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SMALL POWER REACTOR PROJECTS IN THE UNITED STATES OF AMERICA AND CANADA

Information gathered as a result of invitations from Member States

CONTENTS

LIST OF ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safety
AECL	Atomic Energy Canada Ltd.
ANL	Argonne National Laboratory
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing Materials
BONUS	boiling nuclear superheater power station
BTU	British thermal units
BWR	boiling-water reactor
CANDU	Douglas Point nuclear power station
cfm	cubic feet/minute
CPPD	Consumers Public Power District
EBWR	experimental boiling-water reactor
EGCR	experimental gas-cooled reactor
ESADA-VESR	Empire State Atomic Development Associates Inc. - Vallecitos experimental superheat reactor
GAIL	General Atomic Irradiation Loop
GETR	general electric test reactor
GNEC	General Nuclear Engineering Corp.
gpm	gallons/minute
HNPF	Hallam nuclear power facility
HP	horse power
LACBWR	La Crosse boiling-water reactor
mil	one-thousandth of an inch
mills	one-thousandth of a dollar
mph	miles/hour
mr	milliroentgens
MWd	megawatt day
MWd/t	megawatt day/metric ton uranium
MWe	megawatt electric
MWth	megawatt thermal
n	neutrons
NPD	nuclear power demonstration
NRX	national reactor experiment
nvt	neutrons velocity time
OMRE	organic-moderated reactor experiment
OPHIR	organic power and heat industrial reactor
ORNL	Oak Ridge National Laboratory
ORSORT	Oak Ridge School of Reactor Technology

pH	hydrogen-ion concentration
ppm	parts/million
psi	pounds/square inch
psig	pounds/square inch gauge
PRAEC	Puerto Rico Atomic Energy Commission
PRWRA	Puerto Rico Water Resources Authority
rpm	revolutions/minute
SGR	sodium graphite reactor
SRE	sodium reactor experiment
t	metric ton
TVA	Tennessee Valley Authority
USAEC	United States Atomic Energy Commission
wt%	weight per cent
ZEEP	zero energy experimental pile

NOTE

All sums of money are expressed in United States currency.

A. INTRODUCTION

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

2. This report summarizes the information gathered on the small power reactor projects in the United States of America and Canada, as a result of the opportunity afforded by these Member States to the Agency.

3. It may be recalled that, at the third regular session of the General Conference, the United States Government 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 since June 1960 it has sent one or two staff members at approximately six-monthly intervals to follow the development of nine power reactor projects in the United States which represent six different reactor systems. Last year, the Agency issued a report [2] summarizing the information obtained through their visits and study of available published literature. The present document, which should be read in conjunction with that document, brings the information up to date and provides additional information on certain phases of the projects already discussed in the last report. Three more power reactor projects are also dealt with, namely the experimental gas-cooled reactor (EGCR), the high temperature gas-cooled reactor (HTGR) and the Hallam nuclear power facility (HNPF).

4. Early in 1962, the Canadian Government expressed its willingness to make available to the Agency relevant information on the NPD and CANDU projects. The coverage of the NPD reactor is based upon the published information supplied by AECL of Canada and the visit by one of the staff members to the NPD site.

5. The Agency wishes to acknowledge with thanks the co-operation extended by the atomic energy authorities in the United States and Canada in preparing this report and the help given by the reactor designers and contractors, as well as the utility companies concerned.

[1] GC(II)/RES/27, GC(III)/RES/57, GC(IV)/RES/86, GC(V)/RES/106.

[2] GC(V)/INF/41 (also published by USAEC as TID-8538).

B. UNITED STATES OF AMERICA

General

6. The small power reactor projects in the United States are part of the Civilian Power Reactor Program for the Development of Nuclear Power. A list of nuclear power plants of up to 80 MWe output under construction or planning is given in Table 1 below.

Table 1

Small power reactor projects in the United States

Reactor	Type	Output MWe	Location	Expected criti-cality	Owner/ Operator	Remarks
Saxton	PWR	3.25	Saxton, Pa.	1962	Saxton Nuclear Experimental Corp.	
ELPHR	PWR, low tempera-ture pro-cess heat	(40 MWth)	-	-	-	Deferred. Revised project has been pro-posed. See OPHIR
SSPWR	PWR	20.5	-	-	-	Deferred in-definitely
CVTR	Heavy water, pressure tube	17	Parr, S.C.	1962	Cardinas Virginia Nuclear Power Associates	
Elk River	BWR, in-direct cycle	22	Elk River, Minn.	1962	USAEC/Rural Cooperative Power Association	
Big Rock Point	BWR	50-75	Big Rock Point, Mich.	1962	Consumers Power Co.	
Humboldt Bay	BWR	48.5	Humboldt Bay, Calif.	1962	Pacific Gas and Electric Co.	
Pathfinder	BWR	58.5	Sioux Falls, S. Dak.	1962	Northern States Power Company	
BONUS	BWR, nuclear superheat	16.3	Punta Higuera, Puerto Rico	1962	USAEC/PRWRA	
LACBWR	BWR	50	Genoa, Wis.	1964	USAEC/Dairy-land Power Cooperative	La Crosse, BWR, for-merly ICRWR con-sidered for Los Angeles
Piqua	OMR	11.4	Piqua, Ohio	1962	USAEC/City of Piqua	

Reactor	Type	Output MWe	Location	Expected criti-cality	Owner/ Operator	Remarks
POPR	OMR	50	-	-	-	Deferred. Revised project has been proposed. See OPHIR
HNPF	SGR	76	Hallam, Nebr.	1962	USAEC/CPPD	Dry criticality achieved January 1962; wet criticality in August 1962
EGCR	GCR	22	Oak Ridge, Tenn.	1964	USAEC/TVA	
HTGR	GCR	40	Peach Bottom, Pa.	1964	Philadelphia Electric Power Company	
OPHIR	OMR	160 MWth 70 MW process heat, 27 MWe	Michigan	-	Under consideration	Combines objectives of POPR and ELPHR

I. THE ELK RIVER POWER REACTOR[1]

General

7. After several postponements and delays with respect to the original schedule, the Elk River reactor criticality appears to be well in sight. The start-up was delayed because of doubts concerning the integrity of the pressure vessel which had shown minute cladding cracks. This matter has now been resolved by the evaluation of the Advisory Committee on Reactor Safety (ACRS) which has recommended full power operation of the plant, limiting the life of the vessel to five years or 250 pressure and temperature cycles, whichever is earlier. Preparations are being made for criticality in October 1962. Modifications to the pressure vessel will have been completed by August 1962, after which several pre-operational tests, including some repeats, will take place. Fuel is already at the site and loading will commence by the end of September. The time schedule for various steps leading towards full power is given in the following table.

Table 2

Elk River reactor: Time table for full power

Item	Expected completion
Reactor vessel modifications	Mid August 1962
Repeat certain pre-operational tests	Mid September 1962
Fuel loading	End September 1962
Initial criticality	Early October 1962
First full power testing	End December 1962
28-day warranty run	End January 1963
60-day power run for training RCPA personnel	End March 1963
Hand over plant to RCPA	April 1963

System modifications

8. The more significant modifications in the system are as follows:

(a) Piping modifications were made to permit better control of heating during heat-up operation. It was observed that, if the start-up heaters alone were used, there was an undesirable water hammer problem. To solve this difficulty the shut-down heat removal pump is utilized for forced circulation of water during initial warm-up of the reactor;

(b) To assist natural convection flow of water, the start-up water will be relocated at the bottom of the reactor;

(c) To assure good air circulation inside the containment building, an induced draught fan has been added; and

(d) As a result of the recommendation of ACRS the following changes are being made in the pressure vessel:

(i) The 8", 10" and 16" nozzles to be modified by grinding to a suitable radius and then reclad; and

(ii) The thermal shield, 8" thermal sleeve, 10" deflector pan and feedwater distribution ring to be modified to permit inspection of certain nozzles and vessel surfaces. A suitable viewing equipment is being designed to carry out such inspections every year.

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[1] Information on the Elk River reactor project was given in paragraphs 54 to 143 of GC(V)/INF/41, which is supplemented and brought up to date by that given here.

Reactor pressure vessel

9. The reactor vessel has been the subject of prolonged controversy which as stated above has delayed the start-up of this plant. The reactor construction was essentially complete early in 1961 and criticality was envisaged towards the end of that year, but doubts regarding the ability of the vessel to meet performance criteria led to repeated postponements.

10. The reactor vessel specifications required that it be built according to the 1956 ASME code and nuclear code cases in effect at the time of procurement (1958). Before it left the fabricator (Pacific Coast Engineering Co., which has supplied several other reactor vessels) it got the seal of approval from the code authorities in the State of California. The vessel was delivered to the site in February 1960 and installed by July 1960. At that time, everything appeared normal and the main concern was about fuel element delivery because of the fabrication problems which had arisen in that connection.

11. The first signs of trouble with the vessel appeared early in 1961. After hydrostatic tests at 1875 psig (1.5 times design pressure) and subsequent heating to 425°F, visual observations indicated a network of circumferential bands of surface hairline cracks in the weld-bead overlay cladding (0.109" thick 308 L stainless steel) on the flange forgings on the vessel and head. It may be pointed out that 90% of the cladding in the vessel shell and head is roll bonded (304 as on 3" thick grade B carbon steel) and no cracks were found in this. These cracks could be repaired without difficulty since the primary purpose of the cladding material is only to protect the carbon steel from corrosion and it is not intended to lend any strength to the vessel. But the fundamental question was whether the cracks would propagate into the base metal and thereby endanger the safety of the vessel.

12. During investigations to establish the seriousness of the cladding problem, other potential problem areas were identified. Examination of records (such as radiographs) revealed that insufficient information existed upon which the adequacy of the vessel to meet the performance criteria could be fully established. A series of test programmes was initiated in November 1961 to supply the needed information, and the problems studied were:

(a) Cladding integrity;

(b) Strength-weld integrity; and

(c) Miscellaneous investigation of vessel adequacy (flange-to-shell match, stress analysis and nil-ductility temperature determinations).

Several experts and consultants were called in to help analyse the adequacy of the vessel. In general, the investigations consisted of a review of all vessel and vessel nozzle extension strength weld radiographs, an ultrasonic inspection of the vessel seam and large nozzle attachment strength welds, destructive testing and analysis of weld-on cladding, vessel flange material properties tests and additional stress analyses for several locations in the vessel.

13. One important part of this test programme consisted of grinding 1 x 1" patches in the weld overlay, down to the base metal, at more than 100 key locations on the vessel shell flange and nozzles and more than 50 spots on the vessel flange head. The patches were dye checked for cracks and etched with nitric acid to determine when the carbon steel base metal was reached. Grindings from about half the overlay samples were analysed chemically for nickel and iron content in a search for brittle areas. No evidence of cracks was found.

14. USAEC reviewed all available data and asked the opinion of ACRS. The Committee expressed the opinion that, in view of the past history of the vessel and several areas of uncertainty, certain restrictions should be placed on its use, and service period be

limited to five years or 250 pressure temperature cycles, whichever came first. Several improvements were recommended in the vessel and restrictions placed on the mode of its operation. The Committee left the door open about its ultimate future by indicating that, if during the five-year period new information which would permit a critical evaluation of the vessel was developed, it might be possible to re-examine the situation.

15. USAEC has accepted the opinion of ACRS and necessary modifications are being made to the pressure vessel in anticipation of start-up in October. It may be pointed out that the vessel has been cleared for full power operation subject to the conditions mentioned before.

Impact of the Elk River reactor vessel experience

16. The experience gained as a result of investigations connected with the Elk River pressure vessel has been very valuable. The techniques developed for testing the vessel will be useful in determining the suitability of other vessels. Already, the BONUS vessel has benefited from the information resulting from the Elk River vessel tests and its inspection quality has been greatly improved. Fabrication and testing methods for the LACBWR vessel are also being influenced by what has been learnt at Elk River.

17. It appears that there will be some further additions to the ASME nuclear code cases to reflect the latest know-how which is being developed. Existence of an up-to-date code is of great benefit, not only to those who prepare the specifications but also to the contractors and fabricators who comply with them. The adoption of a more refined and stringent code involving better quality control and inspection may add to the cost of a vessel but this increase will be insignificant as compared to the time lost in conducting modifications and repairs.

18. Another thing to watch is the actual operating experience after the Elk River plant goes on the line. If the vessel shows no signs of trouble and indicates no evidence of concern about crack propagation as a result of regular checks during the next five years, it might be possible to review the entire matter and perhaps extend the life of the vessel.

19. Meanwhile, USAEC is continuing its research and development programme concerning the pressure vessel testing and inspection and determination of the influence of hair-line cracks on its integrity. So long as definite results are not available, it is felt prudent to be conservative. This in essence is the line of thinking which has led to the imposition of certain restrictions on the Elk River vessel.

Pre-operational tests

20. Formal pre-operational tests were made during the period of September 1960 to July 1961. Since that time, the plant has been standing idle with only routine maintenance being performed. The core internals and control rod drives are to be re-installed before this programme begins in mid-September 1962.

21. In order to prepare the plant for operation, some re-testing will be necessary although it will not be an exact repeat of the original ones. These new tests will be more of system checks rather than tests of individual components and their proper installation. Those components (such as control rod drives, etc.) which have been disturbed will be re-tested and certain instruments and recorders will be recalibrated after long lay-up, before the system tests are made.

22. The main areas for pre-operational re-test are:

(a) Process and mechanical equipment. This includes pressure test on the vessel at 10% above design value (instead of 50% above design as prescribed by the old code), hot hydrostatic tests of the primary system, manual operation of control rods and drop test, emergency cooling system, purification system, building spray system containment leak rate, and water spray system (using air); and

(b) Electrical and instrumentation equipment. This includes all nuclear instrumentation, area radiation and process radiation monitoring systems, reactor level and temperature recorders, fission product monitor recorders, secondary system recorders and transmitters and evaporator and superheater temperature and flow indicators.

All these re-tests will be concluded in about four weeks' time.

Operating staff

23. During the unforeseen delay in plant start-up and the long period of relative inactivity, the Elk River reactor has lost some of its trained staff. Four shift instructors who were to hold key positions in the reactor plant have left. At the same time, over a dozen highly qualified persons from other plants and national laboratories have applied for the vacancies thus created. This indicates that in the United States there already exists a reservoir of trained and experienced reactor operators which can help make up any last-minute personnel losses. This, however, will not apply to those countries which may be starting out with their first nuclear power plants. It would seem advisable to take into account the fact that long delays in plant start-up could lead to a certain turnover in the staff and this might create difficulties.

II. THE BONUS POWER REACTOR[2]

General

24. The initial criticality of this 16.3 MWe nuclear superheat reactor has been delayed from December 1962 to about July 1963 principally because of the delay in the delivery of the pressure vessel. The difficulties which the fabricator has had with the Elk River vessel have influenced the delivery time for the BONUS vessel. Every other phase of design and construction is progressing well, and as of June 1962 construction was 63% complete as against 65% scheduled. The actual construction costs are running slightly ahead of the original estimates.

Table 3

The BONUS power reactor: Revised time schedule for the project

Item	Original estimate	Actual or expected
Start of construction	August 1960	August 1960
Start of personnel training	August 1960	August 1960
Erection of containment shell	June 1961	August 1962
Pressure vessel installation	October 1961	October 1962
Turbogenerator and steam equipment	August 1962	December 1962
Completion of training	September 1962	December 1962
Construction essentially complete	November 1962	May 1963
Cold testing	September 1962	May 1963
Initial criticality	December 1962	July 1963
Full power operation	February 1963	October 1963

Design changes

25. As a result of constant review by the designer, some modifications have been made in the original design; the important ones are summarized below.

26. Originally, the enrichment for the boiler fuel was contemplated to be 1.85% and for the superheater fuel 3.5%. These enrichments have now been changed to 2.4% for the boiler and 3.25% for the superheater. The new enrichment combined with the fuel-management scheme discussed later will enable the reactor to achieve 10 000 MWd/t burn-up.

27. The pressure vessel thickness has been increased from 2.75" to 3.12" with consequent weight increase from 57 to 61 t. This will enable the vessel to have a greater safety margin.

[2] Information on the BONUS reactor project was given in paragraphs 238 to 288 of GC(V)/INF/41, which is supplemented and brought up to date by that given here.

28. In order to assure complete dryness of the steam entering the superheater, a set of tubes heated with exit superheated steam has been located above the superheater. Before entering the superheater, 2% moist steam from the boiler first passes through conventional corrugated plate-dryers which reduce the moisture to below 0.1%. The remaining traces of moisture are removed by passage over the heated tubes which increase the temperature by about 2.5 ^{o}F to 542.5 ^{o}F.

Reactivity control

29. The BONUS reactor has 17 top-mounted motor-driven control rods with rack and pinion drives. Nine cruciform rods are located in the boiler region designated as 1 central rod, 4 side rods and 4 corner rods, each 0.125" thick with a span of 7.75". Eight slab rods called "blades" are situated between the boiler and the superheater region, each 0.125" thick with a span of 14.25". The length of the poison section for each of the 17 rods is 62.375", and the composition of the poison consists of 1.09 wt% boron in stainless steel.

30. The initial U-235 enrichments in the boiler fuel and the superheater fuel are 2.40% and 3.25% respectively. A neutron absorbing shim is used initially in the central moderator space of each boiler assembly. Shims are also located at the interfaces between the boiler region and the superheater regions. With these shims in place, the control rods are able to shut the initial cold clean reactor down with a margin of 3% k. The auxiliary boron injection system is capable of shutting the cold clean reactor down by 5% k even with all control rods out and with no shims in the reactor.

31. The core is designed for an average integrated exposure of 10 000 MWd/t in both the boiler and the superheater. In addition to the reactivity held in control rods to compensate for reactivity loss due to burn-up, the removal of shims, and of the four natural uranium boiler fuel assemblies which are initially loaded at the centre of the core, provide the additional reactivity needed. There will be an inward shifting of the outer fuel elements as burn-up progresses. This will assist in flux flattening, in addition to reducing total reactivity requirements.

Flooding and unflooding the superheater

32. The changes in reactivity resulting from flooding and unflooding the superheater core have been investigated both theoretically and experimentally. The flooding coefficient depends upon the fuel-moderator ratio in the superheater and upon the coupling between the superheater and boiler. In order to minimize the absolute value over the entire range of operating conditions, the fuel-moderator ratio has been chosen so as to give a slightly negative temperature coefficient at room temperature and a slightly positive one at designed operating conditions. The cold mock-up studies indicate that the measured reactivity change due to flooding of the superheater ranges from -0.30 to -0.64% $\frac{\Delta k}{k}$ depending upon the control rod position. The calculated value of reactivity change due to flooding of the superheater under hot conditions shows a net gain of 0.16% $\frac{\Delta k}{k}$.

