactivity in % $\frac{\Delta k}{k}$ for conditions: (1)	(2)	(3)
Initial excess reactivity	4.97	4.97	4.97
Sector rod allowance	-0.4	-0.4	-0.4
Long-term reactivity allowance	0	+0.5	1.57
Effective initial excess reactivity	4.57	5.07	6.14
Effect of temperature	-1.94	-1.94	-1.94
Effect of xenon poisoning	-1.82	-1.82	-1.02
Effect of flattening	-0.81	-1.81	-2.88

65. The largest rapidly occurring reactivity transient expected during actual operation is the insertion or removal of the grab-head during refuelling, giving 0.05% reactivity which can be counteracted by movement of rods in the appropriate sector.

Development and construction

66. Fuel element. During its development programme, the fuel element ran into several difficulties both on the thermodynamic and on the mechanical side. With an element with 40 fins, which was the number finally adopted for the first charge at Bradwell, it was possible to obtain a performance acceptable to the over-all plant design, but still slightly less than was originally assumed.

67. On the mechanical side, problems began to appear when several channels of typical fuel elements were loaded into a Chapelcross reactor in December 1959. After a very short period of irradiation it was discovered that severe deformation of the gas flow splitter assemblies was occurring and all the elements had to be immediately discharged. A large programme of investigation was then undertaken both by the designers and by the UKAEA to examine this phenomenon. The element was successfully tested in out-of-pile endurance rigs and in its redesigned form was loaded once more in the Chapelcross reactor in January 1961.

68. The presence of the spring-loaded spider arms used to stabilize the elements necessitates the addition of the petals to the fuel element grab to guide its latches past these arms before gripping the element. This apparently simple operation has, in fact, caused a good deal of difficulties both in the design of the grab and the subsequent testing and commissioning of the charge machine.

69. Reactor core. The core was designed as separate columns of nominally 8 in. square graphite bricks each containing a 4.2 in. diameter channel. These columns, together with the surrounding graphite reflectors, formed a stack 31 ft high and 45 ft diameter.

70. In March 1958, due to revised data on the storage of Wigner energy following temperature excursion in one of the Windscale air-cooled reactors, it was considered necessary to increase the bottom graphite temperature. It was too late to raise the reactor design gas inlet temperature and the only alternative was to fit insulating sleeves in the graphite channels. Sleeves were trepanned from the existing graphite bricks for the complete core of No.1 reactor. However, further data on graphite received in November 1959

made it difficult to justify the use of sleeves, which still had many mechanical problems. A decision was then made to revert to the original design, which meant the scrapping of one complete core of trepanned bricks and the machining of a new core. Revised graphite data also necessitated changes in the gaps between graphite bricks.

71. The problem of providing the lateral support to the columns in such a way that irradiation-temperature-time effects are insignificant, was solved by adopting a horizontal metal mesh through the bricks of a large number of small zirconium struts or pins, at levels where shrinkage was expected.

72. Fuel Handling. In the interests of simplicity, a universal charge machine was adopted capable of carrying out all the refuelling operations over a standpipe during one visit.

73. The fuel is handled by an element grab which has automatic mechanical operation of its three latches with an electric solenoid lock. Considerable development has been necessary to obtain an element grab cable with the necessary qualities of flexibility, life and electrical reliability under the reactor conditions in which it has to work.

74. Throughout the construction of the station the fuel handling equipment probably experienced more design changes than any other item of the plant, in order to meet the new requirements of the fuel element and core, as well as to overcome problems inherent in the movement of fuel at high temperatures.

75. Considerable difficulties were also met in designing a charge chute capable of reliable operation under the high thermal stresses encountered in an on-load reactor.

76. Reactor vessel and gas circuits. In fabrication of the vessels both machine and hand welding were employed. Four plate panels, made up by machine welding, were hand welded into rings weighing up to 200 tons, which were then lifted into the reactor vault and built up into the vessel by hand welding of the horizontal joints. The heavy lifts were made by a Goliath crane, which was at the time a unique feature of the Bradwell site. All seams were checked by portable X-ray and, while the amount of cutting out and repair work was small, some trouble was experienced on No.1 vessel due to a defective plate in the top cap. This defect was a lamination which had not been revealed by inspection in the works and was found when cutting a hole for a standpipe nozzle.

77. The layout of gas circuits at Bradwell was determined by two major considerations:

(a) The need to keep six boilers under the Goliath crane; and

(b) The desire to keep the building height to a minimum.

78. To minimize parasitic stresses, great care was taken in installing the duct work. Parallelism between closing flanges before bolting up of better than 0.01 in. over 60 in. was achieved. During the early weeks of operation site measurements were taken of the deflection of the bellows and the movement of the ductwork at temperatures up to 200°C and were found to be in close agreement with the design figures.

79. The valves required in the gas ducts to isolate each boiler from the reactor presented a considerable problem. In 1956 no manufacturers could offer a valve to meet the requirements in the sizes needed but a number of them took up the problem and by 1957 promising designs were produced, although a great deal of development and proving work was needed.

80. Main blowers. In view of the novel design of the blower, a prototype was constructed to establish the performance characteristics and mechanical behaviour of the proposed design. The tests were carried out at the normal working temperatures and above them. It was decided to assemble and works test each subsequent blower and its "pony" motors in order to reduce commissioning time on site. After the blowers had run for a considerable time on site, but while reactor commissioning was still continuing, a routine inspection revealed that damage to the inlet guide vanes was occurring due to buffetting of the shroud ring which was held on the tips of the vanes and extended over the rotor blades to provide the running clearance. Modifications have been carried out to rectify this defect.

81. Control. In the original design the 120 control rods were divided into two groups, one of 12 safety rods and the remaining 108 for start-up and control. Later studies showed that as fuel burn-up increased the effective moderator temperature coefficient of reactivity decreased from its original negative value, and eventually would become significantly positive. With a core size somewhat larger than at Calder Hall there was also a probability that azimuthal instability of power distribution could occur due to xenon poisoning or asymmetrical changes of reactivity.

82. These two factors made it necessary to adopt a more sophisticated temperature control system, operating in nine discrete regions or sectors. In each of the eight peripheral sectors there are three temperature control rods, while the central sector has four such rods. The rods in each sector are positioned as a group by a controller which derives its input from twelve channel gas outlet thermocouples distributed within the sector.

83. Instrumentation. Apart from special instruments provided to measure initial start-up conditions, the most important permanent instrumentation of the reactor is concerned with temperature and flux measurement. There is comprehensive thermocoupling of the fuel elements, graphite moderator, channel outlet gas and reactor pressure vessel.

84. To control azimuthal flux instabilities which can develop after prolonged irradiation of the fuel, two additional thermal columns were installed symmetrically around the equator of each reactor vessel. Outputs from detectors in all three thermal columns are used to drive the reactor shutdown system in the unlikely event of excessive flux disturbances at any power level from zero up to the full design value.

85. The instrumentation gives information to the operators for control of the plant on a minute-to-minute basis, and also for reactor performance analysis. The 150 point high speed temperature scanner has become an operating tool rather than, as was originally intended, giving a historical record.

Commissioning

86. Each reactor at Bradwell is designed to give 150 MWe sent out. This required a heat generation of 531 MWth.

87. The Bradwell Power Station was synchronized on 1 July 1962. The commissioning pattern was as follows:

(a) Mechanical proving tests, following completion of construction

This occupied 4 months. A number of troubles came to light during hot testing, arising from temperature effects, particularly on the machinery for charging fuel on load. The first assessment of gas leakage on the first reactor indicated a loss of nearly 2 tons per day. This was reduced to 0.65 tons per day of which 0.1 ton per day could be accounted for. Nuclear phase commissioning, which followed the mechanical testing, extended to 42 weeks, twelve of which were spent in plant modifications and further mechanical testing;

(b) Fuel loading phase. This entailed the loading by hand of some 21 000 fuel elements and the recording of the position in the core of each individual element. The reactor was "loaded to critical" in a number of stages in which only seven, out of a possible eight elements, were loaded initially in each channel. After the critical size was determined, further fuel channels were loaded and a series of experiments was carried out to determine lattice constants. During the period of loading to full size a series of approaches to critical experiments was carried out to make a preliminary estimate of the reactivity worth of the control rod system. The built-in reactivity was measured using the air poisoning technique and the results were in good agreement with the theoretical predictions;

(c) Control rod and absorber calibrations. Several absorber patterns were calibrated. These were chosen because of the uncertainty at the time as to whether the reactor would go to power with a strong or weak absorber sector rod system. The control rod calibrations fell into two parts,

   (i) Sub-critical calibrations were carried out using both "counting at constant pressure" and "rod drop" techniques; and

   (ii) Super critical calibrations being made for various rod positions by repeated balance and doubling time measurements using air poisoning to maintain criticality. The measurements obtained required a considerable amount of analysis in order to obtain a figure for the shutdown capacity of the control rods, and some difficulty was experienced in this analysis. It was found that the design estimate of the bulk rod calibration curves was in good agreement with that measured over the super critical range and gave a total worth in fair agreement with that measured;

(d) Adjustment of fuel element channel flows. While experimental control rod calibrations were being made the design organization was engaged in the correlation of the flux distributions measured during the absorber calibrations with those predicted using the design model. When an acceptable degree of correlation was achieved the designer predicted the reactor power distribution for the "equilibrium" conditions. From this power distribution the channel coolant flow requirements were calculated for every fuel channel and converted into the flow pattern required under commissioning conditions. The work of adjusting the orifice setting for each individual channel was carried out using a b ttery of anemometers for gas flow measurement. This work was carried out inside the vessel with the blowers running continuously with air at ambient conditions. Some difficulties were experienced arising from inconsistencies in anemometer calibration curves. In addition to the setting of the channel flow orifices experiments were made to investigate the flow patterns obtaining with one and more gas circuits isolated; it was found in fact that the mass flow pattern was almost independent of the number and combinations of gas circuits in use. During the period of orifice setting which took approximately seven weeks, final decisions were reached regarding the bulk control rod and sector control rod groups.

(e) Final mechanical proving tests. These ran concurrently with control rod system modifications and occupied four weeks. The final examination showed some loosening of the fixed blades in the root serrations and dislodgement of a few blades in one blower. The blowers were modified and were in action again within 60 days of showing up the defect. This delay allowed for the replacement of 30 bulk control rods by rods having a built-in secondary release device operated by rapid change in gas pressure. At the end of this period the reactor was raised to low power and the burst cartridge detection system satisfactorily proved.

(f) Raising to power. Power was raised to 10 MWth on 17 June 1962 and, after a few hours' running, a check on the integrity of the fuel elements was made. A small number of suspect elements were detected and discharged from the reactor. Trouble affecting the operation of the burst cartridge detection system has been experienced due to the decomposition products of protective coating, employed in the gas circuits, at the temperature between 210°C and 390°C. It was decided to raise the temperature rapidly with only a portion of the burst cartridge detection system in operation and to purge off the decomposition products at the end of this period. In order to reduce flux distortions during start-up the bulk control rod system was lifted in two groups, thirty equally distributed control rods being allowed to remain fully inserted in the core. The reactor was raised to 20% on 1 July and two out of three main turbines synchronized on the same day. After the first set had been synchronized, the reactor was shut down owing to excessive moisture in the gas system. The source of moisture was found to be leaks in a boiler which was then isolated. Generation was resumed 73 hours after shutdown. During the following two months leaks occurred on a number of occasions and it was shown that, provided leaks did not occur simultaneously, it was possible to detect and isolate them without taking the reactor off load. On two occasions the grid connection to the station was lost without adverse effect on safety or plant operation. The plant has been run with the maximum assessed fuel element can temperature restricted to 450°C. With this temperature power of over 90% of the designed electrical output has been achieved since August 1962 and the main turbo-alternator sets have been proved at their full rating. When this temporary temperature restriction is removed, it is intended to increase power to full designed value.

Operational performance

88. The careful and thorough approach to the preliminary testing and commissioning of equipment at the station has resulted in a very satisfactory initial period of operation. It is considered that as operation continues the performance achieved to date can not only be maintained but improved upon.

89. The over-all performance results up to 30 June 1963 are given in the table below.

Table 7

The Bradwell nuclear power station: Over-all performance results

	Units sent out x 10^6 kWh	Reactor availability %	Load factor	Maximum generation MWe (sent out)	Fuel irradiation maximum channel average MWd/t
No. 1 Reactor	1339	81	76.2	287	710
No. 2 Reactor		86.5			510

90. <u>Reliability</u>. In operation the reactors are very stable, even though the reactivity temperature coefficient has now reached positive values. No rapid restorative action has been necessary in the event of load changes, even when the automatic controls have not been in use. It has also been found possible to operate with the main turbines on speed governor control and to follow the slow reaction on the steam pressure without having to use the automatic control loop.

91. Two rapid failures of fuel element cans have been experienced on No. 1 Reactor and these have caused the loss of 40 hours of reactor availability. An examination of the first failure indicated that the burst was of a typical kind previously experienced by the United Kingdom Atomic Energy Authority.

92. The performance of the boilers, turbo-alternators, blowers and auxiliary machinery has so far indicated a full capability to meet their designed output.

93. The general radiation levels which exist under conditions of full power operation have been under constant assessment during the initial stages of operation and have been found lower than the estimated values.

94. The removal of absorber elements and latterly of fuel from the reactor at load has shown that the shielding of the charge/discharge machine is completely effective, and background radiation levels were negligibly increased.

95. <u>Flexibility of operations</u>. During the working up to power stage it has been necessary to shut down the station a number of times and on these occasions it has been noted that the reactors are flexible in operation and load can be shed or taken up quickly. The rate at which load can be changed is limited by the rate of change of fuel element and pressure vessel temperature, but these limitations are in practice matched by limitations on the turbines and boilers.

<u>Safety aspects</u>

96. The principles on which radiological protection is based are as follows: The ICRP and MRC recommendations were used as a basis and implemented by the CEGB. In their practical application to the design and operation of the station an endeavour has been made to reduce radiation levels to such an extent that, in general, the anticipated integrated dose to the "individuals employed in the process" would not exceed half the maximum figure recommended by the ICRP.

The staff permanently employed outside the reactor area in the turbine hall, workshops, administration block, etc., are not expected by the designers to receive a dose exceeding 1.4 rem/year. The estimated background radiation levels in permanently high radiation areas with reactor on load, such as boiler house roofs or main gas valves, is much higher (60 and 500 mrem/h respectively) and accessibility to these is restricted. These doses are not so high as to preclude inspection or manual operation in an emergency. Handling of irradiated fuel is controlled from an adequately shielded control room. With the reactor off load for inspection and maintenance, the highest radiation levels are expected to be of the order of 160 mrem/h. With the reactor on load the necessary inspection and maintenance can be adequately carried out. Failure of fuel hoist mechanism resulting in the irradiated fuel element becoming stationary in the interspace between the pile cap and the secondary floor would result in an estimated irradiation dose rate of 3 rem/h at the base of the machine. This high radiation level would decay rapidly and if the manual winding is required three minutes after the fault, the dose rate is expected to be 8 mrem/h.

97. Radioactive effluents and wastes. Shield cooling air is filtered prior to discharge at a height of 137 ft above ground level. The total flow per reactor is 216 000 cu ft/min. The annual integrated dose from the plume and direct radiation was designed to be below the recommended ICRP levels for members of the public living in the vicinity. The nearest dwelling is 250 metres from the effective point of emission of the plumes, and even assuming adverse meteorological conditions, the total dose is expected to be less than 0.5 rem/year, which is the maximum level recommended by the ICRP.

98. Liquid effluents are conveyed in a system of two concentric pipes so that, in the event of a leakage, they are contained within the outer system and cannot spread contamination. The effluent is treated or filtered so that only a minute amount of soluble activity is planned to be discharged into the station cooling water where it is thoroughly mixed and diluted before being discharged into the estuary. Cooling ponds are designed to take a full reactor charge of fuel elements. Sampling for analysis in the laboratory is planned to be relied upon for establishing the type and quantity of activity held in either primary or secondary hold-up tanks. In the case of solid radioactive wastes,

highly active items may be stored in holes in the biological shield or transported to specially designed underground storage facilities constructed close to the reactor buildings. Lightly radioactive items of a combustible nature can be incinerated in a closed cycle incinerator. The routes and methods of transporting radioactive wastes to their destinations are arranged in such a way that the doses to the personnel handling the items are within the safe limits.

99. Pressure circuit integrity. Pressure-containing components in the main gas circuits have satisfied Lloyds' Class I requirements, having been designed basically in accordance with British Standard Specification No. 1500, with appropriate additional allowances for the stringent conditions of reactor service. Careful selection of the materials of construction and a conservative choice of working temperature stress have been made to ensure ductile behaviour and the absence of significant creep during the design life of the reactor. In the highest temperature zone, creep is expected to be not more than 0.2% in 20 years and arrangements have been made to measure the displacements occurring between selected standpipes due to pressure, thermal expansion and creep during the reactor life. Each vessel was subjected to pneumatic test pressure of 1.5 times the design pressure before being accepted for service.

100. Various tests on pressure vessel material were conducted to determine the crack arrest temperature of the plate so that the possibility of brittle fracture can be excluded by maintaining the vessel above a minimum operating temperature. An additional safety factor was introduced by designing to the limitation based on the crack arrest temperature and not on the crack initiation temperature, which is much lower. Boilers and main gas circulators were designed, manufactured and tested to the requirements of British Standard Specification No. 1500 and Lloyds' Class I rules. The gas ducting, expansion joints and valves were all tested before erection.

101. Fuel charging and discharging. This is accomplished while the reactor is on full load and adequate protection is provided for shielding operating personnel and preventing risk of inhalation hazards. Maintenance of leak-tightness of the coolant circuit, avoidance of damage both to the pressure vessel and the fuel elements have been assured as far as possible during this operation. The table below shows the maximum rates of reactivity increase due to control rod withdrawal.

Table 8

The Bradwell nuclear power station: Maximum rates of reactivity increase due to control rod withdrawal

Continuous withdrawal of control rods	Maximum rate of reactivity increase
Of all bulk rods	0.0003% $\frac{\Delta k}{k}$/sec
Of safety rods	0.0001% $\frac{\Delta k}{k}$/sec
Of all sector rods	0.0001% $\frac{\Delta k}{k}$/sec on start-up (manual control)
	0.002% $\frac{\Delta k}{k}$/sec at power (auto control)

In an emergency, the electrical supply to all groups is interrupted and all the rods fall under gravity to shut down the reactor.

102. An increase in reactor power level resulting in an increase in moderator temperature will give rise to an over-all increase in reactivity if the fuel irradiation is such that the moderator temperature coefficient is greater than + 0.002% $\frac{\Delta k}{k}/^{\circ}C$. The flux divergence was analysed by Fourier methods and the control system was designed to control the fundamentals, the 1st, 2nd and 3rd azimuthal modes and the 1st radial mode by means of nine groups of control rods.

103. Safety circuits. Most of the detecting devices have been designed on the "Fail to Safety" principle, i.e. loss of power supply or general malfunction of the equipment will result in its operating to the trip condition. The safety protection has been triplicated and arranged to trip the reactor if two out of three safety circuits operate. Neither the operation of trip circuits nor the insertion of the control rods requires the guarantee of a power supply.

104. Tripping parameters fed into the safety circuits, particular to magnox reactors such as Bradwell, include negative rate of change of coolant pressure, and high fuel element cladding surface temperature.

105. Guaranteed supplies. In the event of failure of normal power supplies, there are certain items of equipment which, for safety of the reactor, will continue to operate automatically from an alternative power supply.

106. Fault and accident conditions. Initially the station is being operated so that in the case of the worst credible accident, which is considered to be the complete severance of a bottom gas duct from the pressure vessel, the sum of probabilities of individual channels exceeding the ignition point of magnox, assumed at 620°C, should be less than 0.01.

107. The following reactor fault conditions have been considered in detail:

(a) Failure of all blowers;
(b) Control rod withdrawal faults;
(c) Fire in a single channel of the core; and
(d) Loss of gas pressure and blowers.

In all these cases a thorough analysis of events and consequences has been presented followed by conclusions and recommended procedures.

108. Plant operation. The operation of the plant is governed by written operating instructions which include the operating limitations of the plant. These are:

(a) Station operating procedures;
(b) System operating procedures; and
(c) Plant item operating instructions.

109. Site emergency organization. There are arrangements for two warning stages, namely "standby" and "alert", together with a procedure whereby "standby" can be bypassed when necessary. A "standby" warning indicates that an incident has occurred which may lead to a site emergency. For an "alert" warning, indicating that a site emergency exists, all persons on site will obey instructions laid down in the Site Emergency Plan.

Fuel management

110. Fully-fabricated natural metallic uranium magnox-clad fuel elements are supplied to the station by the UKAEA. The fuel elements are manufactured at Springfield near Preston, Lancashire, and transported to the station by road. Arising from batch production of elements for several stations, about three months' supply of fuel for each reactor (approximately 20 tonnes) is stored in fuel stores at the power station.

111. Each fuel element is uniquely identified by number and the fuel records system is arranged so that the location of each element is known from the time it arrives on site. The system also allows the identification of fuel elements in any position to be found. The records system is maintained by using punched card machinery capable of giving data print-out sheets on demand.

112. When fuel elements are discharged from the reactor they lose unique identify and are stored in the cooling pond in groups according to their irradiation. After a cooling period of approximately three months the irradiated fuel is returned by railway to the UKAEA processing plant at Windscale.

Training of operating staff

113. The CEGB decided to recruit senior staff at an early stage and give them special training in all facets of nuclear power generation.

114. The basic qualifications required from the technical personnel selected for training were:

(a) Sufficient academic knowledge to enable unfamiliar work to be understood;

(b) Adequate operating experience of modern power plant; and

(c) Experience in labour control in varying degree according to prospective appointment.

115. All those selected had some appropriate professional qualifications. The manual staff had no experience of reactor operation and, when required, training was given by the station technical staff. All station staff received instruction in general health physics, the amount of detail being dependent on a man's prospective employment.

116. General training. The present pattern of training falls into four main divisions:

(a) A course of theoretical training undertaken at a college of advanced technology, where the lectures cover nuclear physics, chemistry, metallurgy, health physics, heat transfer, reactor materials, nuclear instrumentation and control. The duration of this course is nine weeks;

(b) A course at which the knowledge acquired at the college can be further extended to reactor technology. This is taken at the UKAEA reactor school at Harwell and lasts ten weeks;

(c) An "operation" course of five weeks' duration held at the UKAEA School at Calder Hall. The course is designed to impart to the student the knowledge and experience which has been gained during the operation of the UKAEA magnox reactors. A reactor simulator is installed as an aid to instruction. Demonstrations and, where possible, participation in various operational procedures are arranged at Calder Hall and Windscale plants; and

(d) An on-job training period which varies according to the experience of a particular individual and the nature of the job for which he is being trained. It involves informal instruction in special aspects of nuclear power station operation and the trainees are encouraged to take an active part in the running of some of the CEGB and AEA plants such as Calder Hall. Further experience is gained by the employees by taking part in commissioning of plant in its various stages.

117. The above courses are flexible and are considered to be the maximum any individual is likely to require. For specialists they can be curtailed according to experience. There is not much difference between training programmes of senior and junior grades, since juniors must be regarded as candidates for promotion.

118. The manual personnel are generally trained on site. A formal course of lectures is given by the technical staff and each lecture is coupled with on-the-job training afterwards. In addition, maintenance personnel receive specific courses of instruction by equipment manufacturers.

119. <u>Simulator training</u>. A reactor simulator has been used to train operating staff in normal and fault condition operation. This was required because of the limitation of opportunity to train staff on the reactors during base load operation. Training on the simulator is comprehensive and includes reactor controls, as well as the essential instruments and controls of the associated boiler and turbo-generator plant. This is a valuable aid in preparing operating engineers to deal with all foreseeable circumstances such as

different techniques of operations which can arise as a result of factors which change as fuel irradiation proceeds. In addition they can be given experience of dealing with fault conditions which cannot be deliberately imposed upon an operating reactor.

Operating staff and responsibilities

120. The staff employed at the station constitutes a self-contained unit which is capable of carrying out the necessary operation, maintenance and administrative work. No contract employment is used on a regular basis. If clerical, administrative, estate services and maintenance work were contracted out it is estimated that the total staff could be reduced by more than 100 persons.

121. The station superintendent, or, in his absence, the deputy station superintendent, is responsible for the safe operation of the plant. The operations superintendent and maintenance superintendent are responsible to the station superintendent for operation and maintenance respectively. The shift charge engineers are responsible for minute-to-minute operation of the plant.

122. The health physicist is responsible to the station superintendent for all radiological and toxicity measurements and records and for giving health physics advice. He organizes and maintains records of station and district surveys, radioactive effluent measurements, personnel radiation exposure measurements, film badge service, manning of selected change rooms, shift monitors and the laundry.

123. The reactor physicist is responsible to the station superintendent for technical advice on the fuel and absorber charge/discharge programme and for ensuring that predetermined operating conditions of the reactors are satisfactory. He is also responsible for keeping full records of every fuel element that enters the station site. From these records it is possible to ascertain the position of any given fuel element or identify which fuel element is in any given position.

124. The chemist is responsible to the station superintendent for all chemical services in the station and for advising on chemical matters.

125. The present staffing plan is given in the table below.

Table 9

The Bradwell nuclear power station: Staffing plan

Category		Number of persons
Management		
Station superintendent		1
Station deputy superintendent		1
Operation		
Operation superintendent		1
Shift charge engineers	5 cycle shift system	5
Assistant shift charge engineers	5 cycle shift system	30
Assistant engineers (reactor, testing, efficiency shift relief)		9
Shift foreman	4 cycle shift system	12
Shift operators	4 cycle shift system	126
Records clerk		1
Maintenance		
Maintenance superintendent		1
Mechanical maintenance engineers		3
Electrical maintenance engineers		3
Instrument maintenance engineers		5
Assistant engineers		3
Foremen		3
Craftsmen (electrical, mechanical, instrument, welders, painters, joiner, bricklayer, insulating)		74
General labour		70
Records clerks		3
Reactor physics		
Reactor physicist		1
Assistant physicists		5
Clerk		1
Health physics		
Health physicist		1
Assistant health physicists		3
Qualified nurse		1
Foreman		1
Health physics monitors		22
Change room attendants		8

Category	Number of persons
<u>Chemistry</u>	
Station chemist	1
Assistant chemists	4
Laboratory assistants	3
Labourer	1
<u>Administration</u>	
Station administrative officer	1
Assistant personnel (wages, costing, accounts, clerical, services)	6
Station warden	1
Canteen supervisor	1
Storekeepers	4
Clerks (including typists and telephonists)	12
Security	6
General labour (gardeners, transport, canteen, cleaners and labourers)	19
TOTAL	453

<u>Cost data</u>

126. The estimate of the construction cost made in 1962 is given in the table below.

Table 10

<u>The Bradwell nuclear power station: Estimated construction costs</u>

Item	Estimated construction costs in £ million
Land	0.034
Civil engineering	12.321
Reactors	28.511
Turbo-generator sets	4.582
Other mechanical plant	2.673
Electrical plant	2.173
Switching station	0.306
Sub-total	50.600
Engineering	0.900
TOTAL	51.500

127. From the above total the following costs are calculated:

(a)	Capital cost of station - total construction cost (excluding interest during construction and cost of initial fuel charge)	£172/kW station output
(b)	Costs of generation - capital charges	0.80d/kWh
	- running costs	0.32d/kWh
(c)	Total cost of generation	1.12d/kWh
(d)	Approximate cost of initial fuel charge	£8¼ million

128. The above approximate costs of generation were based on the following considerations:

(a) Load factor of 75%;

(b) Useful life 20 years;

(c) Interest rate 6% per annum;

(d) Fuel burn-up 3000 MW days per tonne of uranium;

(e) Capital charges:

(i) Cover interest at 6% per annum, depreciation and an addition of 1.5% per annum to meet the requirements imposed by the Government on the Electricity Supply Industry to earn a gross return of 12.5% per annum on its average net assets; and

(ii) Relate to (a) the capital cost (construction) of station plus interest during construction and (b) the initial fuel charge;

(f) Running costs cover fuel replacement cost, fuel hold-up cost, operation, repairs and maintenance, management and insurance;

(g) The amortization of the initial fuel charge is included under "capital charges" and not under "running costs". The "running costs" do, however, include an allowance for interest on the reserve stock at the station and for the irradiated fuel held at the cooling ponds. The figure is extremely small and has a negligible effect when "running costs" are quoted to two places of decimals;

(h) Bradwell has only recently started operation. The costs of generation are based on planning data and do not necessarily represent current actual costs of generation;

(i) No allowance is made for the residual value of the final charge in the reactor; and

(j) No credit has been allowed for irradiated fuel, the assumption being that the value of the plutonium produced will be offset by the reprocessing and transport costs.

Integration into the power system

129. Utilization. Under the present pattern of generation, transmission and consumption of electricity when it proves economically attractive there is room for considerable additional nuclear generating capacity in the C.E.G.B. system. It has been calculated on the basis of a study of load and plant characteristics that the proportion of nuclear plant could approach 30% of the total before the annual load factor of the next increment falls below 70%.

130. Siting. The geography of the coalfields in relation to the pattern of electricity consumption is a major factor in the siting of the nuclear stations. Although distances in the UK are not great, the average delivered cost of heat in coal at present is some 35% higher in the London area than in the Midlands. The Thames area and the South West zone together account for nearly 40% of the total electricity consumption of the country and they have negligible coal reserves. Therefore, the nuclear stations are being sited mainly in the south of England and will progressively supply more of the base load in these areas where delivered coal costs are high. Collectively, the siting requirements of nuclear power stations are so exacting that the desired conjunction of physical characteristics is geographically rare in England and Wales.

131. Most intractable technically is the need for very large volumes of cooling water. The use of magnox-canned fuel elements in the nuclear stations so far planned limits the steam cycle temperature to 715°F and entails high exhaust-heat rejections. Therefore, the cooling water requirements are about 60% greater than those of comparable contemporary coal-fired plants. The largest English rivers have insufficient dry-weather flows for direct cooling of the smallest nuclear power station in the programme, and consequently coastal or estuarine sites are preferred.

132. Because of the design features of the nuclear stations, notably the large pressure vessels, much more space has to be allowed for site fabrication than in the case of coal-fired stations. Not all of it is however required for permanent buildings.

133. Foundation conditions are important because of the heavy load under the reactors.

134. The choice of site is further restricted by safety considerations. Thinly populated places are sought to promote safety in the remote event of an accidental release of radioactivity.

135. As facilities for bulk fuel movement are not required at nuclear stations, it is not essential to have railway connections to the sites.

136. Technical siting requirements, as well as economic considerations, make it desirable to provide for the development of individual nuclear sites to the limit of their capabilities and in some cases over 2000 MW can be envisaged.

137. Bradwell, at the north of the Blackwater Estuary, proved to be the nearest site to London offering a reasonable conjunction of facilities for a nuclear station.

Development potential and limitation of magnox reactors

138. The chief characteristics of the reactor system are that it is CO_2-cooled, graphite-moderated and fuelled by natural uranium rods clad in an alloy of aluminium and magnesium (magnox). The merits of this system are that these materials are relatively cheap and readily available, that enrichment of the fuel is unnecessary and fuel elements are relatively cheap to fabricate. On the other hand, the capital costs per kW of output capacity are high. The system is well suited for use in large base load stations. However, because of the high capital costs which increase further with decreasing capacity, reactors of this type in sizes of less than 100 MWe would probably be uneconomic.

139. This reactor system is a development of the Calder Hall system. Its main limitation in its present form is the relatively low ignition temperature of the magnox cladding of the fuel elements. This again imposes limitations on the temperature of the steam cycle and makes the station over-all thermal efficiency slightly lower than that of conventional stations. It also limits the rate of heat removal from the fuel elements and therefore contributes to the relatively high capital costs per kW.

140. Within the limitations imposed by the fuel cladding, substantial advances have been made with this reactor system from its application in Calder Hall to the latest magnox reactors under design and construction. Several factors have been contributing, but the most important technical factor leading to improvements in the economics of gas-cooled reactors has been an increase of gas

pressure. An increase in coolant pressure in a gas-cooled reactor is important because it brings about a proportional improvement in both heat transport and heat transfer performance in the reactor core and the boilers.

141. In addition to an increase in gas pressure, it has been possible to increase the outlet temperature of the coolant, though still keeping safe operating temperatures of the fuel elements. As a result of these two improvements, the electrical output from each tonne of fuel as well as the station over-all thermal efficiency have been increased from station to station.

142. The increases in these important parameters from Calder Hall to the latest magnox stations are shown in the following table.

Table 11

Important parameters for UK magnox stations

Item		Calder Hall a/	Bradwell	Berkeley	Hinkley	Hunterston	Trawsfynydd	Dungeness	Sizewell	Oldbury
Year of commissioning first reactor		1956	1962	1962	1963	1964	1964	1964	1965	1966
Coolant pressure	psia	100	147	140	200	165	254	284	279	365
Coolant outlet temperature	C^{o}	336	390	345	375	402	399	410	410	410
Net electrical output per ton of fuel	MWe/t_e	0.266	0.628	0.595	0.69	0.654	0.854	0.905	0.901	0.955
Station over-all thermal efficiency	per cent	19.0	28.2	24.6	25.7	28.8	28.9	32.7	29.7	33.6
Net output of station	MW	138(a)	300	276	500	328	500	550	580	560
Estimated running costs	d/kWh/s.o.	n.a.	0.32	0.38	0.33	n.a.	0.33	0.23	0.24	0.22
Estimated total generating costs	d/kWh/s.o.	n.a.	1.12	1.25	1.02	n.a.	0.97	0.74	0.73	0.73
Estimated capital costs	£/kW/s.o.	n.a.	171.7	176.7	142.5	n.a.	135.7	111.4	106	111.6

a/ Design value, present net electrical output exceeds the design value by approximately 35%.

143. It will be seen that the gas pressure at Oldbury is more than double that at Bradwell, even higher pressures being postulated for highly rated versions of the magnox reactor. The outlet temperature of the coolant, the net electrical output per ton of fuel and the station efficiency have continued to rise but at a decreasing rate. The increased performance of the magnox reactors has been achieved with a decreasing capital cost per kW. This is due partly to economies achieved as a result of increasing station output and partly due to real improvements in station performance. As a result the estimated total generating cost per kWh produced has continued to fall.