Normal reactor start-up and shut-down

33. The start-up and shut-down of an integral nuclear-superheat reactor is different from that of an ordinary boiling-water reactor because of the need to assure necessary cooling for the superheater fuel elements. In the BONUS design, the superheater is kept flooded in the initial stages of start-up and gradually drained as the conditions permit. The major steps in a normal start-up are summarized below.

34. At first, the comprehensive pre-start-up procedures are completed to make sure that the equipment and controls are in proper order. The water level in the reactor is brought to normal and flow is established in the circulation loop. The pre-heater starts warming up the reactor to 400 ^{o}F and 230 psig. The superheater assemblies are drained by opening the valves, and the temperature is raised to 525 ^{o}F. The automatic control

of the reactor water level and pressure is started. At 525°F the reactor is made critical by withdrawing the boiler rods, and the power increases to produce 20 000 lb/hr steam. The pre-heater is turned off and control rod in the superheater withdrawn to make it critical and gradually increase the steam temperature to 700°F. From then on, the steam output and temperature is gradually increased to rated values and a power split of 74% to 26% is maintained between the superheater and the boiler by manipulation of control rods and shims.

35. To shut down a reactor under normal circumstances, the power is reduced to 30% of rated value and steam by passing to the main condenser. The superheater rods are inserted full in. The power output is decreased to 10% and the boiler rods introduced to make it sub-critical. The vessel is depressurized by blowing steam to the condenser, keeping the cooling-down rate to 125°F/hr. The superheater is flooded and reactor gradually cooled down.

Reactor power control

36. The BONUS plant has been designed for base load duty and no automatic power regulation has been incorporated but a manual power regulation system. The control of an integral BWR superheater requires maintenance of approximately constant reactor pressure and superheater steam temperature at varying power outputs. To keep proper power-split between the boiler and the superheater, preferential control of each region is essential.

37. Assuming that the reactor is in operation and it is desired to increase its power output, the operator normally withdraws the control rod in the boiler region, thus adding a small increment of reactivity. Reactor steam output increases along with the voids which tend to offset the effects of added reactivity, and a new power level is established. Increased steaming rate will tend to push reactor pressure up, increasing the steam flow through the superheater. Since the withdrawal of the control rod also increases the power output of the superheater, there is an initial rise in the exit steam temperature which later on falls to an equilibrium value determined by the new superheater power. In case of the withdrawal of the control rod only, the ratio of boiler to superheater output will increase slightly and the exit temperature will fall somewhat. To raise the temperature of the steam, the operator moves the eight slab rods to readjust superheater output until original steam temperature is restored. Thus, while the boiler rods control the amount of steam generated, the slab rods influence the exit steam temperature.

38. Besides using the central control rod for readjusting reactor power, the output of the boiler region can be controlled by varying the circulating water rate. Under steady state conditions, the steam production rate at constant pressure is directly proportional to the water recirculation rate. The actual performance of this method of control will indicate its feasibility as an automatic power control system.

39. In order to control the temperature of the superheated steam entering the turbine and minimize the movement of control rods during power level changes, a conventional attemperator spray controller is installed in the main steam line of the primary system. Thus, even if the reactor exit temperature were to rise to 950°F, the temperature of the steam delivered to the turbine could be maintained at 900°F.

Training of personnel

40. The training of operating staff for the BONUS plant has several interesting features. The staff is drawn entirely from Puerto Rico and consists of relatively young people who did not have any prior experience in running nuclear or conventional plants, except for two of the shift supervisors who have worked in steam plants. But their choice seems to be very good and their record during the training period has been excellent. This procedure offers many parallels with the developing countries where similar situations

with regard to the lack of technically qualified people exist. It is interesting to examine in detail the training programme for BONUS personnel because it affords a good example of how a small utility such as PRWRA can start from scratch and build up a qualified, competent team from among its own personnel to take full charge of the operation and management of a nuclear plant in a relatively short time.

41. The training programme can be divided into two phases: one before starting the actual feasibility and conceptual study of the BONUS plant, and one afterwards, when this project became a reality. Soon after PRWRA became interested in the possible use of nuclear power in 1955, it sent a small team for advanced training abroad. At this stage, the object was to know as much as possible about the most promising power reactor types at that time; consequently, the trainees were engaged in studying numerous reactor systems such as the organic, aqueous, homogeneous, pressurized-water and boiling-water types. The actual decision about the reactor type was not made until three years later in December 1958, when PRWRA had already at hand a small core of trained persons and had accumulated sufficient information and experience to make a final choice. The feasibility study took another year and the contract for the BONUS plant was signed in January 1960. The construction started in August 1960 with scheduled start-up in December 1962.

42. The experience of PRWRA indicates that for a utility such as this, serving a region which is not highly industrialized, it takes about eight years between the conception and fruition of a nuclear power project. It also shows that it is necessary to start training a small number of specialists soon after a utility begins considering the possible application of nuclear power. These specialists can be extremely useful in conducting the comparative studies of nuclear versus conventional alternatives, and selection of a suitable reactor system and making the necessary plans for the integration of the nuclear plant into the system.

43. During the first phase between 1955-58, PRWRA sent six engineers for training to various AEC national laboratories and universities in the United States. One electrical and one mechanical engineer were sent to Argonne in November 1955; the mechanical engineer, after completion of the course at the International School of Nuclear Science and Engineering, stayed on for on-the-job training with the Reactor Division of the Laboratory. He was assigned to the staff crew in charge of erection, start-up, testing and operation of EBWR. The electrical engineer, after attending the course at Argonne, continued advanced studies in nuclear engineering at the University of Michigan and specialized in control dynamics and stability of boiling-water reactors. In 1956, a chemical engineer was sent to the Oak Ridge National Laboratory to attend the Oak Ridge School of Reactor Technology. Afterwards, he stayed on with the Reactor Engineering and Experimental Division for one year, during which he participated in the start-up, testing and early operation of homogeneous reactor test (HRT-2). At about the same time, an electrical engineer was sent to the University of New York, who, after his M.Sc. in nuclear engineering, worked at Atomics International, dealing with the engineering evaluation of the organic moderated power reactor. Still two more mechanical engineers obtained their M.Sc. in nuclear engineering from the University of New York and from the Puerto Rico Nuclear Center.

44. Meanwhile, the negotiations between USAEC and PRWRA led to the signing of a contract in December 1958 for the BONUS feasibility study. In January 1960, the final contract for the design and construction was signed. Serious consideration was given to the training programme of the personnel responsible for the operation of the plant.

45. PRWRA, in consultation with USAEC and GNEC, prepared a comprehensive plan for the training of personnel, the basic objectives of which were:

(a) To give participants a broad perspective of the nuclear energy field;

(b) To give specific theoretical background in nuclear engineering;

(c) To give operating experience in both conventional equipment and nuclear reactors and auxiliaries;

(d) To make participants thoroughly familiar with the BONUS power plant; and

(e) To prepare for the AEC licensing examination.

46. In order to fulfil these objectives, the general programme was divided into three phases. In each phase supervisors and operators go through specific training at various locations, except during phase 3, when all of the participants have the same training schedule.

47. Supervisors' training. Phase 1: Three mechanical and two electrical engineers were chosen to attend the Nuclear Reactor Operations Supervision Course at ORSORT which started on 1 September 1960. This course is designed to prepare engineers to supervise the operation of power reactors. Emphasis is laid on the supervisor's responsibility for an operating reactor and his acquisition of technical competence to carry that responsibility. The course consists of a year of classes, laboratory instructions and participation in reactor operations.

48. Of the five engineers, one electrical and one mechanical completed the course successfully. The other three engineers, after nine months' work at ORSORT, were transferred to Argonne National Laboratory (ANL) to start, in June 1961, phase 2 of the training. Those who stayed in Oak Ridge started phase 2 at Argonne two months later (August 1961).

49. Phase 2: Eight engineers - five mechanical, two electrical and one chemical - are undergoing training in this phase of the programme. Six of them will act as shift supervisors, one will be the instrument engineer and one the health physicist. The last one is a chemical engineer who has successfully completed the course at the Puerto Rico Nuclear Center obtaining a masters degree in nuclear science. It was the object of this part of the programme to give PRWRA's engineers the opportunity to familiarize themselves with the operation of a boiling-water reactor like EBWR. Since the original agreement between ANL, AEC and PRWRA was to start training only in October 1961, rather than in June 1961 as it happened, hasty arrangements had to be made under the most adverse conditions especially due to the status of EBWR which was being modified for higher power operations. Accordingly, a schedule was worked out based upon the availability of EBWR for training.

50. During the first three months, PRWRA's engineers were introduced to the facilities available at the ANL International Institute. They attended a series of lectures given at the Associated Midwest Universities group to study NBWR, observed its operation, and performed experiments on the AGN-201, Argonaut and CP-5 reactors. Later they participated in the loading and unloading operations at the CP-5 research reactor.

51. After unsuccessful efforts to get EBWR made available for the training of PRWRA's six shift supervisors, it was decided, with the consent of the Oak Ridge AEC office's representative, to utilize the CP-5 reactor. At the present time, PRWRA's engineers are on shift work on this reactor, where opportunities are given to them to log a maximum number of start-ups and shut-downs in the short time available before starting phase 3.

52. A special programme has been worked out for the instrument engineer and the health physicist. The first has been assigned to the Instrumentation and Control Division and the second to the Industrial Hygiene and Safety Division.

53. The progress reports from ANL show PRWRA's engineers to be performing according to expectations.

54. Phase 3: Starting on 5 March 1962, all shift supervisors, instrument engineer, health physicist and reactor and plant operators will go through an on-the-job training at the BONUS site, which constitutes phase 3 of the training programme. During this time, they will follow the construction of BONUS and be given an extensive course of lectures

using the mock-up console as a training aid. The participation of GNEC, as reactor designer, is of utmost importance. Arrangements were made among AEC, PRWRA and GNEC to make possible this co-operation. All interested organizations agree this phase 3 to be the most important period in the training programme.

55. The general subjects to be taught all related to the BONUS plant are the following:

(a) Details of the plant system;

(b) Details of reactor internals;

(c) Details of auxiliaries;

(d) Details of electrical systems, instrumentation and controls;

(e) Physics;

(f) Abnormal behaviour of the plant and safety;

(g) Operating procedures;

(h) Supplementary work; and

(i) Trial examination.

56. Operators' training. Twelve men were chosen to undergo training as reactor and plant operators for the BONUS plant. Minimum requirements for selection were twelve college credits in mathematics, eight in chemistry and eight in physics. The qualifications of almost all those selected were above these requirements. The training programme formulated for these 12 men necessarily has to cover the conventional part of the plant as well as the nuclear part. In this particular case, phase 1 and phase 2 were considered as stages to prepare these men for phase 3 of the programme.

57. Phase 1: The operators in training were assigned in April 1961 for a four-month period to PRWRA's San Juan Steam Plant. This plant was chosen because it is the only one in PRWRA's system that has units of 20 000 kW which are similar to the one to be installed at BONUS. A series of intensive lectures was given covering fundamentals of electricity, mechanics and mathematics. The detailed outline for these courses is given in PRWRA's document of 1 March 1961.

58. Besides the lecture courses, the students were given the opportunity to familiarize themselves with the equipment and operation of the conventional plant under the direct supervision of one of PRWRA's experienced shift supervisors. All operators in training completed satisfactorily phase 1 and started immediately phase 2.

59. Phase 2: Since August 1961, the 12 operators in training have been attending the Puerto Rico Nuclear Center at Mayaguez, Puerto Rico. Basically, phase 2 consists of a series of lectures on the following subjects: reactor fundamentals, health physics, physics and engineering, mathematics, kinetics and instrumentation and Puerto Rico Nuclear Center reactor operation. Laboratory work supplements these lectures. The students' performance up to the present time can be regarded as satisfactory.

60. Phase 3: This part of the training will be the same as that for supervisors which has been discussed in paragraphs 47 to 55 above.

61. Throughout all these years, PRWRA has assumed full responsibility for training and licensing of the PRWRA operating staff for the BONUS plant. In order to do this satisfactorily, PRWRA has requested and obtained at all times the full co-operation of representatives of both the Oak Ridge AEC and PRAEC offices. Always PRWRA has tried to follow, even though it represents additional expenditures, the advice of AEC representatives. PRWRA has co-ordinated part of the training with GNEC because it is felt that their help is extremely necessary to round-off the training programme.

62. PRWRA is willing to discuss the training at length and is open to suggestions that may assure the success of the BONUS project.

Experience in design and construction

63. Pressure vessel fabrication. There has been a considerable delay in the completion of pressure vessel fabrication which, in turn, is holding up the project as a whole, costing four thousand to five thousand dollars a day in overhead expenses. It may be pointed out that the BONUS vessel is not in the same situation as the Elk River vessel because it has not been installed and is still in the shops of the fabricator where it can be subjected to all kinds of inspection and tests.

64. The BONUS vessel calls for all weld overlay of stainless steel, while the Elk River vessel was mostly roll bonded and only partly overlaid. This vessel started with a higher standard of basic specifications. After the Elk River experience, testing and inspection standards were made more rigorous, placing greater emphasis on quality control. Some errors in fabrication were discovered and in many instances the defective overlay had to be removed and redone. A vigorous in-plant inspection programme is continuing and a number of tests will be performed before delivery. The vessel is expected to be delivered by October 1962 after undergoing very thorough examination. There seems little doubt that the BONUS vessel will be a very high quality vessel.

65. Chloride-stress corrosion. One of the major problems in connection with nuclear superheat reactors is the development of a suitable superheater fuel element which will not fail under severe temperature and pressure conditions to which it will be subjected.

66. The cracking of stainless steel because of chloride-stress corrosion is not a new problem associated only with nuclear reactors. It has been frequently encountered in the superheat sections of conventional boilers and other similar equipment. But in this case the remedy is relatively simple. One can start with thicker tubes to reduce the chances of failure and the failed tubes can be easily replaced.

67. The consequences of failure in stainless steel cladding of a power reactor fuel element can, however, be more serious because of the possibility of contamination of the system and of release of fission products. Therefore the cracking of stainless steel cladding of fuel elements cannot be tolerated and every possible measure has to be taken to prevent it.

68. At this stage the phenomenon of chloride-stress corrosion is not fully understood and an extensive research and development programme supported by USAEC is under way to determine the exact nature of the problem and possible remedies.

69. This research and development programme covers several areas, including:

(a) Fuel development and testing: for example, nine different types of fuel elements (five inconel, three incaloy and one low-carbon stainless steel) arecurrently under test and several more will be tested in the near future;

(b) In-pile loops;

(c) Construction of ESADA-VESR nuclear superheat facility at San José;

(d) Out-of-pile corrosion tests: the effect of various parameters such as steam, oxygen, hydrogen and chlorine on different Ni-alloys and low-carbon steels and other materials is being studied;

(e) Steam water separation;

(f) Development of steam and water flexible mechanical seals; and

(g) Feasibility of a mixed-spectrum superheat reactor: this would have a fast central region for superheating and a thermal outer region for boiling. Since neutron economy would not be too much of a problem in the central portion, thicker stainless steel cladding could be used, thus avoiding the problems of chloride corrosion.

70. What is currently surmised is that minute traces of chlorine when carried over to the superheater deposit on the cladding surfaces and, with the passage of time, this deposit increases. The chlorides attack the grain boundaries of the crystal structure of the stainless steel cladding which then crack and the cracks propagate, leading to a failure of the cladding.

71. It appears that the following factors contribute towards such failures:

(a) Presence of moisture in the steam entering the superheater section. This can be avoided by thoroughly drying the saturated steam before it is admitted into the superheater fuel assemblies. Satisfactory progress has been made in this direction and steam 99.9% dry can be easily obtained;

(b) Presence of chlorine and other similar impurities in water. The nuclear superheat reactor calls for a very strict control of system water purity to remove last traces of chlorine, etc. Although this represents some additional cost, it can be easily achieved;

(c) Inherent stresses in the cladding material. It appears that the latent stresses in the cladding itself may have something to do with the cracking, although more research is necessary before any firm conclusions can be drawn;

(d) Alternate wetting and drying. This contributes to the accumulation of chlorides on cladding surfaces. Once the plant is in operation, wetting and drying cycles can be greatly reduced but in the initial stages many such cycles have to be tolerated;

(e) Steel composition. This appears to be one of the major factors. Results obtained so far indicate that the presence of nitrogen, carbon and oxygen enhance stress corrosion cracking while high Ni-alloys resist it. Experiments are under way to test such materials as inconel, incaloy, hestaloy and Types 304, 316, 348 and 406 steels; and

(f) Condensation. The condensation of moisture on superheater elements may leave a harmful deposit of chlorides.

Fuel fabrication

72. Several unforeseen problems have arisen in connection with the fabrication of boiler and superheater fuel elements. These have forced a delay in the delivery of fuel but the postponement of the criticality date is primarily due to the pressure vessel delays rather than fuel element fabrication schedule.

73. Boiler fuel. In this case, three principal problems were encountered, namely:

(a) In assembling fuel rods it was observed that the lower spring yielded under the the weight of the pellets and did not return to its original length. With respect to the first core, it has been decided to accept this spring, with some modifications, because this is not considered to be a critical part. For subsequent cores, however, a design change is envisaged;

(b) It was difficult to achieve a satisfactory weld between the thick end cap and thin tube because of great difference in thicknesses of materials involved. The specifications called for no porosity and 100% penetration. While the desired penetration was achieved, it was felt that a little bit of porosity will have to be tolerated; and

(c) The standards of inside finish prescribed for Zircaloy tubes were hard to meet.

74. Superheater fuel. As in the case of other nuclear superheat reactors, the BONUS reactor faces the problem of developing satisfactory fuel assemblies for the superheater region which would not fail due to chloride-stress corrosion. Several improvements have been incorporated to protect the assemblies against stress corrosion failures. At the steam water interface some chloride build-up is expected. Therefore the outer pipe is now made of inconel instead of stainless steel. The tubes of the preheater-dryer have also been changed to inconel. To hold the superheater rods together, brazed joints are needed and the question is how much chlorine is retained in the joint. Although most of the chlorine is expected to disappear in the brazing cycle, some traces could be left. The significance of this is being studied. Another aspect requiring attention is that the cleaning of fuel assembly surfaces requires HCl. This can be washed away but some traces might remain. It would be better to avoid HCl altogether but the manufacturer feels that the development of a new method for cleaning may be too costly and it will be possible to remove HCl to the desired extent. The end cap welding problems, similar to those in the boiler elements, were encountered and duly resolved.