144. In the period covered by the table (1956-68) the unit cost of conventional power in the UK has also been decreasing but at a slower rate than that of nuclear power. Present estimates of nuclear and conventional generating costs per kWh made by the C.E.G.B. indicate that for base-load purposes nuclear power stations to be completed in the 1970's will be able to produce electricity as cheaply as conventional stations to be completed at the same time. The assumptions used in calculating capital charges for the purpose of these estimates are an interest rate of 6% and a load factor of 75%. In addition, the useful life of conventional stations is taken as 30 years whereas for nuclear stations the more conservative figure of 20 years is used.

145. The adoption of concrete pressure vessels in the Oldbury design has opened the way to further development of the magnox reactor system. The increase in gas pressure possible with the concrete vessel, and also its greater inherent safety, will enable the designer to propose much higher fuel element ratings without increasing the can surface temperature. In addition it is possible to contemplate reactor units of much greater output than those already under construction (1963) because larger concrete vessels involve no reduction in gas pressure and therefore no performance penalty.

146. Further economies by introducing sophisticated fuel cycles involving axial and radial shuffling of fuel elements are also being considered. It is hoped in this way to increase the mean fuel irradiation life from 3000 MWd/tonne to 4000 MWd/tonne.

Selected references

147. The following papers are suggested as providing suitable references to features of Bradwell Power Station and possible developments of the gas-cooled reactor systems:

VAUGHAN, R.D., Bradwell Nuclear Power Station - British Nuclear Energy Society's Symposium (27 June 1963).

PEDDIE, R.A., FARKASCH, G., RUTTER, R.L., Commissioning Programmes and Procedures (27 June 1963).

WEEKS, R.J., Operational Performance (27 June 1963).

EXLEY, J., Nuclear Fuel Management (27 June 1963).

GOTT, H.H., and TROTTER, R.D., Control and Safety Requirements for the Electrical Engineering of the C.E.G.B. Nuclear Power Stations, Institution of Electrical Engineers (17 April 1963).

Proceedings of the symposium on the advanced gas-cooled reactor, 14-15 March 1963. Journal of British Nuclear Energy Society, Vol. 2 (April 1963).

VAUGHAN, R.D., The Technical and Economic Development of the Gas-Cooled Reactor, World Power Conference, Sectional Meeting, Madrid (June 1960).

VAUGHAN, R.D., Panel discussion on development of the gas-cooled reactor: Proceedings of Anglo-Japanese Nuclear Power Symposium, Tokyo (26 March 1963).

D. THE UNITED STATES OF AMERICA

III. THE ELK RIVER POWER REACTOR

General

148. The Elk River power reactor achieved its first criticality on 19 November 1962. This project has been beset with several unforeseen problems, notably those concerning the pressure vessel. But the experience gained through overcoming the difficulties associated with the Elk River reactor has been most valuable and of great benefit to other projects.

149. After zero-power testing it was shut down to permit certain modifications in the reactor vessel internals. It became critical again on 31 May 1963 and, after going through 35 days of power testing, is expected to achieve full power by early September. This is to be followed by a 28-day warranty run by Allis-Chalmers and another 60-day power run for training the total utility personnel. It appears that the plant will be handed over to RCPA by the end of 1963 or early in 1964.

Critical experiment

150. Approach to criticality began on 14 November and fuel loading was carried out in 13 steps; the number of fuel elements added in any loading step did not exceed four. On 19 November, with 41 fuel elements in the core, the reactor achieved criticality.

151. Although no mock-up of the ERR core had been arranged previously, the measured core parameter checked fairly closely with the predicted values. Deviations were noted in certain instances and the interesting features described below came to light.

(a) The actual critical mass for the cold clean core was 40.4 elements versus 30 ± 6 predicted. The discrepancy is explained partly by the fact that in original calculations the effect of the stainless steel in the hollow can containing the plutonium-beryllium source was overlooked. It was later noted that the steel in the can, which had the configuration of a dummy fuel element, had an appreciable effect because of its location near the centre;

(b) The temperature coefficient was positive for the 45-element core, while it was negative for the 69- and 148-element cores. The explanation is that the 45-element core approaches the conditions of an unrodded core because of the small excess reactivity to be controlled and is also affected appreciably by the central water channel. In the 69- and 148-element cores, the 13-rod bank is inserted far enough to influence the over-all temperature coefficient of the core. In this case the central water channel also has lesser importance and the two factors combined together give a negative coefficient;

(c) During rod calibration it was found that for certain rod configurations it was possible to experience an increase in reactivity with control rod insertion. This results from the fact that the zircaloy followers are slightly shorter than the active core height so that when a rod is moved from the full "out" position into the core the displacement of water by the zircaloy follower in a more reactive portion of the core has a stronger effect than the initial poison insertion. Since the core is overmoderated for low temperature conditions, the net effect is a gain in reactivity for the first few inches of rod insertion. The total reactivity increase observed was about 0.04% $\frac{\Delta k}{k}$; and

(d) The maximum reactivity insertion rate for various rod configuration was determined to be 0.0012 $\frac{\Delta k}{k}$/sec.

Table 12

Elk River reactor: Results of critical experiments[a]

Item	Predicted value	Measured value
Critical mass	30 $\pm$ 6 elements	40.4 elements
Total excess reactivity	11.2% $\frac{\Delta k}{k}$	9.32% $\frac{\Delta k}{k}$
Total control rod worth	18.3% $\frac{\Delta k}{k}$	18.1% $\frac{\Delta k}{k}$
Worth of centre rod	3.40% $\frac{\Delta k}{k}$	3.16% $\frac{\Delta k}{k}$
Shutdown margin	2.00% $\frac{\Delta k}{k}$	3.48% $\frac{\Delta k}{k}$
k_{eff}	1.1265	1.1028

a/ Except for critical mass all values refer to 148 regular fuel element core.

Core optimization

152. Detailed studies were conducted to determine the optimum core loading taking into account the shutdown margin, the reactivity for burn-up and power distribution. It was decided to use a core with 128 regular 4.3% enriched elements with 20 spiked 5.2% enriched elements. It is expected that the addition of the spiked elements will extend the lifetime of the core from 6700 MWd/t to 9000 MWd/t. The additional reactivity provided by the spiked elements will also permit operation with slightly higher voids. The spiked elements have been so located that the average to peak flux ratio is actually decreased as compared to the regular core.

153. The k_{eff} for the spiked core is 1.117. The shutdown margin with the most reactive rod out and the remaining 12 rods inserted is 3.1% $\frac{\Delta k}{k}$.

System modifications

154. Reactor internals. The Elk River reactor is very similar to the EBWR. In January 1963 when the results of the operation of the EBWR at power levels up to 100 MWth became available, the ERR performance was re-evaluated in the light of the new data. It was concluded that just as in the case of the EBWR, steam carry-under into the downcomer, reactivity in the voids, operating water level and water carry-over to the evaporators might be significantly higher than anticipated. These misgivings were later confirmed by extensive analyses and concurrent air-water tests. Several possible modifications suggested by the ANL experience were considered and suitable tests performed by Allis-Chalmers Company to select the best scheme.

155. Consequently, it was decided to install a core riser (chimney) having the shape of an inverted cone pierced with holes, provide baffles over the riser and downcomer and raise the steam collection baffle to the top of the reactor vessel. Suitable parts were fabricated and installed in two months. In-core instrumentation for temperature and pressure was also added. It is expected that, as a result of these modifications, the steam carry-under will be kept to a sufficiently low level to permit achievement of rated thermal power level and fuel lifetime within safe operating limits.

156. Containment vessel. A safety review indicated that, in order to avoid the possibility of brittle fracture when the containment vessel is pressurized in cold weather, all pressure-containing parts of the reactor containment

vessel should be maintained at $30^{\circ}F$ above the nil ductility transition temperatures of the containment vessel materials.

157. Testing. Due to the long delay in the start-up, certain pre-operational tests were repeated before permission was obtained to reach criticality. In general, they proved satisfactory but the following points are worth noting:

(a) Purification system. In one of the tests boric acid solution was introduced in the system. However, during the system clean-up, one ion exchanger did not work properly. It was found that the resin used was not of the right type and it had to be replaced. The resin in the other one will also be replaced;

(b) Hydraulic system. Minor leakages were observed at certain locations. The pumps performed well;

(c) Control rods. These worked well with the exception of the position indicators which had to be adjusted; and

(d) Start-up heating system. Pre-operation testing indicated some difficulties in this system. An external heat exchanger (fed by steam from a conventional plant) and a piping loop connected to the reactor vessel is used to heat the primary water by natural convection. Since the lower piping loop connection to the vessel was above the bottom of the vessel, as the bulk temperature of the water in the vessel increased, cooler water tended to stagnate in the lower part of the vessel. This situation persisted until the bulk stagnant water temperature was approximately $125^{\circ}F$. At this point the cool water at the bottom rose rapidly to the top of the vessel thereby quickly cooling the upper part of the reactor vessel and vessel head surface. This resulted in unnecessary temperature stresses. To overcome this the recirculation loop piping was connected to a line which enters the bottom head of the vessel. Appropriate piping and valving changes were made which now permit the use of the decay heat cooling system during start-up heating operations.

158. Containment shell leak rate. The leak rate was tested in the second half of 1962 and found to be less than 1% of the volume/day. This test will be repeated every 18 months. All the penetration to the containment shell will be checked every three months.

159. Building spray system. This could only be checked by using air. When the pressure inside the containment shell reached 2 psi above the atmospheric pressure, the alarm sounded, which indicated that the system was working. The alarm had to be stopped within ten minutes to prevent automatic spray.

160. Pressure vessel. Upon the recommendation of ACRS, USAEC has put certain restrictions on the use of the Elk River pressure vessel by limiting its service life to five full-power years, or 250 pressure temperature cycles, whichever occurs first. These restrictions are not final inasmuch as they can be reviewed in the light of additional information. In order to define more accurately the cyclic life of the Elk River vessel and develop basic technical data useful for other vessels as well, USAEC has initiated a four to five year test programme at the South-West Research Institute which is divided into three phases:

(a) Phase 1. Studying of the fatigue properties of steels, weld deposits, and dissimilar welds typical of those employed in the Elk River pressure vessel;

(b) Phase 2. Studying of the influence of neutron irradiation on the nil ductility transition temperature and fatigue properties of steels and welds typical of those in the ERR pressure vessel; and

(c) Phase 3. Developing of reliable remote non-destructive testing procedures for critical locations in the ERR pressure vessel.

161. It is expected that by the end of this four to five year research programme enough information will be available to re-examine the future of the Elk River reactor pressure vessel and to extend its life further if the evidence so warrants.

Fuel management

162. In November 1962, USAEC and CNRN signed an agreement under which CNRN will, at its thorium fuel reprocessing facility in Italy, reprocess the Elk River core I fuel and refabricate it. Under this arrangement it is envisaged that:

(a) ERR fuel will be discharged in one-third lots and shipped to CNRN;

(b) The shipping casks will be provided by USAEC. Each cask will weigh approximately 30 tons empty and 31 tons loaded, and will contain 19-25 fuel elements;

(c) CNRN will reprocess the irradiated fuel elements and the refabricated elements will probably constitute the third core for ERR; and

(d) The total cost of reprocessing of the complete core and its fabrication by CNRN will amount to about US $1 750 000 including use charge; U^{235} make-up will be provided by USAEC.

163. It is tentatively planned that the second core will also consist of UO_2-ThO_2 and will be similar in design to the first core. The fuel elements for this core will be fabricated in the United States.

164. CNRN reprocessing facility for thorium fuel is currently under construction and is scheduled to be in operation by July 1964. It is being built with the help of Allis-Chalmers Company and has a capacity of 15-30 kg/day. It will also be used to reprocess fuel from the proposed Italian RAPTUS (Rapid Thorium Uranium Sodium) reactor which is planned to be completed in 1968.

Operation and maintenance

165. The total staff for the nuclear part of the plant has been increased slightly with the addition of one reactor specialist or shift supervisor and one instrument technician changing the earlier strength from 27 to 29. It does not include four to five persons required for maintenance of the conventional part, such as the turbo-generator plant and for administrative work, because these are available from the conventional unit in operation at the site.

166. In addition, a consulting engineering firm has been hired for the initial period to provide the necessary technical and advisory services at a cost of US $100 000 for the first year.

167. The annual operating costs of the plant, excluding nuclear fuel, cost of consulting engineering services and the salaries of the four to five persons also employed in the conventional plant are given in the table below.

Table 13

Elk River reactor: Estimated annual operating costs

Item		Cost US $
Staff		
Plant superintendent and assistant plant superintendent	23 000	
Shift and relief operators	64 000	
Shift supervisors	48 000	
Maintenance technicians	73 000	
Sub-total		208 000
20% overhead		40 000
Materials		
Coal for superheater	172 000	
Others	50 000	
Spare parts and tools	30 000	
Sub-total		252 000
TOTAL		500 000

Cost data

168. A report giving the final detailed cost breakdown is being prepared. Meanwhile, certain data relating to main cost categories and major components is available and is summarized in table 14.

Table 14

Elk River reactor: Cost breakdown

Item		Cost US $
A. MAIN COST CATEGORIES		
Direct costs		
Reactor equipment (including fuel, rods and drives, vessel, access and instrumentation)	1 295 000	
Process system equipment (including heat exchangers, pumps, tanks, instrumentation)	511 000	
Superheater	380 000	
Contingencies	188 000	
Sub-total		2 372 000
Engineering design and overhead		
Architect engineer	399 000	
Engineering and design	869 000	
General administration and overhead	2 041 000	
Sub-total		3 309 000
Construction, materials and installation		
Reactor plant structure	1 447 000	
Reactor	244 000	
Process system	569 000	
Superheater	483 000	
Indirect construction	971 000	
Sub-total		3 715 000
TOTAL (Contract price)		9 395 000

Item	Cost US $
B. MAJOR COMPONENTS[a/]	
Reactor vessel (including internals and supports)	212 000
13 control rods (including absorber section and tooling costs)	89 000
Control rod drives (14 units)	180 000
Control rod guide and shroud assembly	144 000
Intermediate heat exchangers (evaporators and coolers) delivered to the site	124 000
Superheater furnace delivered to the site	198 000
Fuel elements (148 regular, 28 spiked)	
Conversion cost UF-6 to UO_2 (excluding losses and use charge)	48 000
Pelletizing (excluding losses, use charge and scrap recovery)	103 000
Cost of thorium (4300 kg of ThO_2)	68 000
Fabrication (including cladding, inspection, assembling and packaging for shipment but not losses and rejects)	240 000

a/ Based upon purchase price, f.o.b. manufactuers' plant unless otherwise shown.

Selected references

169. A list of selected references concerning the Elk River power reactor is given below:

Operation of the RCPA Elk River Reactor, ACNP-63518, Allis-Chalmers Manufacturing Company, Washington, D.C. (March 1963).

FISCHER, J.R. and DIAZ, A., Experimental Evaluation of the fully-loaded Elk River Reactor, summary of a paper presented to the American Nuclear Society Meeting (June 1963).

Elk River Reactor, Thermal and Hydraulic Performance, ACNP-63539, Allis-Chalmers Manufacturing Company, Washington, D.C. (April 1963).

Elk River Reactor, Technical Specifications, Allis-Chalmers Manufacturing Company, Washington, D.C. (May 1963).

PURSEL, C.A., Post Construction Testing of the Elk River, Hallam and Piqua Power Reactor Plants, USAEC, Argonne, Ill. Abstracts for Conference on Operating Experience with Power Reactors, Vienna, 4-8 June 1963.

IV. THE BONUS POWER REACTOR

General

170. The BONUS power reactor is expected to become critical in January 1964 and reach full power by 15 May 1964. It is planned to hand over the plant to PRWRA by 30 June 1964. The project has suffered a delay of over one year. The delivery of the pressure vessel was put off by 17 months, six of which were made up by accelerating other construction work. The changes in fuel element design and its fabrication have also been responsible for postponement of criticality. As of June 1963 90% of the construction was complete and is expected to be finished by 15 November. After pre-operational tests, fuel loading will start on 1 January 1964.

System modifications

171. The superheater fuel element cladding has been changed from 348 ss to Inconel as discussed later in the report. The lifting lug on the fuel element has been removed and the arrangement for its handling modified accordingly. Shims have been added between the superheater and boiler regions in order to provide additional control during early operation of the reactor.

Start-up

172. The fuel loading is expected to begin in January 1964. Before this, all pre-operational tests relating to various systems will have been carried out. Important stages in the start-up programme are listed below:

(a) The initial criticality will be achieved on a 6 x 6 array of boiler fuel assemblies without shims. The fuel will be loaded dry and the water level will be raised at the rate of 0.5 inch/min, to reach criticality;

(b) The shim rods will then be added and the experiment repeated with 8 x 8 array of boiler fuel assemblies;

(c) The reactor will be tested by using start-up heater; temperature and void coefficients, as well as power distribution, will be measured;

(d) One superheater fuel bundle will be added and criticality attained. This will be repeated for four bundles;

(e) The superheater characteristics will be studied for unflooded and flooded as well as cold and hot conditions. The flooding coefficient with all superheater rods in is expected to be $-0.6\% \frac{\Delta k}{k}$ for cold and $+0.16\% \frac{\Delta k}{k}$ for hot conditions. With control rods in criticality position, this coefficient under cold condition is expected to be $+0.3\% \frac{\Delta k}{k}$;

(f) With full boiler core in place steam will be generated to test the equipment and steam quality. This may last from two weeks to two months; and

(g) The four superheater bundles will be put back. All the superheater control rods in the shim rods will be withdrawn to raise the steam outlet temperature to design conditions.

Construction experience

173. Although all the major components for the reactor came from the continental United States, Puerto Rico supplied most of the skilled labour needed for building the plant. Only four highly trained stainless steel welders had to be brought from outside for two months to supplement the local force. It is estimated that the total number of man-hours needed for the USAEC portion of the BONUS plant construction will be about 657 000. The average number of men working on site has been 55 and the maximum was 155.

174. Containment shell. It took about nine months to put up the containment shell. The contractor came with a weld inspector who set up a training school for 14 local welders. One week of training was needed for each of the welders who had some basic experience before.

175. The carbon steel plates were pre-shaped and 10% of the welds were X-rayed. Stiffeners had to be added to strengthen the shell to withstand 5 psig designed pressure. The leak test was performed at 6.25 psig. The shell held 5 psig pressure for three consecutive days. The leak rate observed was 0.1% vol/day as against 0.2% vol/day designed. The leak test will be repeated at the end of the construction.

176. Concrete work. No problems were encountered in this work. Except for one supervisor, all foremen and labour were local. Mostly ordinary concrete was used except for isolated areas, where steel punchings were used. In all, 20 000 cu yds of concrete were poured.

177. Component installation. Except for the supervisors, all work connected with pipe-fitting and installation of components was carried out by local craftsmen. The X-ray technicians and specialists for reactor internals and control rod drives came from outside. All non-nuclear electrical work was performed by Puerto Rican personnel. The nuclear instruments were calibrated and tested by the technicians supplied by the manufacturers.

178. The components causing major delays were the reactor vessel (late by 17 months), instrument package (late by 6 months) and the emergency condenser. Certain reactor internals were modified and their delivery was put off.

179. Pressure vessel. As a result of experience with the Elk River and BONUS pressure vessels, it appears that this component presents several potential problems. One of the complicating factors is that the codes used so far for such vessels have been inadequate and cannot provide sufficient guidance to the fabricator to meet the stiffening requirements imposed by latest methods of inspection. As a result, the ASME code is undergoing a revision and one of its sections will specifically refer to the nuclear reactor vessels clearly stating the standards which have to be met. This will enable the fabricators to quote more realistically on these pressure vessels and make sure that their products can meet the latest inspection criteria.

180. The fabrication and delivery of the BONUS pressure vessel took 29 months instead of the expected 12 months. Many factors were responsible for this delay, including an increased emphasis on perfection because of the difficulties experienced with the Elk River vessel which was also fabricated by the same vendor. Many micro fissures were discovered in the cladding overlay in the fabricator's shop. The twin-arc oscillation method was used for overlay and the overheating led to greater penetration into the base metal than allowed. The high standards of inspection used revealed "inclusion" at many places. It also appears that the carbon steel employed was of ordinary quality and had many impurities which were responsible for certain troubles. The refined ultrasonic tests revealed defects in nozzle welds which had to be redone.

181. The pressure vessel weighed 61 tons. Although the site is on the sea-shore, the vessel was not shipped by sea because no large crane was available for handling it. Instead it was transported from San Juan by road and took two days to reach destination. The actual installation of the vessel took nine days and required four men.

Fuel

182. The cladding of the BONUS superheater fuel has been changed to Inconel from 348 ss. This change has been dictated by the experience gained with the test conducted at VBWR, which indicated several cracking failures with 304 ss clad elements. These cracks are caused either by chloride stress corrosion or intergranular corrosion because of carbon content. Although 348 ss appeared to be better than 304 ss, it was not considered safe enough. Repeated tests with high nickel alloys, Incoloy, Hastelloy X and Inconel have shown that they stand up much better in the superheater environments. Nevertheless, these alloys may not provide the ultimate answer because they tend to lose ductility at high temperatures and become brittle. No fabrication problems are envisaged with Inconel (or other high nickel alloys) because tubes can be made from this material to close tolerances.

183. The Inconel superheater core is expected to cost US $910 000 for two short tons of fuel corresponding to US $491/kgU and US $25 300 for one fuel assembly. The expected burn-up for the 348 ss clad superheater fuel was 10 000 MWd/t. For the new Inconel-clad elements, it will be about 7500 MWd/t. It may be mentioned that this burn-up could have been higher by using higher enrichment, but it was too late to change the enrichment.

184. The boiler fuel fabricator experienced some difficulties in welding the end cap to the thin tube. The original welding method led to thinning of the tube wall. Later on, with the development of an improved technique, this difficulty was overcome.

185. For the future, the General Nuclear Engineering Company is conducting studies for the development of an advanced type of superheater fuel element which might be considerably less expensive. It will eliminate the need for a pressure tube and instead allow boiling on the outside. One concept of the multitube elements consists of a zircaloy outer tube with small Inconel tubes spaced uniformly inside. UO_2 fuel will fill the space between the tubes by vibratory compaction. The water will boil on the outside and the steam will be superheated while passing through the inner tubes. This concept has the advantage of higher neutron economy but presents several design and material problems. The development of the advanced superheater elements for BONUS may cost over one million US dollars.

Cost data

186. There has been an increase in the BONUS project costs due mainly to an increase in engineering design and fuel fabrication changes. The latest estimates are given below by category.

Table 15

The BONUS power reactor: Cost estimates (January 1963)
(in thousands of US dollars)

Category		Cost
Plant construction		
Engineering design and inspection		
Title I, III and IV design		2 350
Start-up		380
	Sub-total	2 730
Direct construction costs		
Power plant building		2 300
Reactor plant equipment		3 707
Accessory electrical equipment		186
Miscellaneous power equipment		163
	Sub-total	6 356
Indirect construction cost		1 021
Contingencies		293
	TOTAL	10 400
Operating expenses		
Research and development		1 549
Operator and start-up training		354
Fuel fabrication (first core, reserve and shipping)		1 419
Contingencies		178
	TOTAL	3 500
	TOTAL USAEC costs	13 900
	TOTAL PRWRA costs	5 044
	TOTAL BONUS project costs	18 944

187. The costs of certain major components are given in table 16.

Table 16

The BONUS power reactor: Costs for selected components

Component	Cost US $
Reactor pressure vessel	423 000
Water treatment equipment	202 000
Control rod drive system	250 000
Control rods	94 000
Boiler and boiler superheater shims	77 000
Instrumentation, controls and board	600 000
Ready-mix concrete	280 000
Piping and equipment installation	565 000

188. The estimated construction costs are US $7 480 000, of which the labour costs are nearly 27%, or US $1 697 000.

Integration

189. The BONUS reactor will be a part of the Puerto Rico grid which already has 600 MWe of installed capacity and is growing at the rate of 15% per year. It will be tied to the system by two 38 kV lines. The reactor is expected to operate as a base load station and as such will stay independent of the system load at any time.

190. PRWRA, which will operate the reactor, consider this unit as a demonstration and training project. For the next ten to fifteen years, it is felt that relatively large thermal stations in the range from 200-250 MW will be needed. If it is found that nuclear power competes favourably in this range and at the current oil prices of 35¢/million BTU, then it will have a place in the future power development programme of the island.

191. Meanwhile, this project is providing excellent experience in, and orientation towards, nuclear power technology and it is proposed to establish a training centre for nuclear power.

Staffing and organization

192. The final staffing plan given in table 17 shows a total strength of 53 instead of 38.

Table 17

The BONUS power reactor: Staffing plan

Category	Number of persons
Administration	
Nuclear plant superintendent	1
Assistant nuclear plant superintendent	1
Administrative assistant	1
Clerical	3
Warehouse men	2
Operation	
Nuclear shift supervisors	6
Reactor and plant operators	8
Relief operators	4
Health physicist	1
Health physics technician	1
Supporting technical personnel	
Performance engineer	1
Plant procedure engineer	1
Nuclear plant engineer	1
Nuclear plant physicist	1
Plant chemist	1
Maintenance	
Instrument supervisor	1
Instrument technicians	2
Technical chemists	2
Electrical supervisor	1
Electrician	1
Mechanics	2
Utility men	2
General helpers	6
Painters and janitors	3
TOTAL	53

193. In addition, two committees have been provided to help and guide the operating staff. These are the Plant Technical Committee and the Reactor Safety Committee.

194. The Plant Technical Committee is responsible for analysing, studying and advising on unusual operating problems. It has four members, three senior engineers of PRWRA from outside the plant, and the BONUS plant superintendent. The committee will retain a nuclear physicist as a part-time consultant.

195. The Reactor Safety Committee, consisting of six members, is an advisory committee to review the operation of the plant, study all procedures and recommend any changes required for safe operation of the plant. The members of the committee, who will be recruited from outside the BONUS operating staff, are required to be familiar with the plant design. Outside consultants will be appointed as necessary. During the start-up period the committee will help review the preparation and performance of all tests.

Selected references

196. A list of selected references concerning the BONUS power reactor is given below:

Bonus Hazards Summary Report 1963, PRWRA-GNEC-5.

Bonus Design Report 1963, PRWRA-GNEC-6.

BEVILACQUA, F., Bonus Power Station Technical Specifications, GNEC-214 (1962).

GIBBONS, J.F. and JAMESON, A.S., Design and Fabrication of the Bonus Pressure Reactor Vessel, GNEC-210, Supplement 2 (1962).

Proceedings of the Nuclear Superheat Meeting No. 8, Chicago Operations Office, USAEC (April 1963).

Proceedings of the Nuclear Superheat Meeting No. 7, Chicago Operations Office, USAEC (October 1962).

V. THE PATHFINDER POWER REACTOR

General

197. The latest schedule for the Pathfinder power reactor envisages criticality towards the end of 1963 and full power operation by the end of 1964. This represents about one year's delay as compared with the timetable reported last year[4/]. This delay has been caused mainly by fuel fabrication problems and the licensing procedures. The plant is essentially complete and a small staff of construction personnel is at the site to carry out minor modifications.

198. By June 1963 36 boiler assemblies had been completed and 55% of the rods were ready. The superheater fuel went into production in July 1963. Since initial criticality only boiler fuel will be necessary and no postponement is expected. The reactor manufacturer is confident that the chloride stress corrosion problems in the Pathfinder reactor can be overcome by proper water chemistry and moisture control. Moreover, the stress corrosion in the case of highly enriched cermet superheater core is not expected to be serious. Even if there is some corrosion cracking the UO_2 will not release any appreciable amount of fission products. At the present time there is still no full understanding of the stress corrosion phenomenon and research and development work on this is continuing. In this connection, practical experience with the Pathfinder reactor will be valuable.

Design and system modifications

199. The value of the flooding coefficient is now expected to be $-0.3\% \frac{\Delta k}{k}$. The unflooding of the superheater will add 0.3 to $0.4\% \frac{\Delta k}{k}$.

200. The superheater will have 12 finger-type control rods instead of 13 cruciform rods as reported before. Boron will be used as the poison material.

201. The nuclear instrumentation design has been changed. Originally the pre-amplifiers were located in the chassis in the control room and there was too much noise in the cable. Now these pre-amplifiers have been relocated in the reactor building closer to the nuclear detectors. The entire circuitry is transistorized and certain transistors have been replaced to make the circuit less sensitive to temperature variations.

4/ See document GC(VI)/INF/54, para. 37.

202. A spray ring has been installed in the reactor vessel for assuring proper cooling of the fuel elements in case of emergency.

203. A cross-connect line has been provided between the cooling tower water and the condensate hot well so that the large volume of water in the cooling tower can be used to keep the reactor flooded if a leak in the primary cooling system should develop. This line is equipped with two motorized stop valves in series, with a leak off between them, to assure complete isolation of the condensate and cooling tower water during normal operation.

Start-up and testing

204. The period of approximately one year between initial criticality and full power operation will be used to conduct a large number of tests to check the nuclear characteristics of its superheater-boiler core and the performance of the system as a whole. This testing period appears to be somewhat longer than usually earmarked for other reactor plants. The reason, however, lies in the fact that the Pathfinder reactor is the first one of its type and a very careful approach to full power is warranted because of the incorporation of a central superheater section. There is no doubt that the time spent on carrying out the tests will yield valuable information of use not only for this plant but for the designing of larger nuclear superheat central stations as well.

205. The tests have been divided into three phases. A brief description of the principal tests in each phase is given below:

Phase A (Power: 10-200 kWth; duration: five to six months)

(a) Assembling slab core made of ten boiler elements. Flooding the core and achieving criticality by rod withdrawal. Checking out and confirming the nuclear parameter of the core. Training personnel in loading, unloading and handling of fuel;

(b) Loading all 96 fuel elements and making the necessary measurements with different rod configurations;

(c) Putting all the boiler shims on, then loading all the superheater fuel elements. Approaching criticality. Keeping the shutdown margin at 10% $\frac{\Delta k}{k}$. Withdrawing the shim rods step by step till all are out;

(d) Establishing the reference core. It is defined as the core which is just sub-critical with the most reactive control rod out and the superheater voided;

(e) Measuring the flooding coefficient in the cold condition. It is expected at about $-(0.3 \text{ to } 0.4)\% \frac{\Delta k}{k}$. Actually the core is most reactive when the superheater is voided;

(f) Preparing a flux map of the core in the axial and radial directions for seven different boron solution concentrations. In the extreme condition all the control rods will be out and the reactivity will be held down by boron solution only. These measurements will help establish the superheater power fraction, axial power distribution and core symmetry;

(g) Cleaning the core and performing refuelling tests to evaluate the worth of boiler elements to help establish the optimum fuel shuffling programme;

(h) Pressurizing the core to measure the pressure coefficient; and

(i) Heating the core with a start-up heater. Measuring the temperature coefficient and the flooding coefficient of the hot core.

<u>Phase B</u> (Power: 200 kWth to 5 MWth; duration: four months)

(a) Increasing the power to 5 MWth with the superheater flooded. Calibrating the instruments;

(b) Draining the superheater and gradually increasing the reactor power in steps. At each step measuring the superheater radiation cooling ability and the maximum fraction of the full power which the superheater can dissipate without steam flow;

(c) Establishing steam flow to the superheater and increasing reactor power to 5 MWth with the superheater power generation suppressed by control rods. Measuring the superheater fuel and bulk steam temperature; and

(d) Testing the performance of the emergency condenser, the process system and the reactor protective system.

Phase C (Power: 5 MWth to full power; duration: four months). During Phase C approach is made from 5 MWth to full power in five steps. At each step the following tests are conducted:

(a) Making power calibrations of nuclear instrumentation by thermodynamic measurements;

(b) Performing radiation survey of the plant, measuring dose levels at various places and checking the validity of the shielding calculations;

(c) By passing steam through the nuclear superheater, measuring the temperature and flow rates of steam and fuel surface temperature;

(d) Measuring reactor response associated with changes in each of the following variables: feedwater temperature and flow, reactor pressure and re-circulation flow;

(e) Evaluating the response of the reactor to rod "runback";

(f) Performing transfer function tests using a special fuel assembly containing an oscillating poison rod;

(g) Determining transient xenon reactivity by reducing reactor power level and observing xenon build-up after shutdown;

(h) Carrying out water level calibrations; and

(i) Measuring the efficiency of the steam dryer for various steam rates.

206. After satisfactory completion of the foregoing tests, the reactor may be run in a routine manner. For the initial period it will be started and shut down frequently to help train additional staff. Various supplementary tests will be carried out in crude activity build-up, and off-gas decay and performance tests may be made of mechanical equipment which are not possible to run during the pre-operational check out.

Reactor operation

207. The reactor is operated as a constant pressure system. To increase reactor output, control rods are withdrawn, which tend to increase power output and the reactor pressure. The pressure control system then admits more steam to the turbine. Since constant water level in the reactor has to be maintained, more water is allowed to come in. This provides the extra steam needed for higher output.

208. The reactor is not a load following plant and has been designed for constant load duty. All load changes are initiated at the reactor and the turbo-generator follows it.

209. If the reactor is shut down for a few days and it is desired to put it on the line, it would take about nine to ten hours to do so as compared to about six hours for the conventional plant of comparable size owned by the Northern State Power Company. This shows that in this respect the nuclear plant compares well with the conventional station. Three hours would be needed for going through the pre-start-up check list; four hours for reactor heat-up at the rate of 150°F/h; about half an hour for going critical; and two hours for warming-up to full load (including draining the superheater, establishing steam flow and exit temperature). The load will be picked up at the rate of 2 MWe/min.

210. Under normal circumstances, to shut down the reactor the power level is first reduced to 9 MWe. The plant is disconnected from the NSP system and the power is further reduced to 6 MWe and the superheater control rods are fully inserted. The generator is taken off the line and the turbine is shut down. Before scramming the reactor the water level is raised to 3-4 ft above normal. A steam flow of 20 000 lbs/h is maintained through the superheater and is by-passed to the condenser. After the superheater has cooled off sufficiently, it is flooded and the reactor is cooled by circulation through the purification system and feed-water system, and the temperature falls to 160°F or less.

Power split

211. Maintenance of proper power split between the boiler and superheater sections of the core is important. In this reactor the superheater rods will normally be in the all-out position during power operation. The K_∞ of the superheater section is less than one and it cannot go critical by itself. Since all superheater rods are out, its output is regulated by the adjacent boiler rods. The superheater therefore follows the boiler. At full power of 188.9 MWth, the boiler output is 157.7 MWth and the superheater 31.5 MWth. At all power levels the superheater output is kept to 16.7% of the total.