75. Since the power density in the BONUS superheater section is low (11.6 kW/l) and the surface area rather large, the chances of the occurrence of chloride-stress corrosion are proportionately reduced.

76. The possibility of having an alternate superheater core made with inconel or hestaloy instead of stainless steel is also being considered.

Fuel management

77. The fuel management plan devised for the BONUS reactor is of particular interest, for several reasons:

(a) Although the physical size of the fuel region is relatively small, the boiler fuel and the superheater fuel are each divided into four zones. After equilibrium fuelling has been established, every fuel assembly will reside for a period of time in each of the four fuel zones; and

(b) Upon discharge, all enriched fuel from the boiler and from the superheater regions will have accumulated the same average exposure, 11 020 MWd/t, which holds for the initial discharges as well as all subsequent discharges. Initially, four of the 64 boiler assemblies will be fuelled with natural uranium rather than enriched uranium, and these four assemblies may be discharged at an exposure different from the enriched fuel.

78. In order to make such a fuelling programme possible, it was necessary to consider how the excess reactivity in a fresh, clean reactor changes as the equilibrium fuel mixture is approached, and to provide appropriate reactivity control and adjustment of the control capability.

79. The fuel enrichments in the initial reactor loading were selected to make possible average exposures of 11 020 MWd/t in all batches of enriched fuel removed from the reactor, for both the first batches discharged and all subsequent discharges. Sixty of the boiler fuel assemblies contain uranium enriched to 2.4% U^{235}, while the four central positions are fuelled with natural uranium. All superheater fuel contains uranium enriched to 3.25% U^{235}. The initial average enrichments of both boiler and superheater are considerably higher than the average enrichment will be after several partial refuellings, at which time the reactor will contain a mixture of fresh fuel and fuel exposed

in varying degrees. It is found that excess reactivity of the core will be considerably higher during the very early stages of reactor operation when following this fuelling programme than it will ever again be during the remaining life of the reactor. In order to avoid the very large distortions of power distribution that would occur if this excess reactivity were compensated only by control rods, absorbing shims are used in the boiler and in the gap between the boiler and the superheater during initial operation of the reactor, until the equilibrium refuelling cycle has been approximated. These shims, made of boron steel, are removable, and it is planned that they will be replaced by shims of lesser absorption during the early life of the reactor, and they will eventually be removed altogether. It is expected that the first change of shims will be made when the average exposure of all fuel in the boiler reaches about 2750 MWd/t. The new boiler shims will be uniformly weaker. When the boiler fuel reaches an average exposure of about 5500 MWd/t, the shimming will again be reduced. At this time, the four natural uranium fuel assemblies (centre of core) may or may not be replaced as dictated by reactivity considerations. Reactor operation is resumed until the average exposure of boiler fuel reaches approximately 8250 MWd/t, at which time the 16 centremost fuel assemblies will have reached about 11 000 MWd/t. These 16 central assemblies will be removed, the remaining 48 assemblies will be moved toward the centre, and 16 new assemblies will be loaded at the edges of the boiler region. From this time onward, new boiler fuel will be added as necessary to maintain reactivity, 16 assemblies at a time, according to the same pattern. At every second boiler fuel reload, the superheater fuel will be rotated 180 degrees about its vertical axis and the absorption of the remaining boiler-superheater shims will be adjusted, until the superheater fuel has reached an average exposure of 8250 MWd/t. At this time the two superheater assemblies closest to centre of each side of the square core configuration should have reached about 11 000 MWd/t. These eight superheater assemblies will be discharged, the remaining superheater assemblies will each be moved one position toward the centre lines, leaving the corners vacant, to be filled by eight new superheater assemblies. Thereafter, eight new superheater assemblies will be added, following the above pattern, at every second boiler refuelling. This fuelling pattern will then be repeated throughout the life of the reactor.

Revised cost data

80. There have been some changes in the estimates presented last year.[3] Plant costs have gone up by about 15% while the estimated fuel costs/kWh for the first core have increased from 4.1 mills to about 5.1 mills.

81. Plant costs. The new plant cost estimates are given in the table below.

[3] GC(V)/INF/41, Tables 20 and 21.

Table 4

The BONUS power reactor: Revised cost estimates

(In thousands of dollars)

Item		Cost
	A. PLANT CONSTRUCTION	
I.	Engineering design and inspection	2 294
	Title I, III, IV	2 001
	Start-up management	293
II.	Direct construction costs	6 188
	Power plant building	2 304
	Reactor plant building	3 557
	Accessory electrical equipment	164
	Miscellaneous power equipment	163
III.	Indirect construction costs	896
	Sub-total (I to III)	9 378
IV.	Contingencies	422
	Sub-total (I to IV)	9 800
	B. OPERATING EXPENSES	
I.	Research and development	1 736
II.	Operators and start-up training	258
III.	Fuel fabrication (first core and reserve)	1 236
IV.	Contingencies	270
	Sub-total (I to IV)	3 500
	TOTAL USAEC costs	13 300
	TOTAL PRWRA costs	4 800
	TOTAL BONUS project costs	18 100

82. Fuel costs. The current estimate for the BONUS fuel costs, based upon actual contract price, is as follows:[4]

[4] Latest revisions in prices for uranium will lower the costs a little.

Table 5

The BONUS power reactor: Estimated fuel cost

(in dollars/kg U)

Item	Boiler	Superheater
Fabrication	178.30	373.70
Transportation	18.37	18.37
Chemical processing, conversion, losses	55.00	54.16
Burn-up	123.50	147.75
Plutonium credit	- 39.75	- 28.80
Use charge	32.83	98.00
	368.25	663.18
Annual fuel cost for reactor		
Boiler	338 000	
Superheater	214 000	
TOTAL	552 000	

83. Based upon the generation of 1.07×10^8 kWh (electric) per year, the unit fuel cost is 5.17 mills/kWh.

84. In the above calculations, the following assumptions were made:

Boiler	37 MWth
Superheater	13 MWth
Irradiation level	11 020 MWd/t for boiler
	11 020 MWd/t for superheater
Plant factor	75%

III. THE PATHFINDER POWER REACTOR [5]

General

85. The construction work at the Pathfinder atomic power station is essentially complete. By the middle of July 1962, the construction contract had been terminated, and only a minor amount of wiring and piping remained to be done. Pre-operational testing of installed equipment is more than half completed. Fabrication of fuel elements for the boiler region of the reactor is less than 20% complete, and fabrication of superheater fuel elements has not yet begun. Training of the operating personnel is in progress. The safeguards report has been submitted to USAEC and the application for an operating licence is under consideration.

86. The project has been delayed six months or more owing to problems in the fabrication of fuel elements for both boiler and superheater. The design of the boiler fuel elements calls for UO_2 pellets in Zircaloy-2 tubes. Four tube segments, each about 18.5" in length, are joined together by screwed joints to make a completed fuel rod, and 81 such rods constitute one fuel assembly. The difficulty has been in getting true alignment of the end caps which are welded on each end of a rod segment. The end caps and screw fitting are an integral piece, and if an end cap is slightly cocked when it is being welded, the two segments will not be in axial alignment. Proper alignment of the rods is necessary to maintain proper distribution of the coolant in the channels between fuel rods to prevent hot spots and possible damage to the fuel rods.

87. The superheater fuel consists of two concentric fuel tubes of highly enriched UO_2-stainless steel cermet, clad in stainless steel. A burnable poison rod is located on the central axis, within the smaller fuel tube. The assembly of poison rod plus two fuel-bearing tubes are maintained at the proper spacing by full length spacer wires and by a lower support fitting. An extensive development programme for superheater fuel elements is now nearing completion. The actual production of the superheater fuel is expected to begin towards the end of this year. It is expected that fuel fabrication will be completed by early 1963. The schedule for reactor completion and start up is as follows:

Table 6

The Pathfinder power reactor: Time schedule for the project

Item	Expected
Partial delivery of fuel elements	November 1962
Initial criticality, critical experiments to measure reactivity and control rod worth	December 1962
Completion of fuel fabrication	March 1963
Operation at low power (up to 5 MWe)	March 1963
Full power operation	July 1963
Routine operation	September 1963

Important design features

88. Objectives. The primary purpose of this plant is to demonstrate the feasibility and economics of integral nuclear superheat in a central power station. It has been preceded by an extensive research and development programme. A distinguishing feature of this reactor is that controlled re-circulation of the coolant is to be used to regulate the power level

[5] Information on the Pathfinder power reactor project was given in paragraphs 289 to 342 of GC(V)/INF/41, which is supplemented and brought up to date by that given here.

between 75% and 100% of full power. Butterfly valves regulate the rate of flow of water in the re-circulation loop; by this means the power level of the reactor can be adjusted to any level in the upper 25% of the operating range with no movement of control rods.

89. Core. The core is 6 ft in diameter and 6 ft in height. The central superheater region is irregular in cross section, having a mean outside diameter of about 30". The boiler section has 96 fuel assemblies of low enrichment (either 2.2% or 3.2% U^{235}) and 16 cruciform boron stainless steel control elements. The superheater has 412 highly enriched fuel elements and 52 control rods. Although the reactor is designed for 825 $^{\circ}$F steam production, it will initially operate at 725 $^{\circ}$F, and under this condition, the power density in the boiler is 45 kW/l of core volume and in the superheater 34 kW/l of core volume. The 99.9% dry steam enters the superheater at 489 $^{\circ}$F and leaves at 725 $^{\circ}$F. The steam conditions at the turbine inlet valves for initial operation of the first core are 525 psig and 722 $^{\circ}$F, with an estimated flow of 616 000 lb/hr to produce 62.5 MWe (gross) or 58.5 MWe (net).

90. Pressure vessel. The cylindrical pressure vessel, made of carbon steel and clad inside with stainless steel, was constructed with a design pressure of 700 psig at 500 $^{\circ}$F, and the vessel conforms with the ASME Pressure Vessel Code. This reactor pressure vessel has an inside diameter of 11 ft with an over-all height of about 36 ft. There are 32 penetrations through the vessel, 22 of these being in the vessel head. The vessel shell and lower head sections are fabricated from ASTM A 212 grade B carbon steel plate, 2 3/4" thick, clad on the inside with 1/4" thick stainless steel, Type 304 L. The hemispherical portion of the upper head is formed from 2 3/8" A 212 grade B plate with 1/8" of stainless steel cladding. The cladding is integrally bonded through an interface layer of nickel to the carbon steel by hot rolling of the composite plate.

91. Shielding. Top shielding is provided by an 18 ft deep pool of water above the reactor with its bottom insulated from the reactor cover. Radiation dose rates were calculated using an IBM computer programme, then the results were multiplied by three to allow for uncertainties. The estimated dose rates, including the uncertainty allowance, show less than 1 mr/hr on the operating floor of the reactor room, 10 to 30 mr/hr at the turbine, and 500 to 30 000 mr/hr in the area around the circulation pumps.

92. Containment shell. The containment vessel, which is also the reactor building, is a steel cylindrical structure 50 ft in diameter and about 120 ft tall. About half of the building is below grade. The building, constructed of preformed steel plates welded together to form a gas-tight enclosure, is designed to withstand an internal pressure of 78 psig. Because the containment building conforms to the ASME code for unfired pressure vessels, which specifies a safety factor of 4, it is expected that an overpressure of twice the design pressure would not cause the vessel to fail. The cylindrical wall section and the ellipsoidal bottom of the vessel are 1 3/8" thick, and the hemispherical top is 11/16" thick. Access to the building is furnished by two personnel air locks and one equipment door. The building was tested by pneumatic pressure at 97½ psig to demonstrate structural integrity, and a leak test at 78 psig showed that the leakage from the vessel and its penetrations did not exceed 0.2% of the initially contained weight of air in 24 hours.

93. Pumps and valves. The principal piping of the plant consists of three water re-circulation loops, the steam line from reactor to turbine, the condensate and feed water return lines, and the condenser cooling water supply. Each of the three re-circulation loops is a closed piping system, taking water from the reactor and returning it to the reactor. Each loop contains a 21 600 gallon per minute pump between two butterfly valves. The re-circulation pumps, made of stainless steel are vertical mixed-flow type, with 22" diameter suction and discharge nozzles, and the pumps are capable of developing a head of 71 ft of water. A floating, bushing-type shaft seal, into which water is injected by a positive displacement pump, prevents leakage of the reactor coolant to the atmosphere. Each pump is driven by a 400 HP, 2300 volt, 3 phase 600 rpm induction motor (constant speed).

94. The stainless steel butterfly valves are designed for a pressure of 700 psig and a maximum differential pressure across the disc of 50 psig. The valve on the discharge side of each pump controls the flow rate, and these valves are controlled so that the rate of change of total flow will not exceed 455 gpm per second. The reactor control rods are interlocked with the pump discharge valves to prevent movement of the discharge valves while a control rod is being moved.

95. In the main steam line (16" diameter) there is a motor operated isolation valve. The steam line divides into two 12" lines at the turbine, with a turbine trip valve in each line. Four safety valves are provided between the reactor and the main steam line isolation valve. Following the turbine and condenser are two condensate pumps and two feedwater pumps. In the condenser cooling water system there are two large circulating pumps connected to suction and discharge piping 42" in diameter.

96. Control rod drives. The control rod drive assemblies are individually mounted on the cover of the reactor vessel and submerged in the shield pool. The rods are suspended from a rack and pinion drive, which is coupled by gears, drive shaft and magnetic clutch to a three-phase reversible induction motor. The rod and rack assembly moves inside a housing tube which forms a seal with the reactor mounting flange, allowing steam from the reactor to condense inside the drive mechanism, which operates flooded with water. In the event of a scram, the magnetic clutch releases the rack and control rod from the drive motor, and the rod drops by free fall, assisted by a scram spring. The 164 pound boiler rod will drop the full stroke of 73 inches, including 6 inches in a dashpot at the end of travel, in less than two seconds.

97. Reactor start-up. Since the start-up of a superheat reactor poses special problems, it may be useful to describe the procedure which has been established for the Pathfinder reactor. Beginning with a cold reactor, the procedure is briefly as follows:

(a) All control rods are in, and the re-circulation pumps are stopped;

(b) The reactor, including the superheater region, is completely flooded;

(c) Pressurize the reactor with nitrogen to 300 psig;

(d) Start the three re-circulation pumps. Water flows through boiler and superheater fuel zones;

(e) Bring reactor to criticality, then increase power to approximately 10 MWth by moving outer boiler control rods;

(f) Permit reactor temperature to increase at the rate of 150°F - 200°F per hour. Increase nitrogen pressure to prevent formation of steam in the superheater. (Maximum set pressure is 550 psig);

(g) When reactor water temperature reaches 420°F, shut down reactor by inserting all control rods;

(h) Drain the superheater zone of the reactor;

(i) Establish a 6000 lb/hr flow of steam from the reactor to the condenser;

(j) Bring reactor to criticality and raise power to about 2 MWth by withdrawing boiler control rods. Superheater control rods remain in;

(k) Establish feedwater flow of 6000 lb/hr;

(l) Increase steam flow to 15 000 lb/hr. Adjust feedwater flow to 15 000 lb/hr;

(m) Increase reactor power to 5 MWth;

(n) Increase steam flow to 60 000 lb/hr, feedwater flow to 30 000 lb/hr;

(o) Increase reactor power to 20 MWth by withdrawing boiler control rods;

(p) Increase reactor temperature gradually by operating reactor between 30 MWth and 40 MWth until steam leaving the reactor reaches a temperature of 475°F;

(q) Take turbine through warm-up procedure, bring turbine to speed, synchronize unit and bring it in line with the transmission system of the Northern States Power Company; and

(r) Adjust reactor power to a minimum of 40 MWth. Withdraw the superheater control rods. Adjust boiler control rods for power operation. The superheater exit steam temperature will increase to approximately 725 $^{\circ}$F after superheater control rods have been completely withdrawn. Normally the superheater control rods will remain fully withdrawn during power operation. To increase power, movement of the boiler control rods will increase the superheater power as well as increasing the boiler power. The power split between boiler and superheater can be maintained almost constant.

98. Load following. The Pathfinder plant is designed as a base load plant. Power control is exercised by manual operation, with no provision for automatic load following. Hence, the plant should be operated at base load as far as possible.

99. Steam separators and dryers. The water-steam mixture which rises through the boiler fuel region reaches the free liquid surface in the reactor vessel, at which point about 80% of the steam is freed by natural separation. The re-circulating water turns downward, carrying 20% of the steam, into 44 centrifugal separators, each 10" in diameter and 102" long. These separators handle about 65 000 gallons of water per minute and 2000 pounds of steam per minute. Steam carry-under from the separators to the re-circulation pump is about 0.1% by volume. Steam from the separators passes upward through 4" pipes into the steam dome area. Total maximum pressure drop through the centrifugal separators is 5.5 ft of water at normal operating conditions.

100. The steam generated in the boiler region is dried by passing through an 8" thick inconel wire mesh assembly before reaching the superheater. This steam dryer assembly extends from the superheater chimney to the inside diameter of the neck of the reactor pressure vessel. The moisture content of the steam leaving the dryer is expected to be less than 0.1% at design steam flow rate.

Fuel

101. Boiler fuel. The design of the boiler fuel elements has been described previously.[6] In order to allow flexibility for the first core loading, 32 boiler fuel elements will be fabricated with 3.2% U^{235} enrichment, in addition to a full loading of 96 elements with an enrichment of 2.2% U^{235}. Additional reactivity will be available, if needed, by substituting boiler fuel of the higher enrichment.

102. Superheater fuel. The development work on superheater fuel design and fabrication required more time than had been estimated. The cermet fuel strip is formed by hot rolling of the sintered UO_2-stainless steel powder encapsulated in stainless steel foil. The strip is formed into a tube, which is then clad inside and outside with seamless stainless steel tubes by draw bonding. Spacer wires are welded to the outside of each fuel tube and the central poison rod. The three pieces - poison rod, inner fuel tube and outer fuel tube - nest together with no clearance. This is accomplished by inserting the poison rod into the inner fuel tube while the latter is elastically deformed. The same procedure is followed to insert the above pair into the outer fuel tube.

103. A problem of particular concern for steam-cooled stainless steel clad superheater fuel elements is the possibility of chloride-stress corrosion cracking. Should a cladding failure occur, however, release of fission products should be retarded due to the fact that UO_2 fuel particles are imbedded in the stainless steel matrix of cermet. Allis-Chalmers is investigating chloride-stress corrosion as a part of their fuel development programme.[7]

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[6] GC(V)/INF/41, paragraphs 319 to 321.

[7] For a general discussion of this problem see paragraphs 65 to 71 above.

104. Fuel handling. Refuelling can begin four hours after reactor shut-down. Four boiler fuel elements at a time (or up to 64 superheater elements) are placed in a transfer box, which is moved on a motor driven carriage to the spent fuel storage pool. It is estimated that one fuel element can be removed every 10 to 15 minutes. The storage capacity of the spent fuel pool is sufficient for more than two complete reactor loadings. One third of the boiler fuel elements are to be replaced at each refuelling. The entire superheater fuel load is to be replaced every third refuelling. If it becomes necessary to remove a control rod, the four adjacent fuel rods will first be removed to maintain a safe margin of reactivity control. It is difficult to ascertain the exact amounts of various wastes to be handled. Therefore, the aim is to design a flexible system to meet varying needs.

105. Waste disposal. Facilities are provided for monitoring and handling solid, liquid, and gaseous wastes. Radioactive solid wastes will be packaged and shipped off-site for disposal. Liquid wastes will be processed, if necessary, and disposed of either by discharge to the Big Sioux River, if the activity is within permissible limits, or by shipping to an off-site disposal area. Radioactive gases produced in normal operation will, after a 15-minute delay in the off-gas system, be passed through an absolute filter, diluted with ventilation air and discharged through a stack to the atmosphere. In the event of a major release of gaseous activity, all of the gas may be retained and compressed into two hold-up tanks which have sufficient capacity for approximately 12 hours of operation at full power.

Analysis of maximum credible accident

106. Although there is no reason to suspect that such an accident will occur, the maximum credible accident is postulated to be a complete severance of a reactor coolant re-circulation line. The water and flashing steam from such a break is further assumed to be released directly and instantaneously to the free-volume of the reactor building. The maximum pressure in the building under these conditions is 78 psig, equal to the rated building design pressure. This would be the case if the rupture occurred during start-up, when the entire reactor is filled with water at 442°F and 390 psig. If the break occurred while the reactor was operating normally at full power, the pressure in the building would be about 68 psig. Following a complete loss of coolant and consequent rapid reduction in pressure, the fuel cladding could be expected to fail and the core would gradually melt. The release of fission products to the reactor building is assumed to be as follows:

	% of core inventory
Rare gases: Krypton and xenon	75
Volatiles: Bromine and iodine	25
Solids: Strontium	1

107. Assuming that an inversion obtained at the time of the accident and persisted for 24 hours, the cumulative radiation dose received by a person 800 meters from the reactor during the first 24 hours would be:

	rem
Equivalent whole body dose due to iodine inhalation	36
Bone dose due to inhalation	38
External gamma dose	0.14
Total equivalent whole body dose	74

Cost data

108. The construction costs are as follows:

Table 7

The Pathfinder power reactor: Construction costs

(in dollars)

Item	Cost	Cost/net kWe[b/]
Land and land rights	450 000	7.7
Structures and improvements	3 070 000	52.5
Reactor plant less fuel	7 600 000	130.0
Turbine generator	5 850 000	100.0
Accessory electrical equipment	600 000	10.2
Miscellaneous plant equipment	300 000	5.1
Outdoor switchyard	700 000	11.9
Additional indirect construction costs	3 930 000	67.2
TOTAL	22 500 000[a/]	384.6

a/ Includes $3 650 000 research and development funded by Central Utilities Atomic Power Associates.

b/ Based on initial plant output of 58.5 MWe (net). If based on planned future output of 62 MWe (net), unit cost would be $363/kWe.

IV. THE PIQUA NUCLEAR POWER FACILITY [8]

General

109. The Piqua nuclear power facility was in the final stages of pre-operational testing in August 1962. The criticality of this plant has been delayed by about 11 months and is now set for October 1962. This delay has been caused, among other things, by the relative inexperience of the construction contractor, unforeseen difficulties in system clean-up and troubles in the non-nuclear part of the system which came to light when the pre-operational tests were initiated. These include leaks in the steam tracing system, valve and pump problems and leaks in the spent fuel storage pool, and inadequacy of the ventilating system. These are discussed in detail in paragraphs 111 to 131 of this report.

110. The latest schedule is given in the table below.

Table 8

The Piqua nuclear power facility: Time schedule for the project

Item	Previous	Latest
Pre-operational testing	September 1961	September 1962
Initial criticality	November 1961	October 1962
Full power operation	February 1962	Early 1963

Experience gained in design, construction and pre-operational testing

111. Overcrowding in auxiliary building. The Piqua reactor design is basically simple. The general design is similar to that of a water reactor but with some of the features of a chemical plant. The important difference lies in the fact that the organic coolant must be kept at 300°F which requires the installation of a steam-tracing system with associated valves and traps. Also needed is the purification system with its associated equipment, much of which has to be kept within the reactor building. This tends to lead to overcrowding. It is now felt that more room should have been left at the Piqua reactor to make system parts such as valves, pumps and instruments more readily accessible for routine check-up and maintenance during operation.

112. Control rods. The Piqua reactor has 13 magnetic jack-type control rods which fit inside the fuel elements. The drive mechanism is submerged in the coolant above the core. During their assembly and shock testing, it was observed that the insulation of the hold and lift coils had some cracks. The quality control procedures were revised and made stricter so that coils which met the highest standards of insulation only were accepted. Some of the coil cans which are to be immersed in the organic developed leaks and had to be replaced. The top connectors use glass insulation and they had some troubles too. The insulation had to be replaced by a more rugged material. Seven control rod assemblies duly tested in the shops were at the site in June 1962.

113. Fuel. The fabrication of U-Mo-Al alloy fuel elements was completed according to specifications. No special problems were encountered in making the fuel elements. A high standard of quality control was maintained throughout their fabrication and assembly.

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[8] Information on the Piqua nuclear power facility was given in paragraphs 144 to 237 of GC(V)/INF/41, which is supplemented and brought up to date by that given here.

114. The problem with fuel lay in its delivery to the site. The elements are about 80" long and consist of four cylindrical tubes (two Al-clad fuel and two steel) which fit into each other. The outer and inner surfaces of the fuel tubes are finned for better heat transfer. The delicate nature of the Al-tubes demands great care during handling and shipment. The Piqua fuel was fabricated at Canoga Park. In order to accurately gauge the effect of shocks which could be encountered during the shipment, a trial run was made in a truck with fuel elements filled with solid organic. The shocks from rapid acceleration and deceleration during the 2000 mile trip were recorded. The results were very satisfactory and no damage to the fuel was observed. The actual shipment of these elements will be made in specially equipped trucks which will further minimize the effect of shocks. The first batch of fuel arrived at the site in May 1962 and was in excellent condition.

115. Biological shield. During pre-operational testing the biological shield, which consists of ordinary concrete, developed hairline cracks which appeared after the hot (190°F) citric acid solution was circulated through the shield cooling system for clean-up purposes. The core was at 45°F. The temperature difference, together with the rate of temperature rise, caused thermal stresses in the concrete leading to cracks. The cleaning process required that the citric acid solution be raised to 190°F at the outlet of the shielding. Since the shield provided a large heat sink, the cleaning solution had to be pumped rather rapidly. It was expected that hairline cracks would be hard to avoid in a concrete mass of this size with expected temperature cycles. An analysis of the cracks has led to the conclusion that the shielding structural integrity has not been affected adversely and no radiation streaming is expected as a result of this. A thorough radiation survey will be conducted prior to start-up with the source in the reactor to evaluate fully any possible radiation streaming effect. Normally, the concrete will be hotter due to heating from radiation, and the water through the cooling pipes will be at 100°F (instead of 190°F) which will further narrow the temperature gradient. There will be no significant thermal stresses in the biological shield to worry about.

116. Steam tracing. The Piqua reactor employs steam tracing to pre-heat the main vessel, pipes and tanks to keep the organic Santowax-R, which is a solid at room temperature, in a liquid state around 300°F. During the pre-operational testing the steam-tracing system developed numerous leaks and three to four weeks were lost in carrying out necessary changes and repairs. Although this system is purely conventional and circulates 175 lbs steam at 320°F, it has nearly 400 valves, 600 steam traps, and miles of thin-walled small diameter carbon steel piping. When steam was passed through, it developed leaks at several places which broke the insulation around. These leaks occurred mainly at tube junctions (swedge lock fittings and brazed coupling joints). On checking, it was found that the leaks had appeared at many flared fitting joints which, contrary to specification, had been buried under insulation. Some of the leaks were the result of poor workmanship by the construction craftsmen. In certain cases the moisture left in the insulation had corroded the thin-walled pipes, causing many holes to appear.

117. Although all detectable leaks have been repaired some new ones spring up now and then. It is felt that the design of the steam-tracing system could be improved and for future work of this type it is recommended that:

(a) The maximum length of thin-walled pipes should be used with minimum of breaks between header take-off and steam traps;

(b) The number of flared fittings and connections should be reduced and welding used wherever possible;

(c) All the connections should be brought outside the insulation for easy repairs;

(d) All the work should be inspected closely while in progress; and

(e) The bucket type of steam traps should be replaced by the simpler impulse types.

118. System clean-up. The cleaning of the Piqua system prior to start-up was found to be a much bigger problem than originally estimated. The approach in this case had to be quite different from that used in common utility practice. The carbon steel system had

first to be cleaned and then passivated to prevent re-rusting. After experimentation, the Atomics International was able to develop a procedure which has shown excellent results.

119. The primary and secondary piping as well as the vessel of the Piqua reactor is made of carbon steel. Under clean conditions the organic coolant does not react with carbon steel and there should be little corrosion problem. During construction the system becomes contaminated and slightly corroded necessitating cleaning before operation, otherwise impurities in the system will cause corrosion which contaminates the coolant. This in turn can lead to fouling of the fuel elements. The important steps taken in system clean-up are:

(a) Pre-cleaning of the system with hot water and steam; this removes loose material, construction debris, rust, sand, etc. Temporary employ of screens in the system to catch wood and cloth pieces, insulation material, sand, etc.;

(b) Opening of the main pipes and valves, pulling out of screens and cleaning by hand;

(c) De-greasing by using a hot and mild solution of caustic soda and detergent at 180^{o}F;

(d) Cleaning with 3 - 5% citric acid solution. Adding a corrosion inhibitor (for example Armohib-31) 0.03% solution. Keeping the temperature between 160^{o} to 200^{o}F maximum; controlling the pH at 3 by adding ammonia. Letting it circulate for several hours (6 to 24 hours) to take all the rust and slag off;

(e) Passivating the system, by increasing the pH from 3 to 9 by adding NH_3. Then add sodium nitrite (0.5% by volume) solution. Proper control of pH and temperature is absolutely necessary. Introduction of air bubbles during passivation is also very helpful. It takes several hours to passivate the system;

(f) Flushing by using first cooling water at a controlled pH of 9 to 10 followed by a final rinse using condensate at the same pH. Drain the system and dry it by using clean dry air or nitrogen which will pick up all the moisture. As the final step in drying, steam tracing may also be turned on.

120. During the initial cleaning-up runs, four attempts were undertaken before the proper method was established. These cleaning runs were conducted throughout the auxiliary system. As a result of these four cleaning cycles, a number of valves in the system began to leak due to excessive corrosion. A special method was developed to permit repair of the weld-in valves and 68 of such valves were repaired using this method. In the case of two others, the seats had to be replaced.

121. The lesson to be learned from these is that during clean-up one has to be very careful about the solution used to make sure that it is fully compatible with the alloys in the system.

122. Valves. Most of the valve problems resulted from corrosion in initial cleaning cycles, and these have been overcome. There are, however, cases where the valves do not shut off properly, especially the dual purpose ones which are also meant to control the flow. These will be repaired before start-up.

123. Fuel element storage pool. During pre-operational testing it was found that there were several cracks in the pool concrete through which water was leaking. These cracks were caused by high temperature gradient at the construction joints which resulted from an excessive ambient temperature in the building. This problem is not serious and will be resolved by painting the inside of the pool with a suitable Epoxy resin lining.

124. Heating and ventilating systems. It was found that the cooling capacity of the ventilating system was not adequate and the ambient temperature in the building rose to 150^{o}F during the tests. This necessitated a halt of the tests until additional cooling capacity was

installed. It appears that the heat losses from various parts of the system were underestimated. The removal of insulation from certain areas for adjustment during tests also contributed to the higher temperature inside the building.

125. Cooling water purity. During test runs it was observed that certain water-cooled pump-bearings became overheated because of excessive build-up of crud and scale inside. The local cooling water had a too high content of impurities (such as calcium) which deposited around the bearings. A plan to use a closed-cycle system of demineralized water employing a cooling tower is under consideration for certain parts of the system.

126. Fuel handling machine. The design of the fuel machine is simple and rugged. It has operated well during tests but it required some rework and adjustment to improve its performance. It is expected to change one fuel element in about ten minutes after the operation has started.

127. Fouling of fuel elements. One of the major problems in organic reactor systems is the possible fouling of fuel elements surfaces. Fouling is like scaling and results in the formation of a thin film on the fuel element cladding which affects heat transfer. The phenomenon of fouling is not completely understood. Extensive research is going on to determine the exact cause of and remedy against this occurrence. It appears that insoluble particles in the coolant form the nucleation sites for the high boiler compounds which coat these particles and become bigger (about 5 micron) in size. Then they tend to selectively plate out on the hot fuel element surfaces. A small molecular deposition can be tolerated but particle build-up increases deposition rate to form a thick (about 150 micron size) film which acts as a heat barrier. It is postulated that this build-up can start even without inorganic nuclei (such as iron carbide) and extremely big molecules of high boilers too can initiate this phenomenon.

128. Several possible remedies are under investigation. It has been found that the precoat filters are not entirely satisfactory. The full flow filters tend to plug up. In the case of fibre glass filters, the pressure drop is small but they are effective against about 40 micron size particles. To catch the smaller ones it is necessary to resort to pre-coat filters or a still which is costly. All in all, this is an engineering economics problem and a combination of filters, evaporator and still may provide the answer.

129. Coming to the Piqua system, it may be pointed out that the fuel elements used are the same as those in core 2 of OMRE, which did not show any appreciable fouling problem and are in a sense of proven type. Nevertheless, to keep the Piqua coolant free from undesirable particles, several provisions have been made. A 25 to 5 micron size strainer has been provided in the pressurization system. Full flow mainline filter will be installed to remove particles down to about 5 micron size without incurring a pressure drop of more than 10 psi. Sixty-nine of such filters are to be located in the reactor vessel in the plenum above the core. In addition, the purification system can be operated at maximum capacity to reduce the high boiler fraction and further cut down the coolant impurities. A side stream particulate removal loop utilizing pre-coat diatomaceous earth filters is capable of removing particles of less than 1 micron size.

130. Coolant recovery from residue. The residue removed by the purification system consists of high molecular weight hydrocarbons called high boilers and amounts to about 50 lb/hr based upon full power operation of the Piqua reactor at 30% high boiler content. This residue may be too valuable to be discarded. Several methods have been considered for the reclamation of the usable low molecular weight hydrocarbons from the residue. Heat distillation is workable but it appears to be too expensive. Decomposition of high boilers into low boilers is possible. Studies show that catalytic hydrocarbon cracking could recover the coolant at the cost of about 5 ¢/lb. Fractional distillation may also be employed for separating the high boiler.

131. Until a suitable economic method for reclamation of usable coolant from the residue has been developed, it is planned to dispose of the Piqua high boiler residue by burning.

Future outlook

132. Except for the difficulties discussed above, the other parts of the Piqua system have checked out very well. Even though some time has been lost in devising solutions for the problems which have been encountered, the experience gained in the process has been most valuable. It will be applied in the design and construction of larger organic plants. It must be remembered that the Piqua reactor is basically an experimental station because it is the first full-scale organic power plant and, as such, certain difficulties could not be avoided. It is now getting ready for start-up and its operation will be watched with great interest and is expected to yield much needed information for the design of larger plants.

133. EGCR has already benefited from the experience of Piqua. The next organic reactor under consideration is a 160 MWth, dual purpose OPHIR which will produce 70 MWth of process heat and 27 MWe of power for a paper factory. This will be a prototype for a large (300 MWe) organic power plant under planning for the late 1960s.

V. THE HALLAM NUCLEAR POWER FACILITY

General

134. The Hallam nuclear power facility (HNPF), the first sodium graphite reactor (SGR) to be used for the production of power, is located near Hallam, south of Lincoln, Nebraska. The reactor uses slightly enriched (3.6%) uranium, is moderated and reflected by graphite and is cooled by sodium. The thermal power is 240 MW and the net electrical output is 75 MW.

135. The plant will operate with higher reactor coolant and fuel element surface temperatures, and higher steam pressure and temperature than any other reactor system currently authorized in the USAEC Power Demonstration Reactor Program. The purpose of HNPF is to extend the knowledge about sodium graphite systems already gained from the sodium reactor experiment (SRE) and demonstrate the technical and economic feasibility of this concept for central station application with high availability factor in a small utility system.

136. The nuclear portion of the plant, including the steam generators, is owned by USAEC and the remainder of the plant is owned by CPPD of Nebraska. The entire station will be operated by CPPD which has the option to purchase the reactor itself at a later date based on its future value as a power producer.

137. USAEC and CPPD signed a contract in September 1957 and the cost estimate and preliminary design were submitted to USAEC in September 1958. Final engineering was initiated in October 1958 and virtually completed in June 1960. Site construction started in April 1959 and was essentially finished in October 1961.

Important design features

138. A summary of important data concerning the Hallam reactor is set out in Annex I and a flow diagram of the reactor system is shown in Figure 1.

139. Objectives. The design, construction, safety features and operating procedures of HNPF are based on experience obtained from SRE, which was designed and constructed by Atomics International and has been operated by them for USAEC since July 1957. The over-all design of HNPF is very conservative; the present core size with modifications could produce about 800 MWth.

140. The advantages claimed for SGR systems result from the fact that sodium is a liquid over a wide temperature range (208°F to 1630°F). Because of this, it is possible to produce the steam conditions necessary for running efficient turbines of the latest design without resorting to fossil-fuelled superheaters. High-temperature operation also leads to better thermal efficiencies which serve to reduce operating costs. As a liquid, sodium has a low vapour pressure which minimizes the need for pressurization. This means that a thick pressure vessel and high-pressure seals are not necessary, that problems of pump cavitation are reduced, and that more varied and complete core instrumentation is possible. Furthermore, sodium is chemically compatible with standard core materials such as graphite, uranium, stainless steel, and zirconium. Sodium has a high heat capacity and along with the high boiling point tends to limit any accident to one which is easily contained within the core vessel and piping galleries. Because of the high specific heat of sodium, low coolant flow-rates are possible which leads to reduced pumping power. Moreover, sodium graphite reactors are inherently stable due to the negative prompt temperature coefficient of the fuel.