212. The outlet steam temperature from the reactor is regulated by the feed-water inlet temperature. Normally the reactor is started with the maximum feed-water temperature and its temperature is then brought down to obtain a 725°F steam outlet condition.

Reactor control

213. The design of reactor control is intended to protect the reactor system against all malfunctions and, at the same time, to assure maximum availability and reliability as a power producer. Some of the interesting features incorporated in the control system are as follows:

(a) Only one control rod can be withdrawn at any one time. The maximum rate of reactivity insertion is not to exceed 0.04% $\frac{\Delta k}{k}$;

(b) A rod cannot be withdrawn while the coolant circulation rate is being increased;

(c) The spare pump cannot be started while a control rod is being withdrawn; and

(d) To reduce spurious scrams and improve plant availability, a "run-back" system is used under which, instead of scramming at once, the control rods are run into the core first to reduce power level. At 105% of set power level, an alarm is sounded to enable the operator to take corrective measures. At 110% of this power level the boiler control rods are "run-back" slightly. If this does not help then at 115% of power level the reactor scrams. It is recognized that certain situations require immediate scram and there is no time for "run-back". In such serious cases, as for instance high pressure build-up in the containment shell and high steam outlet temperature, turbine trip and low reactor pressure, a scram is initiated without delay.

Construction experience

214. Damage to the bellows. At the penetrations for steam lines in the containment building bellows have been provided to take care of expansion to the extent of $1\frac{1}{2}$ in. During the leak testing of the containment the pressure was increased to 97.5 psig. Due to an oversight the effect of this relatively high pressure on the bellows was not taken into account and no suitable stops were provided to restrict the expansion of the bellows to within allowable limits. The result was that the bellows expanded without control and stretched to 18 in. causing considerable deformation of the steam line elbow. Much time was lost in repairing the damage and work on the installation of reactor internals

had to be halted for six months. The bellows were cut out and replaced. All the six steam line elbows were stress relieved and closely inspected and dye penetration checks were performed. Fortunately, there was no damage to the steam lines as such. To forestall this type of incident, concrete 'I' beam structures have been put in to serve as stops against undue bellow expansion. Another stop will be installed inside the building. These will eliminate the problem in future pressure testing of the building.

215. In the re-circulation butterfly pumps the clearance between the disc and the pipe was found to be larger than intended. This caused some leakage. Repairs had to be carried out to overcome this difficulty.

216. It was difficult to obtain good welders for stainless steel welding because the area is lacking in such skilled workmen. A suitable training programme for welders had to be conducted which proved to be successful.

Operating staff

217. The latest plans indicate that the total operating staff will consist of 52 instead of 50 as reported earlier. There has been an addition of one test engineer and one maintenance man.

Fuel and fuel management

218. The fabrication of the boiler fuel has been proceeding well. The fabricator believes that the contract price is realistic. The second core loading of boiler-fuel will be made with two segments per rod instead of the present four because experience has shown that the alignment of four sections is difficult. Moreover, the two-section approach will reduce the peaking effect. The fuel rods will be assembled in a 9 x 9 array. However, the four corner rods of each 9 x 9 array will contain no fuel but boron poison mixed with aluminium oxide.

219. The next superheater core will have UO_2 of relatively low enrichment (6% to 7%) and will contain 6% to 10% more uranium. The cladding material will probably be Incaloy 800. The superheater fuel rods will be grouped in 7-rod clusters, with spacer wires made of a burnable poison, gadolinium, which has a very high cross-section. Each rod will be of free-standing type and the design of the cluster will ensure that there are no vibrations.

220. Delivery of the superheater fuel for the initial loading was expected by the end of July 1963.

221. Fuel cycle costs. The exact fuel cycle costs cannot be determined at this stage because of the many unknowns, especially with respect to the superheater fuel. The first complete core is expected to cost US $2.25 million and according to the design estimates the fuel cycle cost for the initial core will be about 6.25 mills/kWh. The second core will be of lower enrichment and the fuel cycle cost is expected to be 5.15 mills/kWh. The third core will also be of low enrichment but have a higher burn-up rising to 18 000 MWd/t in the superheater and 18 000 MWd/t in the boiler. This will lead to fuel cycle costs of about 4.35 mills/kWh. In the fourth core, when the equilibrium conditions have been reached and the optimization has been carried out to obtain 15 000 MWd/t burn-up, the fuel costs are expected to be 2.9 mills/kWh.

222. For the sake of comparison, it may be mentioned that for a 50 MWe conventional plant a few miles from the Pathfinder site, the present fossil fuel costs are 38.8¢/million BTU for coal and 27.7¢/million BTU for interruptible supply of natural gas. Under the equilibrium conditions, the nuclear fuel costs/kWh will be considerably lower than the average fuel cost for the comparable conventional plant.

223. Fuel management. Detailed schemes have been devised for managing various core loadings until the equilibrium conditions have been reached which involve shuffling the fuel elements from outside to inside or from the area of low power density to that of higher power density for maximizing the fuel burn-up and reactivity control.

224. The boiler core has been divided into three distinct groups with 32 elements adjacent to eight outer control rods forming group I; 32 elements adjacent to eight inner control rods as group II; and the remaining 32 elements as group III.

225. The proposed fuel cycle for the equilibrium core requires reloading at a third of the boiler fuel. For an average fuel burn-up in the boiler core of 10 000 MWd/t, this means refuelling every 5½ months at 0.8 load factor.

226. For the first high enriched superheater all the fuel elements are to be replaced at the end of about nine months' operation at 0.8 load factor. This is predicted to yield an average fuel burn-up of about 17% of the initial fuel loading. The flat flux over the superheater region will produce uniform burn-up of all these elements.

227. Using the three group reshuffling schemes the average burn-up of the first one-and-a-third boiler core (128 elements) will be about 7800 MWd/t. The first third batch to be discharged will have an average burn-up of 4000 MWd/t, the second 7300 MWd/t and the third 9400 MWd/t.

228. After discharge the fuel will be stored at the site for 90 days to permit cooling off. A contract has been signed with a private concern called the Nuclear Fuel Services for reprocessing of fuel. The irradiated elements will be shipped by rail in 76 ton casks developed by another private company to serve the Dresden and other nuclear plants as well.

Selected references

229. A list of selected references concerning the Pathfinder power reactor is given below:

Technical Specification for the Pathfinder Atomic Power Plant, Northern State Power Co. (May 1963).

Pathfinder Atomic Power Plant, Program and Organization for Preoperational and Nuclear Testing, ACNP - 6112/Rev.1, Allis-Chalmers Manufacturing Company (May 1963).

Proceedings of the Nuclear Superheat Meeting No. 7, Chicago Operations Office, USAEC (October 1962).

Proceedings of the Nuclear Superheat Meeting No. 8, Chicago Operations Office, USAEC (April 1963).

VI. THE PIQUA NUCLEAR POWER FACILITY

General

230. The Piqua reactor achieved initial criticality on 10 June 1963, requiring 21 fuel elements at precisely the rod position predicted. A full core loading will contain 85 fuel elements, although with entirely fresh fuel this would represent too much excess reactivity. It is therefore planned to load about 60 elements, then operate at power for six or eight months until the reactivity has decreased to the point that the remaining elements may be loaded.

231. The initial nuclear testing programme at very low power will require several months, after which another three to four weeks will be needed to reach full power. After reaching full power, another set of operational performance tests will be run over a period of about seven weeks, of which two weeks will be continuous operation at full load and another two weeks will be load following operation. The plant should then be ready to be turned over to the operator, the city of Piqua, by December 1963. Thus the schedule of activities, as currently planned, may be approximately as follows:

Table 18

The Piqua nuclear power facility: Time schedule

Item	Expected
Initial criticality achieved	10 June 1963
Loading of additional fuel, low power testing, such as determination of temperature coefficient, shutdown margin, calibration of control rods, and checking hydraulic performance	Mid-June to end of September
Approach to full power	October
Full power achieved	End of October
Testing at power, adjustment of temperature and pressure controllers, check performance of boiler and superheater, determine power coefficients at full coolant flow and reduced flow	Completion, end of November
Sustained full power operation, load following power operation, and measurement of xenon transients	December 1963
Turnover to City of Piqua Municipal Power Commission	1 January 1964

Experience in design, construction and pre-operational testing

232. Control rod drives. The final testing of the magnetic jack-type control rod drive system at Piqua was completed after the rod drive system had been installed in the plant. A number of circuit problems were encountered, for example, short circuits in power leads due to faulty insulation, the discovery that certain circuit breakers and rectifiers needed to be replaced with equipment of a higher rating, and the failure of rectifiers in the driver coil circuits due to high reverse voltages. The testing and modification required in the final application of the system was greater than had originally been anticipated.

233. Heating and ventilating system. The additions to the cooling capacity of the building ventilation system, as reported last year[5/], have proved effective. After doubling the cooling capacity by the installation of four water-cooled air re-circulation units, peak ambient temperatures in the reactor building were maintained below 110°F during the summer months.

234. Leakage of organic coolant at pump seals. In a number of instances shaft seals of pumps and packing glands on valves in organic coolant lines have leaked or failed. The organic fluid confined by these seals must be kept hot enough to prevent freezing of the organic but cool enough to avoid rapid deterioration of the sealing material. The two pressurizing pumps in the auxiliary organic circuit have needed close attention. The Teflon seals in these pumps became overheated and failed several times over a period of six months. Measures taken to improve the performance were to provide additional cooling water to the seal, to install a serrated-throat bushing of the proper size designed to keep the hot coolant away from the seal, to plug eight holes in the hold-down plate at the seal bushing, and to replace Teflon with a more heat-resistant material. The main coolant circulating pumps and a number of small gear pumps have also given trouble. Some seal leakage has been encountered with the large, main coolant pumps, and seals have been replaced. In addition, bearings in the main coolant pumps have shown some wear and have required replacement. The small pumps used in the purification system, new coolant storage area and drain tank room showed abnormal wear of impeller bearings, developed leaks and failed after a short period of operation. The metal bearings on these pumps have been replaced with bearings of another material, and the pump speed has been reduced. The bearings now have an acceptable life expectancy.

5/ Ibid., para. 124.

235. Leakage at valves and flanges. Leakage of hot organic fluid at the valve stem of a large butterfly valve in a 20-inch diameter line has been encountered. Modifications were required to adequately cool the packing gland in order to reduce this leakage. In other cases, valves did not function properly because the organic fluid became too cool and froze in the valve stem packing gland. The problem therefore is one of temperature control. Flanged connections have also leaked, and have required the attention of the maintenance force. Of particular interest is the closure of the reactor pressure vessel. The reactor vessel head is a flat steel disc, 9 inches thick and more than 9 feet in diameter, which is bolted to the top flange, sealed by a soft metal gasket. Proper sealing of this head has been difficult. As the temperature of the system increased, leaks appeared, in spite of additional tightening of head bolts during temperature increase. Care was required to prevent galling of the threads of the head bolts. When it was desired to take off the vessel head, the head bolts were difficult to remove. These bolts are to be replaced with rugged stud bolts, and stud tensioners have been ordered which will apply stress to the studs accurately and uniformly prior to the system temperature increase. This change is expected to correct this leakage problem.

236. Improvement in the steam tracing system. As noted in last year's report, the large number of tubes, pipes and steam traps in the steam-tracing system led to overcrowding of parts of the plant, especially so in the organic purification room of the auxiliary building. The steam tracing system was extensively modified by replacing large numbers of bucket-type steam traps with a compact impulse trap and by combining many of the small, individual steam lines into common lines which reduced the number of traps required. The net result of these modifications was an impressive improvement in the appearance of the purification room and other parts of the plant, with adequate work space now being available and piping and equipment being readily accessible for maintenance.

237. Fire in reactor building. A small fire broke out in the reactor building on 5 September 1962 due to leakage of hot organic coolant from the main heat transfer piping. the leak occurred when a thermocouple well fractured within the main coolant piping. The fire actuated the automatic sprinkler system which extinguished the fire within about three minutes. It is estimated that about 3 to 5 gallons of the organic material burned, and that another 8 gallons flowed

out of the pipe and solidified on nearby surfaces before the flow completely stopped. Damage due directly to fire was limited, but smoke damage was extensive in the main pump room and in the main heat transfer room. Clean-up operations required more than a week.

238. The thermocouple well which failed consisted of a tube 7/16 inches in diameter, extending about ten inches into a 14-inch pipe within which organic coolant at 585°F and about 110 psia was being pumped at the rate of 6000 to 7000 gallons per minute. It is thought that this tube was subjected to severe vibration which resulted in fatigue failure of the metal. The thermowell and others of similar design were replaced with wells which were shorter and of increased wall thickness to prevent recurrence of this type of failure.

Fuel management

239. 120 fuel elements have been manufactured for Piqua. Since only 85 fuel channels are available in the reactor, the existing fuel supply should last about two and a half years at the design irradiation level of 3300 megawatt-days per metric ton of uranium, or even longer if fuel performance is better than this. It is possible that an irradiation level of 5500 MWd/t uranium may be achieved. The problem of the fouling of fuel elements by deposition of solids on the fuel cladding surfaces should be resolved during operation with the first fuel loading. Plans for fabrication of a second batch of fuel elements are not yet definite; design objectives for new fuel will be higher operating temperature and longer life (perhaps 10 000 MWd/t). The next fuel could be uranium dioxide, with finned aluminium cladding, or another promising fuel for this organic reactor is uranium carbide. Experience with uranium carbide fuel in the Hallam sodium-graphite reactor may be of benefit to the Piqua station.

Operating characteristics

240. Steam produced in the Piqua nuclear station is delivered to the City of Piqua Municipal Power Station, located near the reactor plant. The steam from the nuclear station is fed to a header which supplies three turbine-generators, each 4 MWe, so the operation of the reactor is not tied to the demand of any single turbine. It is planned to operate the reactor as a load-following unit, varying the power level as required by the demand for steam. Under these conditions, the reactor can operate anywhere between 0 and 100% of full power. The reactor will follow load automatically in the range 20% to 100% of full

output. Within this range, the maximum rate of change of load is 4% of full load per minute. The reactor can be controlled manually at any time, and over the full power range. About 5 minutes will be required to go from hot standby operation to 20% of full load.

241. Superheated steam delivered by the reactor to the turbine supply header will be at the same condition of temperature and pressure (550°F and 441 psia) at all power levels; i.e. only the steam flow rate will be varied. The 12 000-gallon-per-minute rate of flow of organic coolant through the reactor will be kept constant during two-pump operation, regardless of the operating power level, and the temperature of the organic coolant leaving the reactor will also be constant at all power levels. Variations in reactor power will, therefore, be dependent upon changes in the temperature of the organic entering the reactor, which will be brought about in the following way. The constant flow of hot organic leaving the reactor will be split into two streams, one through the steam generator and the other by-passing the steam generator, with the two streams being reunited before returning to the reactors. Variations in power will be accomplished by varying the fraction of the organic coolant flow which is passed through the steam generator, the rest being by-passed around the steam generator. Thus if more steam is required, control valves are adjusted so that a greater proportion of the organic flow will be passed through the steam generator, and the organic returned to the reactor will be cooler, the temperature rise in the coolant passing through the reactor increases and the reactor power goes up. Therefore the primary control parameter is the quantity of steam demanded by the turbines, and this in turn determines the quantity of steam generated, the flow rate of organic through the steam generator, the temperature of the organic fluid entering the reactor, and the reactor power level.

Operating staff

242. Personnel of Atomics International and the City of Piqua will work together during the initial operation of the nuclear plant. After the final testing has been completed, the City of Piqua will assume responsibility for operation, with a few Atomics International people remaining at the site for a short period for technical support. When the turnover has been completed, the City of Piqua now plans to employ 42 people to staff the nuclear power station. This does not include the staff in the steam turbine and generator

facility which is a part of the conventional portion of the station and located in a separate building. The revised staffing plan given in table 19 represents an increase of 17 as compared with the plans of two years ago. It has been found that the maintenance force had to be increased to 17 instead of the earlier estimate of 7. The other increases are additions to the technical strength of the organization. The City of Piqua will employ an operations engineer, who will attend to the day-to-day operating problems and who will analyse operating data. A process engineer, a process chemist, a reactor physicist and a laboratory technician will also be employed to furnish technical support to the operating personnel. Thus the regular staff should be able to cope with most of the operating problems likely to be encountered. In unusual situations, outside consulting services will be employed as needed. In addition to this, Atomics International, under a separate contract with USAEC, will assign a small operations analysis group to the Piqua plant. This group will observe the operation and performance of the organic reactor system, analyse the data and make the information available to the civilian reactor programme.

Table 19

Piqua nuclear power facility: Staffing plan

Category	Number of persons
Administration and General	
Reactor superintendent	1
Secretary	1
Accounting analyst	1
Health physicist	1
Health physics technician	1
Process engineer, process chemist, technician	3
Reactor physicist	1
Operation and maintenance	
Supervisor	1
Operations engineer	1
Operations	
Shift supervisors	4
Reactor operators	10
Maintenance	
Instrument maintenance	5
Mechanical and electrical maintenance	12
TOTAL	42

Future outlook

243. Looking beyond the Piqua reactor and considering the concept of the organic moderated reactor system, there are several observations which may be made.