141. On the other hand, the SGR system has its drawbacks. Sodium is highly reactive with water and to a lesser extent with air, thus requiring extra safety measures such as inert atmospheres and double piping in critical areas. In HNPF double piping is provided from the reactor to the blocking valves to preclude the possibility of draining the

core if a pipe should fail within the reactor cavity. The steam generators are the only place where double piping is provided for actual separation purposes. The induced radioactivity of Na^{24} makes the primary heat transfer system inaccessible for immediate maintenance. Moreover, graphite tends to absorb sodium and become weakened; therefore, moderator and reflector must be clad. Also, high thermal stresses may be encountered because of the quick thermal response of sodium to power changes. The above problems, although restrictive, present no insurmountable engineering barriers and can be overcome by careful attention to system and component design as was successfully demonstrated by the construction and operation of SRE.

142. Core. The cylindrical reactor core is 13 ft in diameter and 13.25 ft high. There are 205 process channels out of which 137 are occupied by fuel elements, 19 control rods, 45 dummy elements, and four locations for the neutron source and core temperature instrumentation. The hexagonal moderator blocks, canned in Type 304 stainless steel and completely submerged in sodium, are scalloped at each corner and when stacked together, three such corners form a process channel for fuel and control rod thimbles. The reflector blocks are identical to the moderator blocks with the circular channels being filled with canned graphite logs.

143. The sodium enters the bottom of the reactor and flows upward around the fuel elements and to the sodium pool above the core and then leaves the reactor. Bypass flow between the moderator blocks provides moderator cooling. The total primary flow rate is 8.4×10^6 lb/hr and the heat output of the core is 240 MW with an average power density of 4.8 kW/l.

144. Reactor vessel. The reactor vessel is essentially an open-top flat-bottom vessel with a 19 ft outside diameter and a 33 ft inside height and is attached to the upper cavity liner by a stainless steel bellows seal. The vessel itself is made of Type 304 stainless steel and varies in thickness from 0.75" to 2" with penetrations for vent lines, and sodium inlets and outlets. The working pressure of the inlet plenum is 20 psig. Between the reflector and reactor vessel is a stainless steel thermal shock liner and a stagnant layer of sodium to preclude the possibility of setting up undue thermal stresses in the reactor vessel. The upper closure is provided by the loading face shield which is described further on.

145. Containment. Because of the low operating pressures and the inherent safety of the system, HNPF is the only nuclear power plant under construction in the United States which is housed in a conventional industrial building. This building has been designed to maintain a negative pressure (relative to barometric) in those areas subject to potential radiation hazard to ensure that leakage of air is inward.

146. The containment of the primary systems is provided by carbon-steel-lined cells which seal the system and maintain an inert gas atmosphere. In the event of a major sodium leak in the reactor vessel or piping internal to the reactor cavity, the outer vessel and associated guard piping will maintain a sodium level sufficient to assure natural circulation in the core. The outer vessel is made of low alloy carbon steels and varies in thickness from 0.5" to 2". Directly beneath the bottom of the outer vessel, permanently mounted heater elements are provided to maintain reactor temperature during prolonged shut-downs and to aid in the pre-heating of the reactor structure.

147. Separation from the atmosphere is provided by a 0.5" to 1" thick carbon steel cavity liner which provides a gas-tight envelope between the reactor cavity (approximately 46 ft deep with an inside diameter of 24 ft) and the concrete biological shielding. This envelope (estimated leakage rate 0.01% of total volume per day) is needed to prevent moisture from entering the reactor cavity as well as to enclose the dry helium atmosphere surrounding the reactor structure and to contain sodium vapours in case of a leak from the reactor vessel or primary piping. Cooling is provided by water circulating through 1" pipes on the exterior surface of the liner to remove heat generated in the concrete and the cavity liner itself.

148. Shielding. Radial shielding consists of the 0.25" stainless steel thermal shock liner, 0.75" stainless steel reactor vessel, two 2.75" carbon steel thermal shields located between the reactor and containment vessels, 0.5" low alloy steel outer-vessel, 12" of superex thermal insulation, 0.5" carbon steel cavity liner, and 5 ft of heavy concrete as the biological shield. Bottom shielding consists of the 2" reactor vessel, 2" containment vessel, 1" cavity liner, and 5 ft of heavy concrete.

149. The shielding above the reactor is in the form of a 300-ton rotatable loading face shield which is a circular, stepped plug of reinforced heavy concrete encased in ferritic stainless steel. The shield is approximately 19 ft in diameter and 7 ft in height. To provide proper atmosphere containment this plug is sealed to the upper cavity liner by a frozen metal alloy seal. The composition of this shield from the bottom upwards is 13 equally spaced polished stainless steel plates which serve as reflective insulation (total space $10\frac{1}{2}$"), 1" steel plate, 1.5" lead, 6 ft heavy concrete, and 1" top steel plate. Cooling is provided by nitrogen circulating in pipes imbedded in the lead. The shield is penetrated by vertical, stepped openings for fuel elements, control rods, and in-core instrumentation. In addition there are three large circular openings through which moderator and reflector elements can be removed or maintenance operations performed.

150. Fuel handling and storage. The fuel and component handling system provides for receipt, handling, cleaning, decontamination, storage and shipment of fuel elements, control rods, moderator and reflector elements, and other replaceable core components required during start-up or normal operation of the reactor. After shut-down the control rod drive mechanisms can be disconnected from the control rods at the upper surface of the loading face shield and moved away, leaving the shield face clear for fuel handling operations. Refuelling is done through removable plugs in the loading face shield with one plug for each fuel element. The charge-discharge machine consists of a lead shielded cylinder with a gas lock, indexing device, and a dual hoist and grapple mechanism to raise and lower components. During normal operation, there is about a 10-hour waiting period to account for decay heat. After that it takes about 30 minutes to remove a fuel element from the core to storage during which there is no cooling other than the circulating helium system in the charge-discharge machine.

151. Present objectives are to have the fuel change period coincide with the periodic plant maintenance and inspection schedule. After the initial testing and full power operation, it is calculated that the steady state refuelling cycle will be seven months of full power operation with a week of refuelling downtime. Ultimately, however, the fuel cycle period and the number of fuel elements to be replaced will be dictated by actual experience concerning the maximum permissible burn-up of the fuel and the plant load factor.

152. After removal from the core, the fuel elements are transported to storage. The three storage vaults contain a total of 383 thimbles (272 for fuel elements) which are 41 ft long. Vault 1 is dry and vault 2 is normally dry but can be flooded with water. Vault 3 is normally filled with water. Fuel is stored only in vaults 2 and 3. Before shipment to another facility for recovery of uranium and plutonium, the fuel elements are stored for about 100 days to permit sufficient decay of fission products to facilitate chemical reprocessing and to reduce shielding requirements.

153. Important items connected with the fuel handling system are also: the pick-up cell used only during the loading of new fuel; the maintenance cell used for fuel cluster exchange and inspection of core components; a shielded fuel shipping cask; and a portable purge unit used to monitor for fission products and to purge the storage thimbles.

154. Waste disposal system. Radioactive liquids come mainly from the fuel wash (cleaning) cell, the decontamination room, the maintenance cell, and the laundry, and have widely varying degrees of activity. Two 5000-gallon storage tanks are provided to hold liquids where the activity has a short half-life (less than 30 days). A 1000-gallon tank is provided for those radioactive liquids with longer decay period requirements and

to contain highly radioactive liquids until they can be packaged for disposal. The storage capacity of the liquid waste system is estimated to be sufficient for a five-year accumulation of liquid wastes. In addition to the above facilities, a tank loading line is provided so that low-level liquid waste may be shipped by tank truck.

155. The major portion of the radioactive liquid waste is generated during the fuel cleaning process. If no fission products are involved, the wash water will contain only radioactive NaOH and will require only short-term storage before the activity level has dropped sufficiently to permit controlled disposal to the plant, ash pit or the leaching field. If a fuel rod has ruptured in the reactor core or during cleaning, fission products will be released to the wash water necessitating prolonged storage, processing and off-site shipment. To detect high activity levels, the wash water is sampled and tested for fission products.

156. The objective of the radioactive vent system is to collect and monitor all radioactive gases. If the activity is below the maximum permissible concentration, the gas is released directly to the atmosphere. If, on the other hand, the vent gas is too radioactive, it is compressed, stored in holdup tanks at 140 psi maximum and after an adequate decay period is discharged by controlled release to the stack.

157. Special auxiliary systems. The functions of the sodium service system are the filling and draining of the sodium heat transfer loops and the purification of the sodium. The system consists of a melt station for new sodium, the primary and secondary system fill and drain tanks, the electromagnetic service pumps, and the purification equipment which consists of two primary and one secondary system circulating type cold trap and plugging meter assemblies. The cold trap removes sodium oxides from the sodium by cooling to a point where the oxide becomes insoluble and the precipitate is filtered out. The plugging meter, consisting of an orifice, cooler and temperature and flow meter, also works on the precipitation principle. As the saturation temperature is reached, the oxide begins to plug the orifice and the flow is stopped. The oxide concentration is determined by comparing the temperature and flow values with a plot of saturation temperatures for sodium oxide in sodium.

158. An unusual feature of sodium-cooled reactors is the requirement of a heating system for all components in contact with sodium which is solid below 208°F. To keep the sodium liquid and to minimize thermal stresses, an electrical pre-heating system is used to maintain components at 350°F. This system requires over 1100 heater sections and control points with a total pre-heat load of approximately 500 kW.

159. The helium system maintains an inert gas atmosphere inside the reactor and in all piping and equipment in contact with sodium. Its principal functions are:

(a) To prevent oxygen to come into contact with sodium;

(b) To maintain constant helium pressure in the piping and equipment which it services;

(c) To provide for purging the system or part thereof; and

(d) To provide for pressure transferring or draining of sodium.

160. The primary functions of the nitrogen system are to maintain an inert atmosphere in the pipe and heat exchanger cells, the cold trap cells, the primary fill tank cell, and the primary service pump cell; moreover, nitrogen is used for cooling the top shield.

161. A water cooling system is located in the biological shield to remove the heat generated by radiation interactions as well as sensible heat.

Safety

162. Safety in design. The SGR concept has many inherent safety features. Because the reactor operates at nearly atmospheric pressure and because all materials used in close proximity are chemically compatible, there are no large amounts of potentially releasable energy within the reactor which would originate from pressurization or chemical reaction. Therefore, there is no need for a high-pressure vessel, and a building of conventional industrial design can be used to house the reactor.

163. Within the reactor building, areas of potential radiation contamination are maintained at a slight negative pressure. The reactor and its associated radioactive components are located in concrete shielded areas below ground level. The shielding reduces radiation levels in all working areas to below normally accepted working levels. An inert gas atmosphere is used in all areas containing radioactive primary system piping and components, thus preventing any possible reaction between radioactive sodium and air. Components of the reactor structure in contact with liquid sodium are made of Type 304 stainless steel for strength, and low creep rate at high temperatures. Although corrision resistance is not a prime factor in the selection of stainless steel, there remain unanswered questions on mass transfer and decarburization of carbon steel in sodium at high temperatures which make stainless steel a better choice for these systems. Conservatism in the structural analysis was achieved by maintaining low stress levels within the reactor structural components as well as by minimizing the number and degree of stress risers.

164. Because of the chemical incompatibility of sodium with water and air, and the induced radioactivity of Na^{24}, special consideration had to be given to the design of the heat transfer system. There are three independent circuits, each consisting of a radioactive primary loop with a sodium-sodium heat exchanger and a secondary non-radioactive loop connected to a sodium-water heat exchanger. This arrangement separates the radioactive primary sodium from the steam system. The intermediate heat exchanger is a shell (secondary) and tube (primary) type with counterflow design. The steam generator is also a shell (steam) and tube (sodium) type. The tubes are double-walled, the annulus being filled with helium at 300 psig which serves as a monitoring fluid for detecting leaks. In the event of a leak from helium to sodium, the pressure in the annulus falls to that of the sodium which is less than 100 psig. On the other hand, a leak of steam to the helium will cause a pressure rise to about 850 psig. Within the reactor cavity the sodium piping is also double-walled as mentioned before. The three steam generators discharge into a common header which leads to the turbogenerator.

165. The sodium instrumentation systems are designed to meet several basic criteria: to withstand temperatures up to $1200^{\circ}F$; to withstand high radiation levels; to allow only a minimum of contact between moving parts and the sodium; and to allow no thin or weak material sections. When either of these last two conditions cannot be met, double containment is provided.

166. Fire protection outside of the building is provided by water hydrants and hose cabinets. Throughout the building the vital electrical equipment areas are provided with carbon dioxide extinguishers. In non-sodium areas there are automatic sprinkler systems. In the sodium areas, the nitrogen and helium atmosphere offer fire protection which is supplemented by hand and wheel type dry chemical systems. In addition, dry bulk calcium carbonate is available in all sodium areas to be applied by shovel in case of a small spill.

167. Control. The control of the reactor is provided by 19 shim-safety-regulating rods which can be operated automatically by the plant control and protective systems or manually by the operator at the control console. The poison column is made of a gadolinium-samarium oxide poison, clad in hestaloy-X. The control rods operate in helium pressurized Zircaloy-2 thimbles suspended from the loading face shield. The

total calculated worth of all rods is $14\% \frac{\Delta k}{k}$. The rod speed is 12.4"/min and the estimated maximum reactivity addition rate is $0.03\% \frac{\Delta k}{k}$ per second for all 19 rods.

168. The scramming mechanism is of the magnetic clutch, gravity fall type. The design of the automatic control system provides automatic operation and load following between 15 and 100% of design power.

169. The system has the following temperature coefficients depending on fuel burn-up and for a clean, initially loaded core:

	$\frac{\Delta k}{k}/^{o}F$
Fuel	-1.56×10^{-5}
Moderator	$+1.33 \times 10^{-5}$
Sodium	$+0.56 \times 10^{-6}$

170. A strong negative prompt fuel temperature coefficient of reactivity provides inherent control over increases in power level. At operating power levels, any increase in power is counteracted by a prompt decrease in reactivity due to the fuel temperature coefficient.

171. Site. The facility is located in Lancaster County in southeastern Nebraska, about 19 miles south of Lincoln. The site is accessible by two roads and a spur line provides railroad transportation. The terrain at the site is characterized by the rolling hills and valleys typical of the Loess sections of the Great Plains region of the United States. Ground elevation is approximately 1400 ft above sea level. The lands adjacent to the site are sparsely settled, being farms and small communities.

172. The village of Hallam, population about 270, is 1.5 miles south of the site and is the closest concentration of population. The largest cities nearby are Lincoln, population 150 000, and Beatrice (19 miles south), population 12 000. The population distribution as a function of distance from the reactor is shown in Table 9.

Table 9

The Hallam nuclear power facility: Population distribution (1952)

Distance in miles	Population density/square mile
0.5	0
1.0	26
5.0	1 052
10.0	6 768

173. Detailed studies of climatology, meteorology, seismology, geology and hydrology have been made for the area. Ground water, obtained from wells, is the source of domestic supply for the Lincoln-Hallam area. Age determinations, based upon the tritium content in the water, indicate a poor communication between surface and sub-surface water flows.

174. In determining the maximum credible accident, many situations were taken into account. Certain events were eliminated as not even being credible. These include:

(a) Complete core melt-down. Because the system involved failsafe circuitry, the core has a large heat capacity, and there exist three independent primary loops, any one of which is sufficient to provide enough natural circulation to prevent a melt-down;

(b) Large radioactive sodium fire. Because of a lock and tag procedure during maintenance;

(c) Large radioactive sodium spills. Because of careful design and construction of the reactor system; and

(d) Major steam generator leak. Since the monitoring system will detect any leakage before two walls fail.

175. However, even these incredible accidents would not create a health hazard at the site boundary. Of the events that were deemed credible the effect of small sodium spills and small radioactive fires have been minimized by careful design, construction and handling. The activity release due to the melt-down of one fuel element in the core can be contained by the primary system. The effect of a fuel element melt-down in the fuel handling machine, due to failure of the machine's cooling system, can be corrected before a hazardous situation arises.

176. Several other minor possibilities exist, but after a careful study the maximum credible accident for HNPF was designated as the dropping of one fuel slug (or its equivalent in smaller pieces) from the fuel handling machine. There is no credible way that a whole rod or a whole fuel element could escape the process tube and drop through the bottom of the fuel handling machine, even if the entire bottom closure of the machine fell off. The activity released by the oxidized fuel is based upon these assumptions:

(a) A decay time of 10 hours after reactor shut-down;

(b) Release of 50% of the gaseous fission products, including iodine, from the fraction of the metal which is oxidized;

(c) The ventilation system remains on;

(d) A heat generation rate of 174 BTU/hr-ft in the hottest slug; and

(e) Oxidation rates of U-10 wt% Mo are as determined experimentally[9].

177. The calculated doses at the site boundary in the first hour are:

Direct radiation	6.8 mr
Iodine	8×10^{-2} mrad
Noble gases	4×10^{-5} mr

178. If any uranium carbide fuel is used in the initial core loading, the dropping of one uranium carbide fuel slug on the reactor floor will be designated as the maximum credible accident. For this case the calculated doses at the site boundary in the first hour are:

Direct radiation	9.5 mr
Iodine	2.2 mrad
Noble gases	1.1×10^{-3} mr

Fuel cycle

179. Objectives. Similar to SRE, the fuel is suspended from the top shield in a matrix of moderator elements, with fuel and other core elements being individually loaded through the top shield by the fuel handling machine. An objective in the design of the core calls for sufficient flexibility so that alternate fuels, other than the initial charge of uranium with 10 wt% Mo can be used. Such an alternate fuel might be a uranium metal alloy, a thorium or thorium oxide fuel, uranium oxide or uranium carbide. The first core with the uranium-molybdenum alloy is expected to give 3300 MWd/t average burn-up with a maximum of 7000 MWd/t. Ten uranium carbide elements may also be inserted in core

[9] See Final Hazards Summary Report for the Hallam Nuclear Power Facility, Appendix A.13, NAA-SR-5700, Atomics International, Canoga Park, Calif. (1962).

shortly after start-up if development proceeds as scheduled. Later on, a full uranium carbide core may be installed and the average burn-up is expected to reach approximately 12 000 MWd/t with a resultant drop in fuel cycle costs. Dry criticality was achieved by using 31 elements. Earlier calculations had shown that 23 elements would make the core critical, but the effect of the neutron source in the centre had been neglected. The source replaced a fuel element in an important location with considerable worth. The core loading at rated power is expected to be about 137 fuel elements containing 27 100 kg enriched uranium (975 kg U^{235}). The first batch of 151 fuel elements to reach the site cost approximately $2 800 000 including production tooling.

180. The fuel will be exposed to an average thermal flux of 1.0×10^{13} n/cm^2 sec, and an average fast flux of 8.0×10^{13} n/cm^2 sec. The average heat flux will be 160 000 BTU/ft^{2}hr with a maximum of 380 000 BTU/ft^{2}hr. The fuel elements will have an average temperature of 890^{o}F (1250^{o}F maximum) and the cladding will be at an average of 813^{o}F (945^{o}F maximum) with a design average film temperature drop of 35^{o}F.

181. Fuel elements will be washed and process tube and hanger rod removed. The element will then be encapsulated and shipped to the reprocessing site in a fuel cask.

182. Fuel element design. A fuel element consists of 18 rods in a hexagonal bundle configuration contained in a Zircaloy-2 process tube. The rods are made up of stacked slugs, which are either 3, 6, 9 or 12" long, with a total active length of 13.25 ft and an over-all length of 18 ft. The diameter of the fuel slugs is 0.59", there is a 0.025" thick sodium bonding, and the cladding is 0.010" stainless steel, thus forming a rod with an outer diameter of 0.66". The process tubes are arranged in a hexagonal pattern with 16" across the flats and 9.25" between the fuel elements. A hollow central tube contains spacers at one foot intervals.

Construction experience

183. The construction of HNPF was started in April 1959 and the reactor achieved dry criticality in January 1962. It is expected that full power operation will be reached by April 1963. Details concerning the time schedule for the project are given in Table 10.

Table 10

The Hallam nuclear power facility: Time schedule for the project

Item	Revised	Actual
Start site construction	April 1959	April 1959
Start training programme	July 1959	October 1959
Start installation of nuclear components	June 1960	June 1960
Delivery of reactor vessel	April 1960	July 1960
Installation of steam generator	December 1960	March 1961
Start pre-operational tests	April 1961	April 1961
Construction complete	October 1961	October 1961
Dry critical	-	January 1962
End pre-operational tests	August 1962	-
Wet critical	August 1962	-
Start post-critical tests	August 1962	-
Full power operation	April 1963	-
Turn over plant to CPPD	June 1963	-

184. After the scheduled public hearings, it is expected that the plant will operate at 15% full power since most of the system tests will be operated in this range.

185. Experience in designing the plant has shown the possibility of effecting numerous improvements. The layout of the plant is of the ranch style, being spread out following the SRE design. The use of bellows was minimized, which made the pipe-runs long whereas considerable savings would be possible by having a closely coupled, vertically orientated system using bellows. The three primary heat transfer loops are separate and heavily shielded to permit maintenance work being done on one of them while the other two are in operation. The shielding could be reduced. Fuel handling could also be simplified by using sodium tanks for fuel element storage instead of separate thimbles.

186. The three double-walled steam generators, costing $1 000 000 each, are expensive because they are meant to be very reliable. Instead, a single-walled steam generator could be employed as in the case of the Fermi reactor, thereby reducing the cost from $200 to $50 per square foot of heat transfer surface. Costs for conventional steam generators are about $20 per square foot.

187. According to CPPD, the operator of the plant, the greatest problems with the Hallam project were those connected with regulation and licensing, which required an unexpectedly large amount of time and effort.

188. There have been no major construction difficulties, although there were cases when the contractor had interpreted the specifications rather loosely as for normal construction practice. For instance a nitrogen system had some leaks which had to be repaired. The reactor vessel fell off the transportation truck into a corn field during shipment, but suffered no detectable damage and passed all tests satisfactorily. The welding of stainless steel required high standards of workmanship which were met by the careful setting up of specifications, the choosing of experienced welders (who were given two to three weeks of additional training on the job), and close inspection of work. Fuel elements and moderator cans have been assembled on the site to avoid difficulties in transportation.

189. Trouble was experienced in the lining of the sodium equipment cells where the thickness chosen for the carbon steel was only 3/32". To weld this on to the heavy T-bars to which it is attached created problems and leaks developed at several points. The lining should have had a minimum thickness of 3/16". The penetrations for wires into the sodium cells also developed leaks and had to be re-worked and filled with sealing resins.

System modifications

190. As a result of the pre-operational tests, several modifications to the system had to be made. The more significant ones are summarized below:

(a) An additional bypass line was provided around the secondary sodium expansion tank. Helium was being entrained in the sodium due to vortexing in this tank and the simplest corrective measure was to reduce the flow rate through the tank itself. Now, approximately 95% of the flow is routed through the bypass line;

(b) In the fuel element top connector the slip ring was modified to ease handling;

(c) The closed-loop cooling water system for the fuel storage cell, reactor-cavity liner, and radioactive gaseous waste compressor is undersized (by about 25%) and has an inadequate capacity to perform all of its functions simultaneously. Modification plans are being studied for installing pumps of higher capacity;

(d) The loading face shield is cooled with compressed nitrogen. Although the manufacturer supplied a large compressor, he underestimated the requirements and the system is still too small. At present the design calculations are being reviewed; and

(e) The use of reciprocating compressors has led to vibration problems in the nitrogen piping system. This is to be remedied by the installation of pulsation dampers.

191. Pre-operational tests. The purpose of these tests was to prove that the various systems and components would perform as designed or were adequate to meet the requirements of the operating plant.

192. In addition to the above-mentioned system modifications there were minor changes as a result of the pre-operational tests.

(a) Some wiring changes were made;

(b) More thermo-couples for the pre-heating system, intermediate heat exchangers, and steam generators were installed;

(c) The radioactive gas-vent system was redesigned. If the liquid (water) tanks were overfilled, liquid went into the vent system and was able to reach the sodium tanks. The vent system is now divided into two parts, one for water lines and the other for the sodium-helium system and these two parts join together only after many check devices;

(d) The amount of moisture contained in the insulation inside the reactor cavity was underestimated. This insulation was selected long ago on the basis of its compatibility with sodium, but recent checks have shown it to be hydroscopic (6% moisture by weight). Since there are 30 000 lbs of insulation, it can contain 1800 lbs of water (over 200 gallons). At SRE this problem was only minor, but at HNPF the insulation was installed under high humidity conditions; moreover, water was used to make a paste used for filling cracks, etc. The tank was then installed and the cavity sealed, the helium line being the only vent. In January 1962 checks under cold ambient conditions showed 25% relative humidity. Upon subsequent pre-heating tests the sodium leak detectors began to give signals, not for sodium but for the condensed water standing on the cavity floor. During the following months over 100 gallons of water were mopped up and the relative humidity is now less than 0.1%. As a further precaution a circulating line in the air vent will be installed to remove any additional moisture; and

(e) There were three cases of failure in the bellows seal valves owing to leakage. In two cases the valves were eliminated and in the third case the bellows were replaced.

193. During the hot-sodium system test, the pumps, valves, heat exchanger piping, control rods, etc. performed satisfactorily. Furthermore, the graphite blocks have been checked out and found to be all right. Except for the replacement of a few gaskets, the fuelling machine has given no trouble.

194. On the other hand, one of the large throttle valves in the main sodium-system stuck while it was two-thirds open. The back seat (stellite ring) was cracked and has been returned to the manufacturer for repairs. It was also found that the pipe hangers inside the cavity did not offer adequate support due to improper installation, moisture accumulation, and rusting. This situation has since been corrected.

Cost data

195. The breakdown of estimated costs for HNPF is given in Table 11.

Table 11

The Hallam nuclear power facility: Cost breakdown
(in thousands of dollars)

Item	Cost
Structure and equipment	36 700 a/
Turbogenerator plant	15 900 b/
First fuel loading (151 elements)	2 800 c/
Training and miscellaneous	150 d/

a/ CPPD provided $5 730 000 of this and USAEC the remainder.

b/ Provided by CPPD.

c/ Provided by USAEC.

d/ CPPD paid the salaries of the men under training and USAEC the rest.

Operating personnel and training

196. The staffing plan for the operation of the reactor station is shown in Table 12.

Table 12

The Hallam nuclear power facility: Staffing plan

Category	Number of persons
Administration	
Plant superintendent	1
Assistant plant superintendent	1
Clerical	3
Operation	
Shift supervisors	5
Reactor engineer	1
Chemical engineer	1
Performance engineers	2
Control room operators	9
Equipment operators	10
Auxiliary operators	8
Health physicist	1
Health physics technician	1
Maintenance	
Maintenance supervisor	1
Material handling men	5
Machinist	1
Mechanic	1
Pipefitter-welder	1
Electronic specialist	1
Instrument repair man	1
Utility men	2
Instrument technicians	4
Electricians	4
TOTAL	64

Integration of the reactor into the utility system

197. To meet their power commitments until HNPF is completed, CPPD placed in operation a conventional coal-fired plant in the spring of 1961. Later, when the reactor is undergoing start-up and the plant equipment is being checked out, the turbine-generator will be supplied with steam produced by the nuclear facility, and the conventional boiler will be used on a standby basis. After the reactor reaches full-power operation, increasing power demands of the area may be fulfilled by using the conventional boiler to supply steam for a second turbine-generator.

198. The sodium heat transfer system is designed to be load-following and has automatic regulation between 15% and 100% load at a rate of 5 MWe per minute, which is more rapid than that of a conventional modern plant. The nominal steam conditions at the turbine throttle are 825°F, 800 psig, and 710 000 lb/hr as against the design conditions of 843°F, 800 psig, and 752 000 lb/hr.

Selected references

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

STARR, C. and DICKINSON, R.W., Sodium Graphite Reactors, Addison-Wesley Publishing Company, Inc., Reading, Mass. (1958)

Preliminary Safeguards Report for the Hallam Nuclear Power Facility, NAA-SR-2700, Atomics International, Canoga Park, Calif. (1958)

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

GRONEMEYER, F.C. and MERRYMAN, J.W., 75 000 Kilowatts of Electricity by Nuclear Fission at the Hallam Nuclear Power Facility, AI-5272, Atomics International, Canoga Park, Calif. (June 1960)

Schedule Status Report, Progress on Hallam Nuclear Power Facility, issued monthly by Atomics International, Canoga Park, Calif.

BEELEY, R.J. and MAHLMEISTER, J.E., "Operating experience with the sodium reactor experiment and its application to the Hallam Nuclear Power Facility", Small and Medium Power Reactors, Vol. I, STI/PUB/30, IAEA, Vienna (1961) p.585

JONES, E., "Integration of the Hallam Nuclear Power Facility into the Consumers Public Power District electric system", Small and Medium Power Reactors, Vol. II, STI/PUB/30, IAEA, Vienna (1961) p.171

"Hallam Nuclear Power Facility", Directory of Nuclear Reactors, Vol. IV, STI/PUB/53, IAEA, Vienna (1962) p.307

VI. THE EXPERIMENTAL GAS-COOLED REACTOR

General

200. The experimental gas-cooled reactor (EGCR) is a dual purpose reactor designed to produce electrical power and to provide facilities for testing fuel and other materials to be used in advanced types of gas-cooled reactors. It is a graphite-moderated helium-cooled reactor having an output of 85 MWth and a net electrical generation of 21 MW. The fuel consists of enriched uranium dioxide clad in stainless steel.

201. The USAEC-owned reactor is located in the Oak Ridge area, and will be operated by TVA. Its design represents a composite of the features of a number of earlier designs developed by the respective design contractors such as these for the partially-enriched gas-cooled power reactor of Kaiser Engineers and Allis-Chalmers, and the GCR-2 of ORNL. The final design is being done by Kaiser Engineers, Allis-Chalmers and ORNL under the direction of AEC's Oak Ridge Operations Office. The H.K. Ferguson Company is the building contractor.

202. As at 30 June 1962, 40% of the entire work had been completed, with 95% of the actual construction (excluding nuclear parts) having been completed. It is expected that the plant will go into operation early in 1964.

History of the gas-cooled reactor programme of the United States

203. The very first studies of gas-cooled reactors in the United States were carried out in connection with the reactors for plutonium production which were to be built at Hanford[10]. The first design programme for these reactors was based on the utilization of gas cooling. But further studies indicated that natural uranium, graphite-moderated, plutonium-producing reactors could be successfully cooled by water. Since it was possible to use conventional materials at the lower temperatures in water-cooled systems, the original plan for gas cooling was abandoned. After 1943 there has been only one other serious study of gas-cooled reactors for power production in the United States, that was the Daniel's Power Pile Project at Oak Ridge that was undertaken immediately after the Second World War.

204. The earlier studies made in 1943 seemed to indicate that it was difficult to achieve sufficient high-power densities in gas-cooled systems to be of interest for power production. But after the very satisfactory performance of the first British and French dual-purpose reactors in Calder Hall, and Marcoule and the optimistic nuclear power plans of those countries, interest in the gas-cooled reactor was revived in the United States in 1957. However, evaluation studies done in that country have indicated that natural uranium-fuelled, CO_2-cooled, graphite-moderated systems as used in the United Kingdom and France are not economically attractive in the United States because of high capital costs. So the major development effort has been concentrated on reactors with partially enriched fuel. It is argued that enrichment has several advantages as it permits:

(a) The use of materials (e.g. uranium dioxide and stainless steel) capable of operation at higher temperatures, leading to higher efficiencies;

(b) Undermoderation in the fuel lattice, or in other words more fuel per unit-volume of moderator and hence higher power density in the core;

(c) Higher fuel exposure within the limitations set by the physical properties of the fuel; and

(d) Construction of reactors which are smaller in physical size than comparable natural uranium systems, thereby having lower costs/kW installed.

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[10] According to The ORNL gas-cooled reactor, GCR-2, ORNL-2500, Oak Ridge National Lab., Tenn. (1 April 1958).

Important design features

205. A summary of important data concerning the reactor is set out in Annex II and a flow diagram of the reactor system is shown in Figure 2.

206. Objectives. The dual role of this reactor as a prototype power-producer and experimental facility for testing advanced gas-cooled reactor fuel elements and materials necessitated some compromise in the design; for example, as a power demonstration reactor, it would not require the type of containment shell which is used to permit more flexibility in experimental operations; and as a test facility, the number of fuel holes directly accessible would have been increased and probably different nuclear characteristics would have been specified.

207. The selection of the proper coolant for the gas-cooled, graphite-moderated reactor involves careful consideration of:

(a) The heat transfer and heat transport characteristics;

(b) The chemical compatibility with the moderator and the fuel cladding;

(c) The availability and costs; and

(d) The influence on the design of the system.

208. Among gases, the heat transfer properties of helium are excellent, second only to those of hydrogen which has such disadvantages as inflammability, diffusion through metals, metal embrittlement, and chemical reactivity at elevated temperatures. Those problems are being studied but hydrogen is not yet a suitable reactor coolant. Therefore the choice lies presently between helium and carbon dioxide. Helium is chemically inert and compatible with all fuel, moderator, and cladding materials throughout the temperature range of interest. Although CO_2 is used in reactors of the Magnox type, these reactors operate at relatively lower temperatures. At high gas temperatures of 1100° to 1200°F, which are envisaged in gas-cooled reactors of advanced design, CO_2 would react with graphite. In addition, the gas could attack stainless steel which would be at temperatures of 1400° to 1600°F with the coolant gas at 1200°F. It does not appear that the cost and availability of helium should seriously limit its future use as a reactor coolant in the United States since there will be some 75 billion cubic feet available by 1975 if the helium-recovery programme of the Bureau of Mines is put into effect. The decision to adopt helium as the coolant for EGCR was mainly due to its chemical inertness at reactor operating temperatures, and its excellent heat transfer properties.

209. Core. The graphite structure of the core is made up of 16" square graphite blocks 19 ft 4" long, held together, at top and bottom, by means of steel grid structures. Each of the blocks contains four fuel channels. The graphite structure is a cylinder, 19 ft 4" high and with a diameter of 15 ft 10", its active part being approximately 14 ft 6" high, with 12 ft diameter. It contains 234 fuel channels, 5.25" diameter, in an 8" square lattice, two experimental through-tubes in lattice position, and two experimental through-tubes at the periphery up the core. Provision has been made for a total of eight test loops, but owing to mounting expenses only the four mentioned above will actually be completed and put into operation. However, in the initial stages of reactor operation none of those loops will be installed. Each fuel channel contains six fuel elements. There are 21 control rods which are equally spaced over the core.

210. Pressure vessel. The core is contained in a 20 ft diameter, 46 ft high cylindrical, carbon steel pressure vessel with hemispherical ends. The cylindrical part is 2.75" thick (nominal), and top and bottom heads are 4" thick (nominal). It is designed for 350 psig and 650°F. The coolant gas enters the reactor at the bottom with a temperature of 510°F, passes up through the core, and exits from the top at a temperature of 1050°F. The pressure vessel temperature is well below 600°F because of the use of a thermal barrier which consists of a 1" thick stainless steel cylinder with a hemispherical head at

the top and open at the bottom. The barrier is fitted within the pressure vessel to form a 2" wide vessel cooling passage between the pressure vessel and thermal barrier. A cooling stream of gas passes through this passage counter to the main coolant gas stream from top to bottom and combines there with the incoming main stream.

211. The thermal barrier is internally insulated by 2.25" of stainless steel reflective insulation; the exterior insulation face is clad with 0.125" stainless steel. The exterior of the pressure vessel is also insulated with conventional high temperature insulation material.

212. Helium leakage. The design of the primary system requires that the total leakage be not more than 1% of the system-volume per day; however, it is expected to be less than 0.1%. The special arrangements to prevent helium losses are:

(a) Rotating shaft seals are backed up by water-buffer systems and enable leak-off recovery of the helium-water mixture;

(b) Valves up to 8" are bellows-sealed;

(c) Valves larger than 8" have leak-recovery systems connected to the packing gland; and

(d) All mechanical joints are double gasketed and in the case of the vessel, fuel charge and experimental nozzles are backed up by a secondary blank flange.

213. Shielding. The shield thicknesses were calculated to limit the dose rate in areas with unlimited access to less than 0.75 mrem/hr; for areas with limited access requiring daily occupancy, the shielding calculations were based on a maximum dose rate of approximately 10-20 mrem/hr during the times occupancy is required. Shielding is provided on sides by the 1" stainless steel thermal barrier, the 2.75" carbon steel vessel wall, and by 10 ft of ordinary concrete; on top by the 1" thermal barrier, the 4" vessel head, and by 7 ft high-density ferrophosphorous concrete; at the bottom by the 4" vessel head, and by 7 ft high-density ferrophosphorous concrete. Furthermore, high density ferrophosphorous concrete is used in areas where the specified thickness of 10 ft of ordinary concrete had to be reduced.

214. It is perhaps worth noting that the fast neutron dose to the reactor vessel is expected to be of the order of 1 to 2 x 10^{18} nvt and that the dose is reduced by the graphite reflector and the 1" thermal barrier. The dose is highest at the mid-plane of the vessel.