244. In the United States reactor development programme, the emphasis on organic reactors has been reduced. This is explained in part in a USAEC report of November 1962 entitled "Civilian Nuclear Power - A Report to the President - 1962", in which the following statement concerning organic moderated and organic cooled reactors is made: "Although showing early promise, this development has been plagued by a tendency of the fluids to "foul"; that is, to form gummy substances that coat the metal surfaces and interfere with heat transfer. This fouling increases markedly with temperature. Although this problem will undoubtedly be solved, at least for moderate temperatures, it is not clear that this reactor has better potentialities than the light water ones for power generation, though it may for process heat because the liquids used do not become radioactive." (page 36). On page 50 of the same report is is stated: "The organic cooled and moderated reactor can be economically competitive with saturated steam water reactors and may have application for process heat generation."

245. Construction of the Experimental Organic Cooled Reactor (EOCR) has been completed at the National Reactor Testing Station in Idaho, but it has been decided for economic reasons not to operate the reactor. The plant will remain in standby status pending a decision on its future use.

246. However, Atomic Energy of Canada, Ltd. plans to explore the heavy water moderated, organic cooled reactor concept with the construction of a test reactor, Whiteshell Reactor 1. Organic cooled, heavy water moderated reactors are also to be built by Spain (DON project), and by EURATOM (ESSOR reactor). An organic moderated and cooled reactor is to be constructed by the CNRN in Italy (PRO reactor). Interest in the organic concept has also been evidenced by design and engineering studies in Denmark, Germany and Great Britain.

247. The extent of future effort on organic reactors will be influenced greatly by the success of Piqua and the other projects just mentioned. If Piqua performs exceptionally well, then considerable interest may be attracted to this concept from the nuclear industry and the electric utilities.

VII. THE HALLAM NUCLEAR POWER FACILITY

General

248. The Hallam nuclear power plant had begun power operation on 29 May 1963, increasing in power level to 50% of full power before having to shut down on 9 June to modify the three pumps in the secondary sodium system, one of which had failed.

249. The plant reached full design rating of 75 MWe on 16 July 1963. After running steadily at full power for three and a half days to check heat balances, the reactor was shut down to observe xenon build-up. A final shielding survey was also conducted. On 22 July, the plant was again taken to power to begin a 30-day test, after which the plant will be operated routinely as a part of the Consumers Public Power District generating system. It is expected that Atomics International will turn over full responsibility to CPPD in about October 1963.

System and component modifications

250. The higher capacity pump required by the closed-loop cooling water system for the fuel storage cell, reactor-cavity liner, and radioactive gaseous waste compressor, mentioned in last year's report, have already been installed.

251. Vibration problems in the nitrogen piping system of the load face shield cooling were due to pneumatic hammering. In order to solve this, snubber accumulator tanks were installed for smoothing peaks and pulsation. Installation of the snubber tanks resulted in an increase of the apparent capacity of the compressors.

252. Excessive leakage was found in the nitrogen system compressors, actually air compressors, in which leakage can be tolerated. The estimated loss of nitrogen amounted to US $2000 per month. To stop this a leak-off system between double seals was installed to compress the leakage and return it back to the line. The cost of this modification was approximately US $5000.

253. Intermediate heat exchangers. On 18 November 1962, it was found that a leak had developed in one of the intermediate heat exchanger units of the No. 1 sodium heat transfer loop. Subsequent testing of both units in this loop indicated that unit 1-A was leaking. The unit was cut off from the heat

transfer loop and removed to the high bay area near the maintenance cell. Piping was arranged in such a way that reactor and plant test programmes could proceed.

254. The cleaning of the defective unit was performed with the help of two short flushes (lasting approximately half an hour each) with ammonia, followed by a nitrogen purge to remove ammonia fumes and vacuum cleaning to remove the loose and accessible sodium oxides.

255. The exact location of the leaking tube was found, the tube was cut loose by a reaming operation and extracted in pieces by pulling it through the bottom tube sheet. All pieces were sent for metallurgical and mechanical examination to the Argonne National Laboratory.

256. The results of tests and studies indicate that the failure was due to fatigue induced by vibration in the baffle plate area. The natural frequency of the failed tube and all similarly supported tubes was calculated to be approximately 42 cycles per second. This is in the range of the frequency of vibrations induced by a flow rate slightly lower than that corresponding to full power. Measurements performed on the other intermediate heat exchangers indicated that the amplitude of vibration was a function of secondary sodium flow only. Measured frequencies had the same order of magnitude as those which had been calculated, namely 37 to 35 cycles per second.

257. Eight proposals for corrective action were considered. Some of them involved modifications in the fluid flow, some involved changes in the tubes subject to strains and some involved strengthening of the support structure of the tubes. The one that was adopted consisted of the installation of dampening shims between the tube rows in the high velocity sodium flow area.

258. All procedures for repair were demonstrated on the unit which was already displaced from its normal position and was, therefore, easier to work on. To ensure that the repair would be effective, a vibration test was carried out on a tube in the affected area prior to and after the repair.

259. The vibration tests conducted on all the intermediate heat exchangers after the modifications showed that the original tube vibration and resonant frequency range had been reduced to such a small value that no appreciable damage should result from this operation at sodium flows up to the design conditions.

Post-critical testing

260. Most of the post-critical tests are "on line" tests conducted at various power levels up to full power. Testing at each power level confirmed plant operability at that level. Tests conducted up to a power of 38 MWth have demonstrated that the plant is essentially performing as designed. Some of the results are given in paragraphs 261 to 266.

261. Power coefficients computed from the results of power ramp tests at low power levels are given in the table below, together with the corresponding values predicted.[6]

Table 20

The Hallam nuclear power facility: Power coefficients

Source	Prompt		Slow		Steady state
	Power coefficient	Period	Power coefficient	Period	Power coefficient
NAA-SR-5700	-0.5¢/MWth[a]	2.5 sec	+0.5¢/MWth	~200 sec	+0.08¢/MWth
2.5 MWth	-0.4¢/MWth	2.5 sec	+0.4¢/MWth	~200 sec	zero
10 MWth	-0.3¢/MWth	2.5 sec	+0.8¢/MWth	~200 sec	+0.5¢/MWth
20 MWth	-0.40¢/MWth	2.5 sec	+0.45¢/MWth	~200 sec	+0.05¢/MWth
38 MWth	-0.35¢/MWth	2.5 sec	+0.40¢/MWth	~200 sec	+0.05¢/MWth

a/ For the HNPF reactor, reactivity in per cent is equal to 0.00693 times reactivity in cents.

262. Xenon poisoning characteristics were evaluated by determination of changes in core reactivity with time, due to xenon and temperature effects, using the differential period techniques.

263. Table 21 summarizes the conclusions of the xenon curve, extrapolated with the help of an analog computer from a power level of 38 MWth (15%) at which the following tests were performed:

6/ Final Hazards Summary Report for the Hallam Nuclear Power Facility, NAA-SR-5700, with Supplements, Atomics International, Canoga Park, Calif. (1962).

(a) Building and subsequent decay of xenon after shutdown from 38 MWth equilibrium;

(b) Initial build-up of xenon as a function of time and power;

(c) Dependence of the above factors on power level.

264. The xenon data predicted by the "KINDLE" Computing Code and the Final Hazards Summary Safeguards Report for the HNPF (NAA-SR-5700) are also included in the table.

Table 21

The Hallam nuclear power facility: Xenon characteristics

Power level	Source of data	Equilibrium xenon		Peak xenon after shutdown	
		Time	Magnitude	Time	Magnitude
15%	Measured xenon	Approx. 3 days	- 59¢ a/	1½ hours	-(69¢) b/
	Theoretical fit to measured xenon		- 61¢		- 62¢
	KINDLE Code		- 62¢		- 63¢
50%	Extrapolation of theoretical fit	Approx. 3 days	-158¢	Approx. 3 hours	-163¢
	KINDLE Code		-159¢		-164¢
100%	Extrapolation of theoretical fit	Approx. 3 days	-238¢	Approx. 5 hours	-263¢
	KINDLE Code		-241¢		-266¢
	NAA-SR-5700		-(204¢) c/		-(228¢)

a/ For the HNPF reactor, reactivity in per cent is approximately equal to 0.00693 times reactivity in cents.

b/ Contains improper temperature correction.

c/ Equilibrium xenon plus samarium is reported as 2.3%. A breakdown results in -204¢ for equilibrium xenon.

265. The results obtained in this test agree closely with those predicted. Furthermore, there are indications that the data obtained at 15% power can be extrapolated with reasonable accuracy to higher power levels. This, however, will be checked against actual measurements to be made during the next few months.

266. Plant radiation surveys have revealed no major shielding problems so far. Several areas in the plant, however, have shown higher radiation levels than anticipated, and have been, therefore, under particular surveillance during radiation shielding testing.

Approach to full power

267. It will take 27 days to go from zero power to full power in seven stages. During the fourth stage a scram test is planned at a 62.5% nominal power; a loss of load test is scheduled to take place during the last step, at a power level of 87.5%.

268. A "Master Schedule" consisting of 12 tests is used for checking the characteristics of the plant at different power levels; these included testing of hydraulic and heat transfer properties, of plant protective systems, of control systems and also determination of power coefficients, power distribution, etc.

269. After reaching full power, a period of thirty days will be allowed to demonstrate operation at full load, as well as the ability of the station to respond to changes in load.

Fuel for second core

270. It is expected that uranium monocarbide (UC) fuel will be used for the second core at HNPF because of its apparent economic advantages over other fuels. Ten full-scale UC fuel elements were fabricated to obtain experience in the production of uranium monocarbide and in the fabrication of fuel slugs. These ten elements are to be placed in the core in order to obtain operating data on UC fuel under HNPF core conditions, and to acquire information on the reactivity worth of UC fuel as compared to U-Mo fuel; also, as operation proceeds, to gather fuel temperature and burn-up data.

271. The initial UC fuel elements were designed to be compatible with the U-Mo fuelled core. The temperature rise of the coolant through the elements will be 335°F during full power operation. The UC fuel will be positioned in the core to ensure a high burn-up rate.

272. The geometry of the first core UC elements is approximately the same as the current reference design for all future UC elements to be used in the HNPF core. Each element contains eight fuel rods. The fuel length is 156 in. and

the fuel slug diameter is 0.872 in., resulting in a fuel cross-sectional area of 4.78 in.^{2} per element. Fuel slugs are contained within an 0.010 in. Type 304 stainless-steel tube with an 0.030 in. annular sodium bond between the fuel and the cladding. A 21 in. gas space above the fuel allows for sodium bond expansion, fuel radiation growth, and gas pressure build-up. Fuel rods are arranged in one ring within a process tube, and in the element centre is a graphite-filled tube which prevents channelling of sodium coolant through this region and also decreases axial neutron streaming through the element. Fuel rod spacers are located at 18 in. intervals along the length of the rods.

273. The maximum design temperature for this UC fuel has been established at 1600°F. The maximum fuel surface temperature for a 1600°F centreline temperature is 970°F. The burn-up objective of these elements is a peak of 20 000 MWd/t.

274. Uranium enrichment for eight of the initial core UC elements is 3.7 wt% U^{235}, and for the other two is 4.8 wt% U^{235}. The latter two elements must be in other than the highest power region to avoid overheating.

275. Core positions were chosen so that the ten UC elements would produce between 1.9 and 2.09 MWth at full reactor power. Fuel element power in any particular core position is approximately proportional to the U^{235} content; thus, a 3.7 wt% enriched UC element with 5.89 kg of U^{235} will produce 83% as much power as a 3.6 wt% enriched U-Mo element with 7.12 kg of U^{235}, and a 4.9 wt% enriched UC element with 7.81 kg of U^{235} will produce 110% as much power as a U-Mo fuel element. The eight 3.7 wt% enriched UC fuel elements are placed in core positions where U-Mo would produce 2.3 to 2.5 MWth at full power. The two 4.9 wt% enriched UC fuel elements are placed in core positions where U-Mo would produce 1.7 to 1.9 MWth at full power. Each of the 3.7 wt% enriched UC elements is estimated to have a slightly greater reactivity worth than a U-Mo element, and each 4.9 wt% enriched UC element will be worth about 10% more than a U-Mo element.

Waste disposal

276. It is estimated that 200 to 400 gallons of radioactive water will be accumulated every year from the cleaning of 40 fuel elements. Each element requires between 5 and 10 gallons of water in the form of steam to take off

the sodium so that the reprocessing plant can treat the U-Mo fuel elements. Uranium carbide fuel elements will be handled by the reprocessing plant without washing in the maintenance cell.

277. Gaseous wastes are estimated to total some three curies per year.

Operating personnel

278. The staffing plan of the station has been changed since last year's report by the addition of ten men, making a total of 74. The post of assistant plant superintendent has been changed to plant operations supervisor and the following posts have been added:

1 plant accountant
1 office engineer
2 more auxiliary operators
1 engineering aide
1 laboratory technician
1 mechanical foreman
2 mechanics
1 utility man.

279. During operation, the shift supervisor has over-all responsibility for operation of the whole Sheldon Station which includes HNPF. A shift operating crew normally consists of the following eight men:

1 shift supervisor
2 unit operators
2 equipment operators
2 auxiliary equipment operators
1 instrument technician or electrician.

Training

280. About 30 Sheldon Station employees of the CPPD attended a formalized training programme on the HNPF during 1960 while the facility was under construction. The programme conducted and developed by Atomics International provided approximately 900 hours of training, and was completed in six months. Criteria for the development and presentation of the first programme can be summarized as follows:

281. HNPF operations personnel should have:

(a) An understanding of the potential hazards of a nuclear power plant, of the considerations given to safety during the design of the HNPF and of the inherent safety features of a sodium graphite plant;

(b) An understanding of the physical and operational characteristics of the HNPF components and systems; and

(c) A basic knowledge of nuclear physics, nuclear instrumentation, and reactor technology.

282. These criteria are in accordance with the prescribed USAEC regulations[7]. It was required that each CPPD trainee:

(a) Be experienced in conventional power plant operation;

(b) Be familiar with many of the features of the HNPF, as a result of his having worked at the facility for a year or more; and

(c) Spend an appreciable portion of his own time studying course material and becoming familiar with equipment and facilities.

283. The programme covered: basic mathematics and physics; atomic and nuclear physics; basic instrumentation; HNPF components and fluid systems; HNPF instrumentation and control systems; training in a research reactor; training in the Sodium Reactor Experiment; safeguards and casualty studies; field trips; health physics; component handling facilities; HNPF electrical systems; review and detailed studies of HNPF instrumentation; and core loading.

284. In addition to the above regular training, the following CPPD personnel received extended training at Canoga Park (AI) for the length of time shown:

1 plant superintendent	1½ years
2 performance engineers	6 weeks
1 chemical engineer	6 weeks
1 health physicist	6 weeks
1 maintenance supervisor	6 weeks
1 electronics specialist	3 weeks
3 shift supervisors	3 weeks.

285. A second HNPF training programme for CPPD personnel was initiated by AI in September 1961; its primary purpose was to train reactor operators preparatory to their taking AEC reactor operator licence examinations. This programme was of a more detailed and specific nature than the initial one. Personnel who participated in this training programme had extremely varied experience and educational backgrounds. Also, the programme was designed to proceed simultaneously with reactor testing. Therefore, trainees were divided into three groups and the programme was carried out in three sessions, each

7/ The USAEC Operator Licensing Regulations, Title 10, part 20, Code of Federal Regulations.

session covering essentially the same subject material. Thus two thirds of the trainees spent full time on plant testing while the remainder were engaged in formal training. Certain AI employees were also required to obtain HNPF reactor operator licences and, consequently, both AI and CPPD personnel have attended each training session. The first session was begun in September 1961 and the third session was completed in February 1963.

Future outlook

286. Operation of the sodium-graphite reactor and auxiliaries in the Hallam plant is expected to furnish valuable information concerning the use of sodium as a reactor coolant and heat transfer medium in a steam-raising system, and also concerning the performance of fuel elements in such a system. The fact that the reactor has already produced steam at 825°F and 800 psi indicates that the sodium-cooled systems can generate superheated steam. If future performance of the Hallam plant is satisfactory, and the improved fuel elements, such as uranium carbide, live up to design expectations, then additional, larger units of this type might be built. The SGR system seems particularly well adapted to being scaled up to large sizes, with appreciable reduction in unit cost. A unit size of 1000 MWe or larger appears to be quite feasible.

Selected references

287. A list of selected references concerning the Hallam nuclear power facility is given below:

Final Hazards Summary Report for the Hallam Nuclear Power Facility NAA-AR-5700, with Supplements, Atomics International, Canoga Park, California (1962).

Consumers Public Power District, Provisional Operating Authorization No. DPRA-1, Amendment No. 1 dated 9 August 1962. US Atomic Energy Commission, Washington, D.C. (9 August 1962).

LOOMIS, J.S., Training of Consumers Public Power District Operating Personnel for the Hallam Nuclear Power Facility, NAA-SR-7997 (25 February 1963).

PURSEL, C.A., Post-construction Testing of the Elk River, Hallam and Piqua Power Reactor Plants, USAEC, Argonne, Illinois. Paper presented at the IAEA Conference on Operating Experience with Power Reactors, Vienna, 4-8 June 1963.

COCHRAN, J.D., Operating Experience at Hallam Nuclear Power Facility. Paper presented at 25th Annual Meeting, American Power Conference, Chicago, Illinois, 26-28 March 1963.

VIII. THE PEACH BOTTOM HIGH TEMPERATURE GAS-COOLED REACTOR

General

288. The Bechtel Corporation, the prime contractor for the Peach Bottom plant, has established the rather short period of 27 months for the construction of the entire project. Construction was started in March 1962 and is scheduled for completion in June 1964. As of 1 April 1963 construction had progressed to about 35 per cent of completion, which agrees closely with the original construction schedule. However, in order to maintain this progress, the order in which certain jobs were done has had to be rearranged. This was necessitated largely by delays in the delivery date of certain items of equipment, including the reactor pressure vessel and the steam generator, which were originally scheduled for delivery in June and July 1963 but which will be delayed. To offset this partially, the installation of the turbine-generator will be completed sooner than originally planned. Another change in scheduling intended to accommodate late delivery of the reactor pressure vessel is related to the containment shell. By September 1963, erection of the containment shell will be carried to the level of the reactor operating floor, which is about 60 feet above grade level. The closing of the top of the containment shell will be postponed until the reactor vessel and steam generator are delivered and have been lifted to the top opening and lowered into place. Earlier it had been planned to set the reactor vessel in place when the containment shell was at grade level.

289. Aside from the reactor vessel and steam generator, all other major items of equipment have been ordered and many of these, such as boiler-feedwater pumps, service water pump, condenser, plant heaters and exchangers have already been delivered to the site. Delivery of the helium compressors and circulators is expected in September 1963. It is still possible that the original target date for completion of construction, June 1964, will be met. If so, the plant should achieve initial criticality by October 1964 and operate at design power by December 1964.

Construction experience

290. The construction at the Peach Bottom site has proceeded smoothly, using craftsmen and construction workers available locally. The prime contractor is

responsible for over-all design, engineering, procurement and construction of the plant. However, the nuclear steam generating portion of the plant is being furnished by General Atomic, so that design and procurement of the reactor vessel and internals, nuclear fuel, control system, steam generator and other reactor plant equipment are the responsibilities of General Atomic.

Status of major components

291. Reactor vessel and steam generators. The reactor vessel and steam generators are being fabricated by Baldwin-Lima-Hamilton. The material of construction is carbon steel (ASTM A121 Grade B), hence the problems of stainless steel cladding are avoided. However, there has been some difficulty in meeting the welding specifications established by General Atomic, which are more exacting than the ASME code. For the large Peach Bottom components, it is required that slag inclusions in weldments shall be no longer than 1/8 inch. The present ASME code specifies that slag inclusion must be no longer than 1/2 inch. It is possible to meet the 1/8 inch specifications but this has been found to require both additional time and additional money. As a result the delivery of the reactor vessel and steam generators may be delayed about six months. As of April 1963, the holes in the top and bottom heads of the pressure vessel were being fitted with nozzles and being welded according to carefully prepared procedures. Parts of the steam generators had been fabricated and welded. Fabrication of the Peach Bottom pressure vessel has benefited from the experience in the fabrication of the EGCR pressure vessel, which was done in the same shop. However, there is a difference in design, since the Peach Bottom steam generators use forced circulation of water from the steam drum to the boiler through a re-circulating pump while EGCR depends on natural circulation. The Peach Bottom steam generators represent an extrapolation of design from smaller units to larger ones, and the units will not be put through a performance test until installed at the Peach Bottom site.

292. Main helium circulators. The two centrifugal compressors had been manufactured and were to be tested in August of this year.

293. Fuel transfer machine. It has been completely assembled and placed in a test stand. All mechanisms have performed well as the machine removed and replaced fuel elements in a mock-up of a portion of the reactor core.

294. Control rod drive mechanism. Manufacturing of the control rod hydraulic drive units and the hydraulic power supply equipment is progressing well. The first drive unit to be completed has been tested at General Atomic. During this performance testing, several deliberate attempts were made to induce malfunctioning, such as dropping graphite chips into the drive mechanism, deliberately misaligning the guide tube, or producing cavitation in the hydraulic motor. In each case the unit performed satisfactorily.

295. Emergency shutdown rods. Fabrication has been started on the emergency shutdown rods and their electrical drive units. Nineteen of these rods will be provided for emergency shutdown, deriving their power from storage batteries.

296. Helium purification compressors. The compressors in the helium purification system have been modified as a result of a test when interaction between two stages produced pressure surges.

297. Fuel fabrication. Peach Bottom fuel is being manufactured by General Atomic at their fuel fabrication facility. Important parameters which must be controlled in the production process are pre-sizing of the uranium-thorium carbide particles, proper temperature for deposition of pyrolytic carbon in order to control deposition rate, maintaining the proper proportions of carbon, uranium and thorium in the powdered mixture, achieving homogeneity of graphite and fuelled particles when forming the compact, and maintaining dimensional tolerances during the sintering operation. Thorium carbide must be kept out of contact with moisture in order to prevent oxidation, but after being pyrolytically coated with carbon, the granules can be immersed in water with no adverse effects.

Helium leak-tight considerations

298. Systems which are to contain helium must be carefully designed in order to minimize leakage at such points as valves, pump seals, joints and fittings. However, the problem of containing helium is not insolvable, as shown by work in this area. For example, the United States Bureau of Mines has accumulated a large amount of experience and technical knowledge in the handling of helium. Experience in nuclear projects has shown that welded joints can be made which are helium leak-tight. In the Peach Bottom project, all components intended for service with helium are checked and leak-tested at the vendor's shop. Furthermore, installation of equipment has been designed so as to require a

minimum number of field welds and for placement of welds where they are accessible for testing. The additional price which was paid to meet helium leak-tight requirements was not as great as had been anticipated, amounting to 1 to 2% for large components and 20 to 30% for small components.

299. The reactor head flanged closure (14 feet in diameter) will be seal-welded in order to insure against leakage. However, the five flanged joints at the nozzles on the top reactor head and the majority of the smaller flanged joints will not be seal-welded, since this is thought to be unnecessary.

Irradiation test of HTGR fuel

300. A test fuel element which is identical in diameter with the Peach Bottom fuel but only one quarter the active length has been operated in the General Atomic in-reactor loop at the General Electric Test Reactor since 17 March 1962. It is planned to continue irradiation of the fuel until perhaps December 1963, by which time it will have attained an average burn-up of 40 000 to 50 000 MWd/t and a peak of 100 000 to 125 000 MWd/t. The test conditions with respect to maximum helium temperature, helium pressure and maximum internal fuel temperature are about the same as design conditions for Peach Bottom reactor, but the over-all average heat flux or average thermal power per unit length of fuel in the test loop is about twice as great as the design figure.

301. By 31 July 1963, the fuel element had produced a total of 453 MWh of fission heat, which is approximately 28 000 MWd per metric ton of uranium-thorium or about 60% of the heat generated by an equal length of an average Peach Bottom element in three years of operation at 80% capacity factor.

Testing of the fission product trapping system

302. The loop in the General Electric Test Reactor described in paragraph 300 above is also fitted with a heat exchanger, a main helium circulating system, and a helium purge and fission product trapping system. Thus the loop contains all the major elements of the Peach Bottom gas handling system and constitutes a realistic test of the fission product release rate and fission product control capability. A previous test fuel element was irradiated in this loop from 12 October 1961 until it was removed in February 1962. The present test has been in progress for 15 months and is continuing. Operation of this in-reactor test loop has given the following significant results:

(a) The method of fission product control has been shown to work well;

(b) It has been demonstrated that the release of fission product activity to the purge gas and to the main coolant gas is considerably lower than the Peach Bottom design figure, and hence that metal cladding of fuel elements is not necessary;

(c) Post-irradiation examination of the first fuel element showed excellent dimensional stability and no change in physical appearance of the element;

(d) The purity of helium in the coolant loop was easily maintained within specifications;

(e) There were no difficulties with mass transfer of carbon from the fuel element to other parts of the loop;

(f) Accumulation of activity on components in the primary loop was so low that direct maintenance was easily accomplished. In fact, piping in both the main helium circuit and in the purge gas lines was disconnected and reconnected, using direct maintenance techniques and equipment;

(g) The increased permeability of graphite in the outer housing of the second fuel element has been shown to be suitable;

(h) There was no problem with helium leakage; and

(i) Variations in heat flux along the fuel element did not cause any damage. Their effect is minimized due to the high thermal conductivity of graphite.

Operating staff and training

303. The Philadelphia Electric Company, the owner and operator of the Peach Bottom power station, began its training programme four years ago. In 1959, the man chosen to become station superintendent of Peach Bottom was assigned to the Shippingport reactor project for his initial training in nuclear plant operation. The training of other operating personnel has been in progress since 1961, with the result that about 25 men will be qualified for reactor operator licences. In February 1963, 15 Philadelphia Electric Company employees participating in the HTGR Operator Training Program were assigned to the General Atomic laboratories at San Diego for an intensive training programme of 17 weeks' duration. The programme included two types of

instruction. Under one aspect of the programme trainees attended two sessions per week in each of which staff members of General Atomic presented a two-hour lecture, followed by a two-hour question period. Subject matter covered in these sessions included reactor physics, core design, start-up experiments, core temperature distribution and heat balance, graphite and system materials technology, system chemistry, shielding design and reactor safety. Thirty-three such sessions were held. Concurrently, the trainees were given the opportunity to participate individually in the other aspect of the programme, which also consisted of 33 sessions for which the trainees themselves prepared lectures and presented them to the group. These lectures form the basis of a set of written system descriptions and treat such subjects as the main coolant system, helium purification system, reactor vessel and internals, steam generators, fuel transfer machine, spent fuel handling, control rod system, plant electrical system, and instrumentation. These trainees, who will be the operating staff of Peach Bottom, were also given the assignment of writing a set of operating procedures which will later become the operating manuals actually used in the station.

304. Another aid to operator training will be an analog computer simulator, which is being built at General Atomic and is to be delivered to Peach Bottom in September 1963 and installed in the reactor control room. The simulator has a console which is a duplicate of the reactor control console. An instructor's board is also provided by means of which the instructor can insert malfunctions into the simulator and observe the responses of the trainee as the latter attempts to take the proper corrective action.

305. The staffing plan for the Peach Bottom nuclear power station is given in the table below.

Table 22

The Peach Bottom high temperature gas-cooled reactor: Staffing plan

Category	Number of persons
Management and administration	
Station superintendent	1
Assistant station superintendent	1
Clerks, storekeepers, janitors and guards	11
Operations	
Plant engineer	1
Shift reactor engineers	5
Shift superintendents	5
Chief operators	5
Chief electrician	1
Plant mechanics	4
Mechanical operators	4
Auxiliary operators	5
Results engineer	1
Plant tests engineers	4
Instrument technicians	3
Supervisor - chemistry and health physics	1
Chemist	1
Health physicist	1
Technicians	2
Maintenance	
Maintenance foreman	1
Maintenance engineer	1
Mechanics and helpers	6
TOTAL	64

306. The normal operating force is six men per shift, including the shif superintendent, shift reactor engineer, chief operator, plant mechanic, mechanical operator, and auxiliary operator.

Selected references

307. A list of selected references concerning the Peach Bottom nuclear power station is given below:

Progress Report, Peach Bottom Atomic Power Station, prepared monthly by Bechtel Corporation and General Atomic for Philadelphia Electric Company and High Temperature Reactor Development Associates, Inc.

First Semiannual Report of Philadelphia Electric Company on the Peach Bottom Atomic Power Station, submitted to the US Atomic Energy Commission (23 August 1962).

Second Semiannual Report of Philadelphia Electric Company on the Peach Bottom Atomic Power Station, submitted to the US Atomic Energy Commission (23 February 1963).

40 MW(E) Prototype High-temperature Gas-Cooled Reactor, Research and Development Program, Quarterly Progress Report for Period Ending 30 September 1962, GA-3676 (15 April 1963).

HEYER, R.A. and RICKARD, C.L., The Impact of Graphite Fuel Development on High-Temperature Steam Generation, GA-4051 (15 March 1963).

Design of Gas and Liquid Waste Disposal Systems, Peach Bottom Atomic Power Station, Philadelphia Electric Company.

PAHLER, R.E., Report of the Objectives and Plans for the AEC's Civilian Power Gas Cooled Reactor Program, Paper presented at 9th Annual Meeting of the American Nuclear Society, Salt Lake City (17-19 June 1963).

IX. THE EXPERIMENTAL GAS-COOLED REACTOR

General

308. Initial criticality of the experimental gas-cooled reactor is not expected until October 1964. The civil works are generally proceeding well, as all structures and buildings have been erected and almost all of the concrete has been poured. The steam generator and turbine have been installed. However, most of the major equipment items for the nuclear plant are not yet in place. Installation of the reactor vessel, which began on 25 January 1963, is expected to require about a year to complete, after which the entire primary system, including reactor vessel, main helium blowers and steam generator will be checked out and given a pneumatic test. Construction and cold testing should be completed by April 1964. The EGCR time schedule, as estimated in May 1963, is given in the table below.

Table 23

The experimental gas-cooled reactor: Time schedule for the project

Item	Expected
Primary coolant blowers arrive at Oak Ridge:	
first	July 1963
second	September 1963
Charge machine - arrive at Oak Ridge	November 1963
Service machine - completely installed	July 1963
Reactor vessel - complete installation and pneumatic testing	March 1964
Phase 3 testing completed	October 1964
Initial criticality	October 1964

Construction experience

309. The major difficulties in this project have been encountered in the fabricators' shops. Construction in the field has been without any significant problems. One noteworthy accomplishment was achieved during the pouring of the 32 000 cubic yards of concrete required for the plant. Concrete was pumped through an 8-inch diameter pipe to a height of 138 feet at one stage of construction.

310. The quality of the welding has been good, with only about 2% of the welds requiring rework. This may be attributed to the availability of experienced, skilled welders in the Oak Ridge area. Some difficulty was encountered in welding the top section of the thermal barrier to the inside of the pressure vessel. Shrinkage of the steel thermal barrier took place as the welding proceeded and this had to be corrected. The lower section of the reactor vessel has been set in place. Welding of the top head, a large hemispherical section, 4 inches thick and 20 feet in diameter, to the cylindrical section of the vessel will be done in the field. The 370-ton service machine has been installed above the reactor. The 450-ton fuel charging machine is being assembled and checked out in the fabricator's shop. This machine will be shipped to the reactor site in November 1963, and will then require about three and a half weeks to install.

Design changes

311. Last year it was reported that the number of test loops to be installed through the core of the reactor had been reduced from eight to four. Now it has been decided not to install any loops. This decision was made in the interest of reducing construction expenditures, since the total funds available to AEC for construction was limited. The absence of in-reactor loops will curtail the utility and versatility of the EGCR experimental programme for fuel and coolant testing, since the loops could have been used to test other types of fuel matrix and fuel cladding, such as uranium carbide fuel or graphite or beryllium cladding. Also, the loops might have been used with coolants other than helium, such as steam or carbon dioxide. Under the present programme, the entire loading of fuel in the reactor will constitute an in-pile test, under the operating conditions of temperature and heat removal capability which obtain in the reactor.

Fuel management

312. Due to the circumstances mentioned above, it is likely that there will be little experimentation with the fuel in the EGCR. The objective will be to obtain a long operating life with the first core, perhaps with some shuffling or repositioning of the elements, then to discharge the entire first core loading at one time. However, it will be possible to insert fuel elements of advanced design in perhaps four channels of the first core loading. Fuel design

for the second core may be of improved design, up-rated in thermal power per foot of length or per channel. However, since the total capacity of the heat removal system is limited, it would then be necessary to reduce the number of fuelled channels in the core, leaving some positions unused.

Moderator life

313. Experiments at Oak Ridge have indicated that the blocks of graphite moderator in EGCR may crack after five to ten years of service. A limited amount of repair work may be possible in EGCR, perhaps involving the blocking of a few channels. However, in an advanced concept, it will be possible to provide for shuffling of moderator blocks during their life and replacement of graphite as necessary.

Safety features

314. Emergency equipment. The type of accident which might be of most concern in a gas-cooled graphite-moderated reactor is loss of coolant from the primary system or stoppage of coolant flow. To anticipate such a possibility, a separate emergency cooling loop has been added to EGCR. All principal components of this separate emergency cooling system, including two compressors and a heat exchanger, are housed in a steel-lined concrete cell (originally intended to be an experimenter's cell) adjacent to the reactor building, and the requirements for containment in both the reactor building and in the emergency cooling equipment cell are fully met.

315. The emergency cooling system can be used to remove decay heat from the reactor core even if both of the main coolant loops become inoperable. Core cooling must be available for removal of fission product decay heat for approximately 90 days after reactor shutdown. The emergency cooling system comprises three independent sub-systems, namely the heat removal system, the purge gas system, and the fission product removal system. The emergency heat removal system is designed to circulate helium, steam nitrogen, air, or any mixture of these following an accident in order to prevent runaway oxidation of graphite and to prevent additional fuel failures. The systems are designed so that credible failures of the main reactor coolant system cannot impair the heat removal capability of the emergency system. The purge gas system is used to blanket the reactor core graphite with nitrogen to minimize graphite oxidization until core temperatures are reduced to a level at which an unlimited

supply of air would not support self-sustained oxidation. Venting of the containment shell to the atmosphere to relieve overpressure is provided via the fission product removal system. The fission product removal system provides hold-up of fission product gases, charcoal traps for the removal of radioiodine, and filters to remove particulate activity before permitting the controlled release of gases to the atmosphere.