215. Containment shell. The entire reactor and its coolant system is housed within a 112 ft diameter cylindrical carbon steel shell with a hemispherical top and a dished bottom. Its height is 216 ft of which approximately 150 ft are above grade. Its nominal thickness is 0.875", but varies from top to bottom. It is designed for a pressure of 9 psig; the leakage rate is approximately 0.5% of total volume per day, excluding penetrations and air locks.

216. Fuel handling machines and fuel storage. Since EGCR is to be, in part, a power reactor, it is designed so that during full load operation:

(a) Charging, discharging, and re-positioning of fuel can be done;

(b) Failed elements can be located and removed; and

(c) As many service and maintenance functions as practicable can be undertaken.

217. Fuel is normally charged, discharged, and re-positioned by a 390-ton machine located at the bottom of the reactor. The machine is a carbon steel vessel 37 ft high with an outer diameter of 8 ft 6", containing a flexible ram assembly and a group of rotating platforms mounted on a movable bridge and carriage assembly which permits the machine to be positioned under each nozzle individually. The machine is capable of holding 12 fuel elements, which is sufficient for two fuel channels. It is capable of removing the nozzle

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(c) As many service and maintenance functions as practicable can be undertaken.

217. Fuel is normally charged, discharged, and re-positioned by a 390-ton machine located at the bottom of the reactor. The machine is a carbon steel vessel 37 ft high with an outer diameter of 8 ft 6", containing a flexible ram assembly and a group of rotating platforms mounted on a movable bridge and carriage assembly which permits the machine to be positioned under each nozzle individually. The machine is capable of holding 12 fuel elements, which is sufficient for two fuel channels. It is capable of removing the nozzle

shield plug, releasing and lowering spent fuel elements from the core and replacing them by new fuel elements, and transferring spent fuel to the storage pool. The machine has to be dismantled and inspected annually to check the wear of various components.

218. At the top of the reactor is a 180-ton service machine which is capable of removing and replacing control rods, inserting and removing flux scanning equipment, monitoring fuel channel temperatures, and removing and replacing experimental or conventional fuel elements under full power operating conditions. When the reactor is shut down, the service machine will also perform other service functions, such as the insertion of television equipment for visual inspection of the core and reactor vessel, insertion of experimental equipment and test devices. The machine consists of two separate grapples for lifting, one attached to a ball lead screw jacking arrangement for handling heavy objects, such as the shield plugs, and the other a motor-driven cable and winch arrangement for handling control rods and fuel elements. Storage space is provided by means of a turret arrangement inside the 6 ft diameter, 50 ft high-pressure vessel of the service machine.

219. Spent fuel elements, control rods, and other active equipment are stored in a stainless steel lined pool, 39 ft x 12 ft, 20 ft deep. After a minimum storage of 100 days, fuel is loaded under water into a shielded cask for transport to the reprocessing plant.

Experimental facilities and testing programme

220. The experimental facilities within this reactor are of two types: those utilizing the experimental through-tubes and those using part of the original fuel channels. The first type allows considerably more flexibility in the scope of experiments and is intended to permit the testing of various advanced type reactor fuel elements, such as beryllium-clad, graphite-clad, beryllia-urania, and uranium-carbide elements in a variety of environments, whereas the second type may be used to test fuel elements of less advanced design where there is less probability of significant fission gas release as with the elements designed for EGCR itself.

221. Experimental through-tubes. As mentioned in paragraph 209 above, it was planned to have eight test loops, but owing to the high cost of mounting them only four of these will be actually completed and put into operation. Two of the loops have tubes with an inner diameter of 6.25" each, located near the centre of the core in lattice positions, the two others are near the outer periphery of the core in the graphite reflector and have an inner diameter of 9.25". Each of these loops is designed to operate as a completely independent system having separate heat exchangers and pumping equipment, and using its own coolant which is separated from the primary reactor coolant system. Thus it can be operated with coolants, such as carbon dioxide, nitrogen, or even hydrogen, under a maximum pressure which has been set tentatively at 1000 psig and at coolant outlet temperatures up to 1300^{o}F. It is thought to be possible to obtain 1.5 MW of thermal capability from each of the experimental loop facilities; the available unperturbed neutron fluxes are 2×10^{13} n/cm^{2}sec (maximum) for the central 4.5" diameter tubes and 0.8×10^{13} n/cm^{2}sec (maximum) for the outer 9" diameter tubes. Experimental assemblies up to 38 ft length can be handled by the reactor service machine, and longer ones may be handled by special arrangements. Access to the tubes is provided by flanged closures at the top of the reactor and may be opened without opening the primary reactor coolant system, thus, within limits, allowing experiments to be serviced under full reactor power.

222. Secondary test facilities. Eight fuel channels positioned directly under reactor nozzles are accessible for insertion of fully contained fuel elements or other materials useful in gas-cooled reactor systems; five of them are initially to be furnished with instrumented fuel elements. Experimental installations must be compatible with reactor operating conditions and amenable to heat removal by the primary reactor coolant. Fission gas release from these experimental holes as well as from all other fuel channels is monitored by means of a burst slug detection system which is not greatly different in principle from the one used at Calder Hall. Experiments may be removed and inserted by the reactor service machine while the reactor is at full power, subject to certain safety and operational restrictions.

223. While EGCR does not offer the high-flux levels which are available in some other test reactors, it is unique in that the irradiation facilities are large by test reactor standards and permit the installation of long elements comparable to those to be used in actual gas-cooled power reactors. Since the interplay of variables affecting the performance of fuel elements becomes so complex that it is difficult to separate and investigate these variables individually, the EGCR will be extremely useful as a comprehensive test facility available for testing such full-scale gas-cooled prototype fuel elements. It is visualized that the experimental capability of this reactor will be useful not only in the initial development of fuel elements, but in the pilot plant operation of fuel elements for large-scale reactors where it is essential to know the performance of elements before carrying the complete charge to its full burn-up at maximum operating conditions.

Safety

224. Safety in design. EGCR, being a gas-cooled, graphite-moderated reactor, has the well-known and outstanding safety characteristics of such reactors. The most important accident in reactors of this type results from loss of coolant and/or coolant flow. Therefore it is necessary that the system be designed so that shut-down heat can be removed. This is accomplished by radiation and thermal convection, assisted by the big heat storage capacity of the graphite moderator. In addition, an emergency cooling system is provided.

225. The primary cooling system of EGCR operates on two loops so that in case of a failure of one blower or of a leak after the blower, sufficient coolant is circulated through the core by means of the second loop. However, if the reactor remains pressurized with both blowers shut down, after-heat can be removed safely by natural convection. If the reactor becomes depressurized, some type of forced circulation is required. As mentioned in the preceding paragraph an emergency cooling loop is being provided which will be completely separated from the primary cooling circuit, so that in case of damage to the primary blowers, this cooling system could be used in emergency.

226. As stated in paragraph 208 above, at the high operating temperatures of EGCR, namely a maximum graphite temperature of 1100°F, the graphite would easily be oxidized by air, steam, etc. However, under normal operating conditions, the oxidization does not present a hazard, since the level of those oxidizing impurities is held well below dangerous limits; the problem arises only in emergency cases. An extensive research and development programme on graphite oxidization has been carried out at Hanford on the following subjects:

(a) Rate enhancement due to impurities;

(b) Rate enhancement due to irradiation;

(c) Rate reduction due to coolant gas purification;

(d) Effects due to activation energy of gamma rays;

(e) Investigation of spread factors; and

(f) Presence of air in various concentrations due to depressurization accident.

227. It has been found that no problems arise during normal operation and as long as the reactor remains pressurized.

228. A rupture in the heat exchangers would result in an in-leakage of steam into the reactor. At high temperatures, say 1800°F, this steam would react with the graphite to form carbon monoxide and hydrogen. However, below 800°F no reaction takes place and up to 1200°F it can be tolerated. Therefore, to prevent such reactions it is necessary to keep the operating temperature of the graphite at low values, as has been done in the design of EGCR.

229. Nevertheless, great care has been taken to prevent steam leakage into the helium system. A very sensitive detecting device will register such a leakage within a very short time; it will not only determine whether moisture is leaking into the system but also the source of in-leakage. It is hoped that the device can be made sufficiently sensitive so that it will differentiate which of the two steam generators is leaking and permit the operator to isolate the defective one for servicing. The steam generators have been designed in such a way that it is possible to plug leaking tubes.

230. Another safety feature of this reactor is its negative temperature coefficient of reactivity. It is for the moderator $-2.0 \times 10^{-3}\% \frac{\Delta k}{k}/{}^{\circ}C$ and for the fuel $-2.1 \times 10^{-3}\% \frac{\Delta k}{k}/{}^{\circ}C$. Besides, the reactor is designed in such a way that it will be held sub-critical by only 14 of the total number of control rods. Since most of the hazards will be closely connected with the experimental loops, none of them will be operated in the initial stages of reactor operation.

231. Control. Control of the reactor is provided by means of 21 control rods composed of boron carbide rings clad in Type 304 stainless steel. These rods are driven by a gear motor assembly attached to a cable winch arrangement. In case of a scram, the rods are allowed to fall into the reactor by gravity by releasing a clutch mechanism. The control system is designed to operate the reactor as a base-load plant with load-following capabilities, and to operate at steady-state conditions for experimental purposes.

232. The control rods are composed of four sections. The lower three sections consist of boron carbide bushings clad with stainless steel and the upper section is stainless steel. The four sections are connected by a stainless steel rod through the centre. The operating temperature of the control rods will vary from approximately 450°F to approximately 1700°F. Each of the boron carbide-containing sections is vented to the coolant stream to prevent the build-up of pressure, but is designed to prevent the escape of B_4C particles into the reactor. All of the surfaces of the stainless steel that are in contact with B_4C are copper plated to prevent reaction of the stainless steel with boron at elevated temperatures. The control rods are cooled by the helium stream used to cool the control rod drive mechanism. The flow is down from the drive through the shroud tube into the inside of the rod and out at the bottom. The outer surfaces of the B_4C-bearing sections are oxidized to increase the emissivity of these surfaces, since about half of the heat generated in the rod is transferred to the surrounding graphite by radiation.

233. The control rods will operate in a vertical position and must move freely in the channel and shroud tube. The clearance in passing through the shroud will be small and, therefore, no protuberances can be tolerated. The control rods must be capable of withstanding the stresses imposed, which may include a deceleration of 25 g during shock loading. In addition, the articulated construction must provide sufficient flexibility to allow movement should the control rod channel become distorted due to shrinkage of the graphite core.

234. Site. The reactor is located downstream on the right bank of the Clinch River. This site is in the Oak Ridge area owned by the Government. It is approximately five miles south of the city of Oak Ridge, 18 miles west of the city of Knoxville, and two miles east of ORNL X-10 area.

235. At present, the nearest permanent installation in the Oak Ridge area is the ORNL shop area which is approximately 1.25 miles west of the site. The land across the river is, however, privately owned, so that the closest uncontrolled approach to the site is approximately 1000 ft. This land is presently being used for farming. It is not now readily accessible, and is not in the direction of any population shift, so that its development as a real estate project seems improbable.

236. Analysis of maximum credible accident. The basis of the maximum credible accident has for some time been fluid. The present design evaluation indicates that the maximum credible accident will allow for the failure of something like 50% of the fuel elements, releasing approximately 90% of the gaseous fission product activity contained within the cladding but not held up within the UO_2.

237. In most reactors a double-ended rupture of the primary coolant system is considered the worst accident. This is not the case for this reactor. A worse accident than this would be the formation of a 60 square inch hole in one of the cold legs of the primary coolant pipes. Such a hole would depressurize the vessel in 30 seconds and disturb the flow pattern in the core. This would result in the worst temperature transient in the fuel claddings and generate the greatest number of fuel element failures. Assuming no flow in the core, 300 to 350 fuel elements would fail. At the same time the contents of the steam generators would flash into the containment shell, giving a pressure rise of 8. 9 psi; but since the shell has been designed for 9 psig and was tested at 11.25 psig, it would hold.

238. The construction of a separate loop with a small compressor located outside is under consideration to circulate the mixture of helium, steam, air, carbon dioxide, etc. for cooling, since for such an accident shut-down cooling will be needed for 90 days. A nitrogen purge is being constructed to remove or dilute the generated amount of hydrogen.

Fuel cycle

239. Objectives. In developing the fuel for EGCR, the principal objectives were to design a fuel assembly which was simple to fabricate and to assemble, and at the same time would have low fabrication costs, and permit of operation at high temperature for a long time. The fuel should be capable of heating the coolant gas to $1050^{\circ}F$ initially, and as high as $1200^{\circ}F$ in later operations. The expected fuel burn-up is 10 000 MWd/t. ORNL has conducted an extensive experimental programme to design and develop an EGCR fuel assembly. Fuel assembly specifications and production procedures were developed, and a contract has been awarded to the Westinghouse Electric Corporation to manufacture complete fuel assemblies for the initial reactor loading under fixed price conditions.

240. Fuel element design. The fuel assembly consists of a seven-rod cluster of cored uranium dioxide pellets in stainless steel tubes, positioned within a graphite tube, or sleeve. The graphite sleeve fits into a channel in a graphite moderator block. The fuel channels are oriented vertically in the reactor vessel, and each channel is long enough to accommodate six of the graphite sleeves, stacked end to end.

241. The smallest unit in the fuel assembly is a pellet of uranium dioxide, in the shape of a hollow right circular cylinder, with an outer diameter of 0.705", an inner diameter of 0.323" and a length of 0.75". Thirty-five pellets are stacked in a Type 304 stainless steel tube (0.75" outside diameter, 0.020" wall thickness, and 26.6" long). An insulation disc of magnesium oxide is placed at each end of the column of uranium dioxide pellets, and the tube ends are closed by stainless steel end caps. An integral part of each end cap is a pin, aligned with the longitudinal axis of the fuel tube. Seven completed fuel tubes are positioned into a hexagonal cluster by attaching a spider to each end of the cluster. The design of the assembly is such that only one end of each tube is fixed, permitting thermal expansion. Mid-plane spacers have been brazed to each fuel tube before they were filled to minimize bowing during reactor operation. Each seven-rod cluster with its spiders is contained in a graphite sleeve, which has an outside diameter of 5", inside diameter of 3" and a length of 29". The ends of the graphite sleeves are designed in ball and socket fashion in order to provide a good fit when six sleeves are stacked in a fuel column, regardless of the alignment within the fuel channels. Graphite spacers are attached to the sleeves to assure that the moderator cooling annulus is maintained.

242. As described in paragraph 209 above, the core is made up of graphite blocks 16" square by 19 ft 4" long, each of which has four channels of 5.25" diameter. Each of the 234 fuel channels contains six fuel element assemblies.

243. The uranium enrichment will be 2.46 wt% U^{235} for the initial EGCR loading. The total fuel load in the reactor will contain 12 200 kg U in 13 900 kg UO_2, for a loading of 1404 fuel assemblies.

244. Fuel element fabrication. The UO_2 pellets are formed by pressing, sintering and grinding to size. Pellets must meet chemical purity, dimensional, and density specifications. The following steps are involved in assembling the fuel. A mid-plane spacer and an end cap are joined to a stainless steel tube. One MgO disc is loaded into the tube, followed by 35 UO_2 pellets, then another MgO disc. These components are then subjected to a drying treatment prior to insertion into a controlled atmospheric chamber for welding of final closure weld. The second end cap is then welded in place. The top end-cap pin of each of seven tubes is welded to the top spider. The cluster of tubes is inserted into a graphite sleeve. The bottom spider is then attached by welding to three of the lower end-cap pins.

245. Fuel management. Initially, it is planned to load 234 channels of the reactor. This would be 1404 fuel assemblies and 9828 fuel rods. The irradiation level attained by the initial loading might be 7000 MWd/t, but there will be a fraction representing the 10 000 MWd/t target lifetime, and it is estimated that some fuel assemblies will be carried to the 30 000 MWd/t limit.

246. Since EGCR is intended to be a test-bed for new fuel element designs as well as a power demonstration reactor, it is likely that as the initial fuel is discharged, the replacement fuel will be of different design. Hence, fuel management will not be directed toward reaching an equilibrium refuelling schedule such as would be desirable for repetitive loading of the same design of fuel element. Design changes will be aimed toward increased power output per unit weight of uranium. This might be accomplished by increasing the outside diameter of the fuel pellets and also increasing the inside diameter of the pellets, keeping the weight of uranium per pellet unchanged. This will provide more heat transfer surface per unit length of tube and more heat output per unit weight of uranium. In addition, heat transfer may be improved by roughening the surface of the cladding material.

247. Fuel costs. Fuel costs of EGCR are given in Table 13. It must be borne in mind that this reactor has not been optimized for low fuel costs since it has an experimental character and will be used extensively as a testing facility. In these calculations a life-span of 7000 MWd/t has been assumed, which is valid only for the first core, since with future cores a 10 000 MWd/t burn-up will be achieved. Fuel fabrication costs are assumed to be $52/kg U, and the plutonium credit is $9.50/g.

Table 13

The experimental gas-cooled reactor: Fuel costs

Item	Mills/kWh
Fabrication	1.19
Transportation	0.23
Chemical processing, conversion, losses	0.94
Burn-up	1.93
Use charge	0.98
Less plutonium credit	- 0.63
TOTAL	4.64

248. For a large reactor of similar type with 550 to 1000 MWe output the fuel costs are expected to be about 2 mills/kWh.

Table 14

The experimental gas-cooled reactor: Fuel assembly and core costs

(in dollars)

Item	Cost
A. FUEL ASSEMBLY	
Tubing	24
Graphite sleeve	33
Spiders	61
Miscellaneous components	14
Conversion to UO_2	58
Fabrication of pellets	62
Brazing of mid-plane spacer	35
Fabrication and inspection of rods	55
Assembly and inspection of fuel assembly	10
Sub-total	352
Transportation, interest, etc.	135
TOTAL	487
B. FIRST CORE	
1709 assembled fuel assemblies, $487 each	832 283
30 unassembled fuel assemblies, $446 each	13 380
TOTAL	845 663[a/]

a/ Each assembly contains 8.8 kg uranium, therefore the unit cost is $55/kg U.

Construction experience

249. Construction of this reactor was started in August 1959 and it was estimated that it would be ready after three years. But owing to difficulties the work could not be completed by that date. The delay has been caused mainly because the reactor vessel has not yet been delivered. Another factor may be the delivery of instrumentation, because for a reactor such as this a complex and elaborate system is needed using 3000 devices and sensors. Moreover, the hazards analysis has changed the instrumentation requirements. Details concerning the time schedule for the project are given in the table below.