316. Containment shell spray cooling system. In addition to the provisions described above for emergency cooling of the reactor core itself, a spray system is provided for cooling the large containment shell (reactor building). The exterior surface of the containment shell can be sprayed with water, after an accident, to remove heat and prevent the pressure and temperature inside the shell from exceeding the design conditions. Quite apart from this emergency use, the spray cooling system may be operated for certain periods during the summer to remove solar heat absorbed by the shell. Although the present plan is to have no insulation on the containment shell, the original design called for an insulated building. Therefore the building heating and ventilation system, which is based on an insulated containment shell, probably will be inadequate to avoid temperatures above 105°F inside the reactor building unless some exterior spray cooling is utilized.

317. Safety margin in control rods. The design of the control rod system is such that if the rod with the greatest worth is removed from the reactor and if the rod with the second highest worth fails in the out position, the reactor can still be shut down with remaining control rods.

318. Analysis of maximum credible accident. The maximum credible accident is defined as that accident having an associated release of fission product inventory which would not be exceeded in terms of dose to the public, during the lifetime of the facility, by any other accident whose occurrence is credible. The maximum credible accident (MCA) in the EGCR is a composite accident and includes the most severe features of several depressurization accidents. For this reactor, the MCA is a depressurization accident (breach in main coolant system) which virtually stops the flow of coolant in the core region for 30 seconds and which results in internal failure of the steam generator in the coolant circuit that failed. Both helium and steam are released to the containment volume. The first condition of this accident would be possible with a relatively small rupture but not with a large one,

while the second condition (steam generator failure) would be caused only by a large rupture.

319. In order to appreciate the significance of the above statements, the two parts of this composite accident may be presented in detail.

(a) 30-second loss of flow in reactor core. Failure of the reactor outlet piping near the reactor vessel produces a pressure differential across the core such that the helium flow is upward through the fuel. Failure of the inlet piping near the reactor vessel produces a pressure differential across the core such that the helium flow is downward through the fuel. Hence, there exists, between the two cases, combinations of failure size and location which could result in little or no pressure differential across the reactor core, and coolant flow in the fuel zone could be virtually stopped. Such a condition could persist for a period up to 30 seconds until the system is depressurized. A breach of approximately 60 square inches in the reactor inlet piping near the blower discharge would have this effect. Fuel element failures would be at a maximum under these conditions, releasing fission products to the containment shell. Approximately 330 fuel elements would fail; and

(b) Failure of steam generator. It is assumed that a large breach in the helium coolant system might be accompanied by internal damage to a steam generator due to large pressure differentials across the tube bank, baffles or shroud. Such a failure would release steam to the containment shell. After the 30 seconds or so required for depressurization, a flow of gas through the core would be restored and no further failures of fuel elements would be expected. The pressure within the containment shell would rise to a peak of 8.9 psig, which would be reached approximately eight minutes after the accident.

320. Radiological consequences of the maximum credible accident. The submersion dose to personnel in the control room after the maximum credible accident is less than 1.5 mrem/h for periods up to 64 hours following the accident. The internal doses due to iodine-131, strontium-90 and caesium-137 are shown in the following table.

Table 24

The experimental gas-cooled reactor:
Internal dose to personnel in control room

Isotope	Organ	Integrated dose (rem)	
		24 hr.	8 hr.
I^{131}	thyroid	4.5	1.5
Sr^{90}	bone	1.1	0.33
Cs^{137}	lung	0.047	0.016

321. The dose at the boundary of the exclusion area and at the boundary of low population area is given in tables 25 and 26.

Table 25

The experimental gas-cooled reactor:
Doses at the exclusion radius, 1060 metres from the reactor, for two hours' exposure

Organ	Dose (rem)
thyroid	0.33
bone	0.023
total body	0.004

Table 26

The experimental gas-cooled reactor:
Dose at low population boundary, 4600 metres from the reactor

Time interval (hours)	Type of release	Thyroid dose (rem)	Bone dose (rem)	Total body dose (rem)
0-24	Leakage from containment shell	0.508	0.048	0.006
24-61.2	Leakage from containment shell	0.903	0.105	0.008
24-61.2	Venting from stack	0.262	0.010	0.021
61.2-92	Venting from stack	0.052	0.002	0.004
	Cumulative dose, 92 hr.	1.7	0.2	0.04

Operating staff

322. Some changes have been made in the staffing plan resulting in an increase of personnel from 120 to 168. The latest plan, giving the organization after the plant has come into full operation, is shown in the table below.

Table 27

The experimental gas-cooled reactor: Staffing plan

Category	Number of persons
Administration	
Manager EGCR project	1
Assistant manager	1
Administrative officer	1
Clerks, typists, accounting, stores	23
Operations group	
Operating superintendent	1
Assistant to operating superintendent	1
Plant operations	
Supervisors	5
Shift engineers	5
Unit operators	15
Assistant unit operators	11
Janitors	2
Plant maintenance	
Supervisor	1
Engineers	3
Engineering aide	1
Foremen	4
Craftsmen	17
Labourers	7
Technical programme group	
Technical programme superintendent	1
Engineering aide	1
Hazards control specialists	2
Control specialist	1
Reactor physicists	4
Experimental engineer	1

Category	Number of persons
Engineering analysis	
Supervisor	1
Engineers	6
Reactor engineering	
Supervisor	1
Engineers	5
Engineering aides	3
Chemical engineering	
Supervisor	1
Engineer	1
Chemists	2
Technicians	5
Controls engineering	
Supervisor	1
Engineers	3
Instrument foreman	1
Instrument mechanics	10
Radiological health section	
Supervisor	1
Health physicist	1
Technicians	5
Clerks	2
Public safety officers	10
TOTAL	168

Training

323. A broad programme of training of EGCR personnel has been carried out, and as of May 1963 the training has been largely completed. The training has been accomplished by classroom study, assignment to other reactor projects, assignment to conventional steam electric stations, and on-the-job training at the EGCR. The academic study included attendance at the Oak Ridge School of Reactor Technology, and the teaching of a special set of courses developed by TVA, the operating contractor for EGCR personnel. The latter series of courses

includes TVA Management Training (20 hours), Elements of Health Physics (20 hours), EGCR Basic Reactor Technology (219 hours), EGCR Process and Control System (690 hours), EGCR Plant Operations (150 hours), EGCR Supplementary Training for Operating Supervisors and Shift Engineers (90 hours) and EGCR Controls Engineering Training (77 hours). In addition to receiving classroom instruction, the five operating supervisors and the operating superintendent have received a combined total of 32 man-months' training and participation in operation of the Berkeley Nuclear Power Station in England plus two man-months at Calder Hall. Four other EGCR personnel spent periods of several months at the Berkeley Station. Shift engineers and unit operators received on-the-job experience with the Oak Ridge Research Reactor, Plutonium Recycle Test Reactor, and the General Electric Test Reactor. Three shift engineers are scheduled to spend four months each at the Consolidated Edison Thorium Reactor at Indian Point, New York. Two operations supervisors each spent one month at TVA's Kingston Steam Plant, a modern, conventional 1440 electrical megawatt station. Another method used for training of men preparing to qualify as licensed reactor operators is simulation of EGCR control behaviour with the ORNL analog computer. Each operator is scheduled to receive a minimum of 40 hours' instruction on the simulator early in 1964. Typical exercises to be performed by the operators include the following: plant heat-up, using main helium blowers, approach to criticality, effects of temperature coefficients, operation both at steady state and during power transients, effects of xenon, effects of changes in blower speed, blower control response to mismatch of power and flow, controlled shutdown, and reactor scram.

Cost data

324. The construction cost for the EGCR power station is expected to total about US $51 700 000, and the initial fuel fabrication cost will be US $1 100 000. Training of the operating and technical staff may cost up to US $2 million. In addition to these costs, there has been a significant research and development effort devoted to EGCR and gas-cooled reactor technology in general.

<u>Selected references</u>

325. A list of selected references concerning the experimental gas-cooled reactor is given below:

Experimental Gas Cooled Reactor, Final Hazards Summary Report, ORO-586 (10 October 1962).

Gas Cooled Reactor Program, Semiannual Progress Report for Period Ending 30 September 1962, ORNL-3372.

PAHLER, R.E., Report of the Objectives and Plans for the AEC's Civilian Power Gas Cooled Reactor Program, paper presented at the 9th Annual Meeting, American Nuclear Society, Salt Lake City, 17-19 June 1963.

ANNEX

Important design features of the Bradwell Nuclear Power Station

Location	Bradwell, Essex, England
Owner/Operator	Central Electricity Generating Board
Type	Natural uranium, graphite moderated and reflected CO_2 cooled
Power	
Gross thermal	1062 MW
Net electrical	300 MW
Over-all efficiency	28.2%
Fuel element	Natural uranium solid round rods
Form and composition	1.155 inches diameter, clad in magnox finned cans. Weighing 11.43 kg U per element. Total element weight 13 kg
Active core	
Dimensions	40.1 ft diameter, 25.9 ft height
Number of fuel channels	2564
Average power density	0.573 kW/litre
Pressure vessel	Spherical 66.5 ft internal diameter. Plate thickness 3 and 4 inches made of mild steel
Gas working pressure	132 psig
Gas outlet temperature	390°C
Gas inlet temperature	180°C
Reactor performance	
Mean fuel rating (heat)	2.22 MW/t
Maximum centre channel rating	3.77 MW/t
Total gas mass flow	5310 lb/sec
Centre channel heat output	253 kW
Centre channel mass flow	2.45 lb/sec
Maximum can temperature (Tcp)	481°C
Maximum uranium temperature (Tfp)	602°C

Fuel Handling

On load

Control rods

Total number per reactor	119	
Type: Bulk and safety (91)	Boron-bearing absorber sections between stainless steel	
Stability (28)	Stainless steel tube	
Turbine steam conditions	High pressure	Low pressure
Temperature	371°C	371°C
Pressure	745 psig	195 psig
Mass flow rate per set	342 000 lb/hr	126 000 lb/hr

Fig. 1 BRADWELL NUCLEAR POWER STATION - Reactor gas circuit diagram

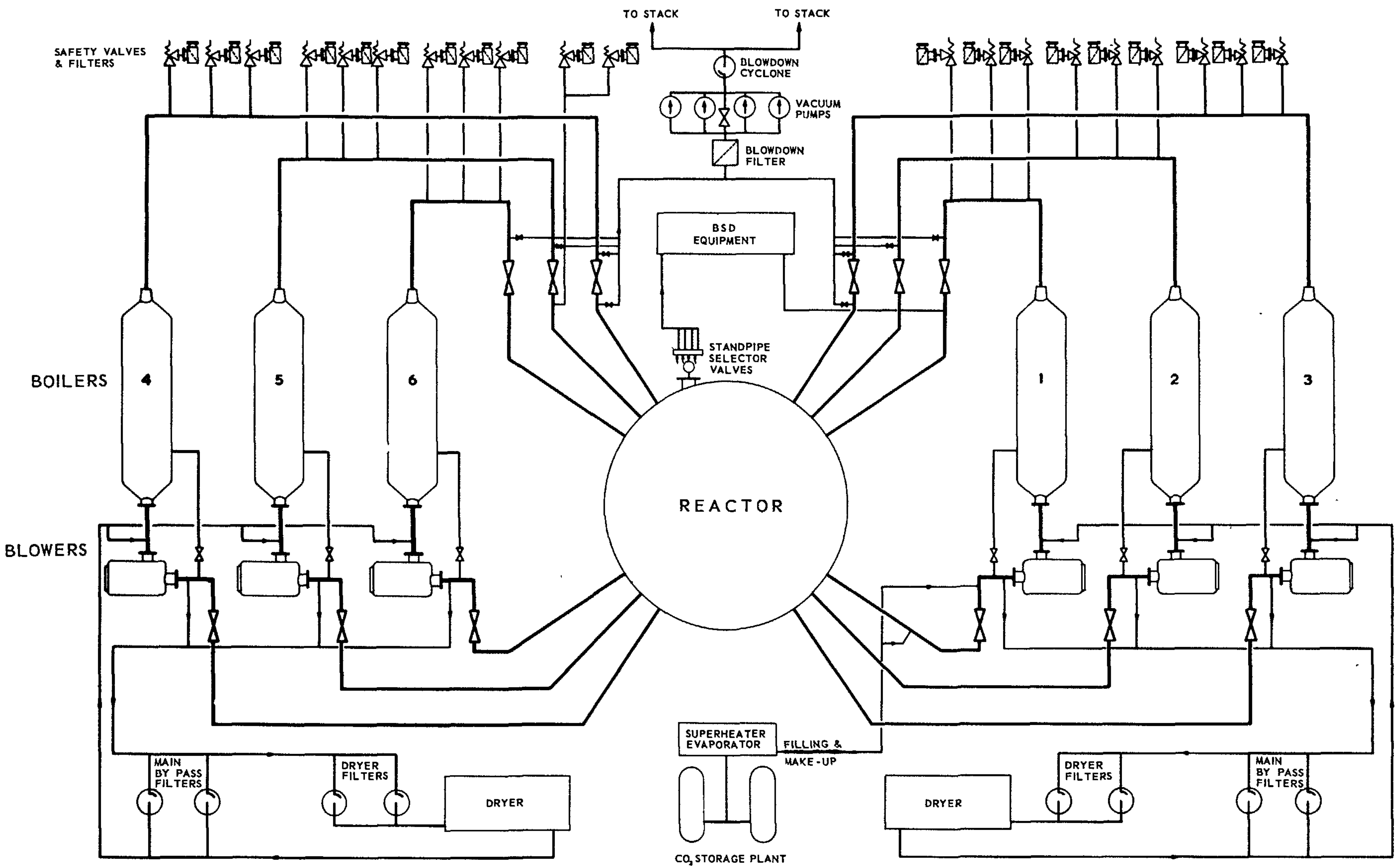

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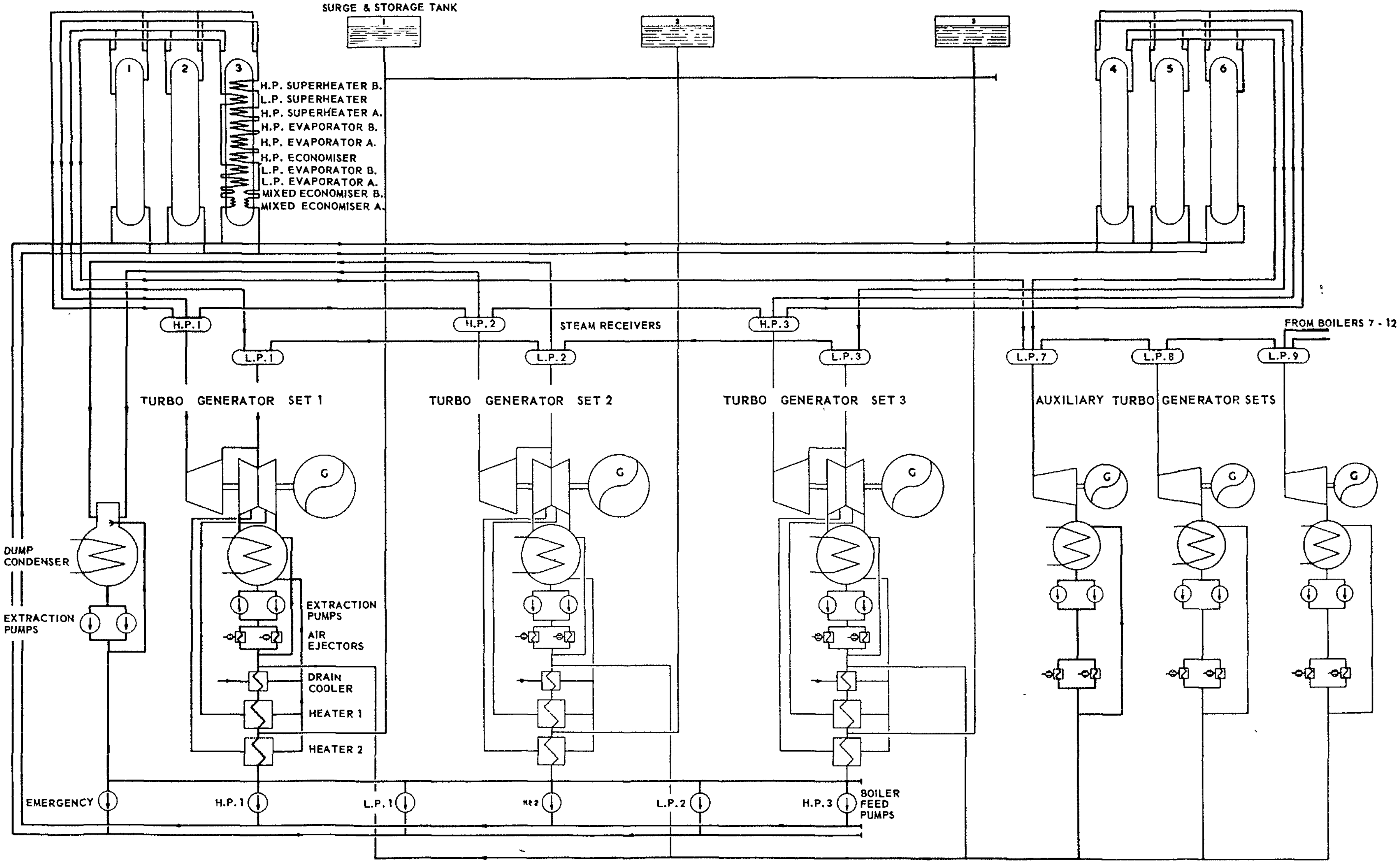
Fig. 2 BRADWELL NUCLEAR POWER STATION - Reactor steam circuit diagram



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