Table 15

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

Item	Start		Complete	
	Original estimate	Actual	Original estimate	Actual
Construction	August 1959	August 1959	March 1964	
Completion of construction, except for testing	-	-	December 1963	
Initial criticality	April 1964		-	
Planned power operation	February 1965		-	
Operator training	November 1960	November 1960	Premised on completion of construction in October 1962	
Containment shell erection				
Phase I	June 1960	April 1960	August 1960	August 1960
Phase II	April 1961	June 1961	December 1961	December 1961
Reactor vessel installation	June 1962		December 1962	
Main blowers	September 1962		October 1962	
Charge machine	October 1962		December 1962	
Service machine	June 1962		October 1962	
Steam generator	June 1961	November 1961	December 1961	December 1961

250. Reactor vessel. The construction of the reactor vessel is being done under a fixed price contract. The total costs will amount to approximately $2.5 million. The main problem in the construction of the vessel was physical handling and the welding of the different pieces according to the close tolerances given by the design specifications, i.e. in the case of the nozzles 0.125" over 72 ft. This is a time consuming and expensive task. All face penetration welds at the nozzles have to be thoroughly inspected by X-rays and by ultrasonic tests. A complete strain analysis had to be done.

251. To conduct all those procedures it was necessary to develop special processes and techniques for loading, inspection, etc. Fabrication and testing time added to the delay. The contract was signed in June 1960, and the latest estimate of completion, i.e. the time when the vessel leaves the vendor's plant, has been given as November to December 1962. This is about twice as long as originally estimated.

252. Charge machine. The charge machine is also being fabricated under a fixed price contract according to a design of Kaiser Engineers.

253. There was a three-phase research and development programme foreseen on dry film-lubricants for bearings and gears. After satisfactory completion of the first two phases, all dry lubricants failed in the third phase, which was a long-life test. This necessitated a complete redesign.

254. The new design is for dry bearings and gears requiring no lubricants at all. Bearings and gears are simply overdesigned (about 15 times) to permit wear and tear. Tests were carried out using helium of 500°F for 500 hrs with inspections afterwards. This lead to the selection of materials and components.

255. Other problems arose with the seals. The seals have to withstand helium at temperatures of up to 500°F and 350 psig. Seals of silicon-elastomers have failed to meet the proof tests. Now new types of seals are under investigation. There are many elastomers which can withstand temperatures from 400° to 425°F. Perhaps the solution of this problem will be the use of a cooler which will bring the temperature down to such a level.

256. Another reason for redesigning and consequent delay is the need for achieving a satisfactory break in the positioning mechanism. Difficulties in proof testing occurred too.

257. The charge machine is 65% ready and expected to be delivered in March 1963. The final cost estimate is $2.5 to $2.6 million.

258. Service machine. This machine is being built after a design by Allis-Chalmers. It was ordered in November 1960 and is expected to be delivered in August 1962. The total costs will be about $2.5 million.

259. Similar to the charge machine, this machine is very complex but operating requirements and quality control are less severe since it operates under helium of only 125° to 150°F. Cold gas passes through the handling tools. The original design, i.e. dry bearings and gears, stayed. The weight of the machine is 400 to 450 tons, similar to that of the charge machine.

260. Main blowers. The blowers are being built under a fixed price contract. Both blowers with accessories will cost $420 000.

261. The fabrication problem with the main blowers lies in metal working. The outer housings are being made to the same standards as the reactor vessel. Shaft sealing is accomplished by water buffer seals developed by Allis-Chalmers under a $350 000 research and development programme. Those seals have been tested under simulated operating conditions and have passed these tests excellently. A slight in-leakage of water is removed by the recovery system.

262. Helium leakage. Leak tightness in this all-welded system presented no major problem, although close design specifications had to be followed and rigorous inspections had to take place. On the other hand enough experience exists in handling helium. Occurring losses were not considered as being disastrous since helium is readily available and the costs of an extremely low leakage system would be unduly high.

263. Fuel elements. As mentioned before, the contract for manufacture of the fuel elements has been awarded to Westinghouse Electric Corporation. But ORNL conducted studies in this direction, and among others fabricated on pilot plant scale 50 fuel elements. This was to check the ability to maintain tolerances, try welding techniques, and cap performance, etc; also man-hour studies were conducted. These studies resulted in the conclusion that no fabrication difficulties should occur. Also the estimated costs were well within a 10% margin of the contract price; it turned out to be $55/kg U excluding use charges and inventory but including material losses.

264. The dimensional specifications can be met easily. But since uranium dioxide is fairly brittle, cracking and chipping presents difficulties and a most rigorous inspection has to take place before loading the tubes. In the case of tubes, close tolerances are required, ± 5 mils outer diameter, and ± 1 mil inner diameter. The machining of the graphite sleeves is also a critical item.

Design experience

265. From the designing of this reactor the following lessons have been learnt which are being incorporated in the design study of GCR-3, a 500 MWe unit:

(a) Fuel tubes would have a larger diameter, resulting in a reduced gas pressure and in lower costs;

(b) To obtain better heat transfer conditions surface roughening would have to be studied;

(c) A concrete reactor vessel similar to those used by some British and French reactors would be considered;

(d) Steam driven blowers would be used instead of the electrical ones in EGCR;

(e) On-stream refuelling as is done presently represents great technical difficulties resulting in high capital costs. On the basis of a thorough economical analysis it would have to be cleared whether this could be avoided; and

(f) Due to the flux gradient graphite would undergo an uneven shrinkage process. This would have to be overcome by either using a flatter flux distribution or by provisions by which it would be possible to shuffle graphite pieces.

Cost data

266. Initially it was expected that the total cost of this plant would be $30 million. Later, this figure was revised to $40 million, and the latest estimates indicate that the costs could reach $52 million. This increase is mainly due to increases in the engineering costs of the reactor system, the inclusion of some additional safety features and some anticipated, but not yet accounted for, increases in the final cost of the reactor vessel and the fuel charge and service machines.

267. Also EGCR, being a scaled-down model of a large power reactor, uses charge machines, not because they are necessary for a plant of this small size, but because they will provide practical experience for a big plant of 300 MW or higher output, where it would be necessary to use such machines. The two service machines used for EGCR cost $2.5 million each.

Operating personnel and training

268. Since EGCR has been designed as a plant both for the production of power and to provide an experimental facility, it has two independent groups of personnel, namely an operation and maintenance group like most power reactors, and a technical progress group which takes care of the experimental programme and reactor development. In addition, being an ORNL project, its personnel is closely connected with all the other ORNL staff. The actual operation of the power plant will be under TVA and it will be interconnected with its grid.

269. The staffing plan for EGCR is given in the table below:

Table 16

The experimental gas-cooled reactor: Staffing plan

Category	Number of persons
Administration	
Manager EGCR project	1
Assistant to manager	1
Administrative Officer	1
Reports control	1
Clerks, typists, accounting, stores	12
Operation and maintenance group	
Nuclear plant operations superintendent	1
Reactor operations engineer	1
Operation	
Plant operating supervisors	5
Shift engineers	5
Unit operators	15
Assistant unit operators	11
Workers and janitors	5
Maintenance	
Plant maintenance supervisor	1
Engineers	4
Engineering aids	2
Foremen	
Craftsmen	
Associated groups	
Health and safety	
Health physicist	1
Assistant	1
Technicians	5
Clerk-typist	1
Reservoir properties	
Public safety officers	10
Technical programme group [a/]	
Technical programme superintendent	1
Chief reactor physicist	1
Reactor physicists	2
Hazards control engineer	1
Electrical engineer (reactor control specialist)	1
Nuclear engineering trainees	5
Engineering analysis section	
Supervisor	1
Nuclear engineer	1
Mechanical engineer (thermal analyst)	1
Mechanical engineer (stress analyst)	1

Category	Number of persons
Reactor engineering section	
Supervisor	1
Mechanical engineer (tests, inspection, codes)	1
Mechanical engineers (tests and equipment)	3
Engineering aids	3
Experimental engineering section	
Supervisor	1
Nuclear development engineers	2
Controls engineering section	
Supervisor	1
Engineers	2
Instrumentation foreman	1
Instrumentation mechanics	
Chemical engineering section	
Supervisor	1
Engineers and chemists	3
Engineering aids	2
TOTAL	120

a/ Mostly for research purposes, because EGCR is an experimental plant.

270. The personnel of the reactor operation group have to receive extensive training before the reactor goes into operation. The operating supervisors should have previous experience of this type of work or should attend the course at Berkeley and should be trained for nine months in plant systems. Shift engineers need training at operating reactors, two months in nuclear technology and nine months of training in plant systems. Some of the unit operators get the same training as the shift engineers, others pass only through the two-months school and the nine-months training in plant systems. The assistant unit operators get also two-months school training in nuclear technology and nine-months training in plant systems.

Selected references

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

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March 31, 1961, ORNL-3102 (26 May 1961)
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December 31, 1961, ORNL-3254 (6 April 1962).

VII. THE HIGH TEMPERATURE GAS-COOLED REACTOR

General

272. The 40 MWe high temperature gas-cooled reactor (HTGR), also known as the Peach Bottom atomic power station, is an advanced type of reactor which employs a mixture of highly enriched uranium and thorium carbides dispersed in graphite which also serves as moderator and is cooled by helium. It is a prototype unit for a large (300 MWe) reactor. HTGR is a part of the third-round demonstration plants of USAEC and is intended to demonstrate the technical feasibility of a high temperature gas-cooled system with relatively high power density and capable of very high fuel burn-ups.

273. The reactor is being built at Peach Bottom in York County, Pennsylvania, 63 miles south-west of Philadelphia. It will be owned and operated by the Philadelphia Electric Power Company. This project is supported by the High Temperature Reactor Development Associates consisting of a group of 53 utilities. USAEC has contributed $14.50 million for research and development and an additional $2.50 million in fuel charge waiver. The prime contractor to the Philadelphia Electric Power Company for the construction of the plant is the Bechtel Corporation. The General Atomics Division of General Dynamics Corporation, as a sub-contractor to Bechtel, is furnishing the nuclear steam supply portion of the plant and, in addition, is carrying out the necessary research and development work supported by USAEC.

Special features

274. HTGR is an advanced gas-cooled system which attempts to combine the high power density of liquid-cooled reactors and the high temperature capability of gas-cooled systems with a very high fuel burn-up and excellent neutron economy. This concept is potentially very promising in spite of many challenges in design and materials which must be overcome. The special features of HTGR are discussed below.

275. Semi-homogeneous fuel. The heart of the reactor is a semi-homogeneous fuel element which consists of U-235 and thorium carbide particles dispersed in a graphite matrix. The cylindrical fuel compacts are housed in graphite sleeves and form a rugged assembly of 3.5" diameter which does not have to be built to great precision, thus reducing the fabrication costs.

276. Fission product control. Since the HTGR fuel has no metal cladding, it serves to improve heat transfer, neutron economy and high temperature operation, but introduces the problem of fission product control. In dealing with fission products a radical departure is made from the standard approach and their complete retention inside the fuel is no longer attempted. Instead, an internal fuel element venting system is used whereby the fission products escaping from the fuel are swept away by a continuous stream of helium and later trapped in suitable adsorbers to prevent them from contaminating the main coolant loop.

277. Neutron economy. The fuel elements have no cladding and graphite acts as moderator, fuel diluent, fuel cladding and core structure material. This accounts for the excellent neutron economy achieved by the reactor. The extensive use of graphite also contributes to a high heat capacity for the core and absorbs temperature swings.

278. High specific power. As compared to other gas-cooled reactors using metal-clad fuel elements where the power density ranges from 0.5 to 2 kW/l, HTGR has a high power density of 8 - 10 kW/l which is nearly 40% of that of some water-cooled power reactors. This high density is achieved because of a compact core with semi-homogeneous enriched fuel elements.

279. High burn-up. The expected fuel burn-up is about 48 000 MWd/t, which is limited only by reactivity considerations. For larger HTGR type units it may go as high as 80 000 MWd/t. The radiation damage to the core consisting mainly of graphite is not

significant. HTGR is a thorium converter, and 16% of the energy from the first core loading will come from U^{233} bred inside the fuel elements. On the whole, 0.55 average fissions per initial fissile atom are envisaged. The expected life of the core is about three years for this reactor and for bigger plants of the same type it may be five years.

280. High temperature. The reactor coolant core exit temperature is about 1380°F which produces 1000°F steam at 1450 psi, enabling the use of standard turbogenerators. This high temperature results from the use of helium as coolant and graphite as cladding and moderator, both of which can withstand very high temperatures and are chemically fully compatible.

Areas requiring research and development

281. The development of any new concept is accompanied by numerous problems which must be resolved. The incorporation of the above-mentioned novel features in HTGR poses many challenges for the physicists, metallurgists and engineers. An extensive programme of research and development is being undertaken by General Atomics to solve these basic problems. The extent of the research and development programme can be judged from the fact that the $14.50 million out of the grant from USAEC for the purpose has already been exhausted, and General Atomics is carrying on the work on its own. The main areas of investigation are enumerated below:

(a) Fuel element development. The semi-homogeneous fuel requires the development of a new technology for the fabrication of graphite matrix fuel compacts which should withstand intense irradiation at high temperatures. In recent irradiation tests, fuel capsules achieved 70 000 MWd/t burn-up without any damage or dimensional changes. The work on carbon coating of uranium and thorium carbide particles is also continuing;

(b) Fission product control. The fission product control by internal fuel element venting appears very attractive but it has yet to be proven. It is proposed to test the design of the fuel elements and purge flow arrangements in a pilot plant in an in-pile loop to ascertain its reliability and effectiveness. This part of the system is of crucial importance and the success of the HTGR concept hinges upon the effectiveness of fission product control;

(c) Nuclear design. The nuclear design of HTGR poses special problems because the lattice involved is quite different. Special attention is being given to the evaluation of thorium resonance absorption and its changes with temperature, calculation of reactor temperature coefficients, control rod work, study of neutron spectrum and its response to temperature changes, and quantitative analysis of thermal base effect. This work is being conducted by theoretical calculations and experiments with a critical assembly;

(d) Helium handling and purification. Helium is a difficult fluid to contain and the problem of its handling, pumping, valving and purification requires extensive developmental work. The aim is to reduce leakage to the very minimum and keep the level of impurity at 10 parts per million. Of special significance in the purification loop is the fission product trapping system which must work efficiently;

(e) Control. The reactor appears to be quite manoeuvrable due to the high thermal capacity of the core and the heat transfer coefficient that varies almost directly with the flow. The installation of thermo-couples inside the core offers real problems which are being overcome; and

(f) Material and component development. This reactor needs a great deal of graphite of an improved kind with relatively low permeability (about 10^{-3}) to volatile materials, resistant to radiation damage and of adequate mechanical strength. It appears that this high quality graphite can be supplied and fabricated. The components needing special attention are the control rod drives for which a mock-up has been installed and fuel handling equipment, including fuel charge and transfer machines which are being designed.

Important design features

282. Pressure vessel. The pressure vessel is 14 ft in diameter and 35.5 ft in height, with nominal wall thickness of 2.5". It is designed to withstand 450 psig at 725°F compared to the operating conditions of 350 psig and 634°F. It is made of grade B carbon steel to reduce costs. The vessel has a full diameter bolted top closure with a seal weld. There is one penetration in the side for the main coolant ducts, five penetrations in the top for fuel handling equipment, and 58 penetrations in the bottom for control rod drives, emergency shut-down rods and fission product purge lines. To protect the vessel from high temperature stresses, the incoming cold helium is baffled so that the vessel is essentially at 634°F instead of being at helium discharge temperature of 1354°F. The inlet temperature of helium was chosen primarily on considerations of radiation damage to graphite and the pressure vessel. At lower temperatures the radiation damage would be greater, and at higher temperatures the thermal stresses of carbon steel would necessitate the use of expensive stainless steel. Yet the temperature is high enough to prevent any significant damage to graphite.

283. Core. The core is cylindrical with a diameter of 9 ft and a height of 7.5 ft, excluding reflector, which is 2 ft thick. Apart from U^{235} and thorium, the core is essentially made of graphite which lends it a very high heat capacity. The entire core construction is homogeneous. It consists of about 804 fuel elements which are supported by a grid plate at the bottom. There is no upper grid plate for the top where the elements touch each other and the lateral support is provided by their leaning against each other and the side reflector. The fuel elements are arranged in equilateral triangular lattices with the in-between spaces for coolant passages. This configuration permits very large heat transfer surface per unit core volume. Heat flows radially from the fuel compacts by radiation and conduction to the outer sleeves from where it is removed by helium. It enters the core at 654°F and leaves it at 1380°F. The average fuel temperature ranges from 1700° to 3100°F. The maximum to average flux ratio is 1.5.

284. Heat transfer system. The choice of helium as the coolant with its good heat transfer properties, low mass and chemical compatibility with graphite permits very high reactor outlet temperatures. The system has two separate heat transfer loops which lend it extra reliability in case of pump or valve malfunctioning in one. Helium is circulated through the reactor at the rate of 440 000 lb/hr at 350 psig. Helium enters the vessel at 634°F and leaves the plenum at 1354°F.

285. Two horizontal single-stage centrifugal compressors circulate the helium through the core. One circulator is used in each of the two parallel loops so that the reactor continues to be cooled in the case of failure of one loop. Each compressor is designed to deliver 33 800 cfm of helium at 325 psig and 628°F. It is driven by an electric motor through a variable speed fluid coupling for flow regulation. An alternate source of power is available from a diesel-driven motor generator set for emergency purposes. Shaft sealing is obtained by helium-buffered, oil-flooded, floating bushing type seals. Similar seals have proved to be effective in conventional industrial plants and helium leakage is expected to be very low.

286. The helium coming out of the vessel is led to the steam generator through a heat valve and a total of 365 000 lb/hr of superheated steam at 1005°F and 1450 psig is produced. The feed water is fed to the steam generator of 425°F. In the steam generator, which is a vertical shell and tube type, special precautions are taken to prevent leakage of water or steam into the helium system. The most probable source of water in-leakage is at the tube-to-tube sheet valves. A baffle arrangement provides a continuous helium purge flow to flush out any steam in-leakages. Also, the leaking tubes can be plugged if necessary. The cold helium at 634°F is returned to the reactor through a concentric pipe surrounding the insulated helium outlet pipe. The important advantage of the concentric ducts is that the outer pressure bearing pipe is cool and the thermal stress problems at the penetrations of the reactor vessel and steam generator are considerably eased. The heat loss from the control pipe carrying hot helium to the surrounding cold helium is kept to a minimum by using proper insulation.

287. Shielding. The radial thermal shield consists of the graphite reflector 60 cm thick, thermal shield and pressure vessel, both representing 15 cm of steel and biological shield of ordinary concrete. Shielding problems are generally similar to those of gas-cooled reactors but some are peculiar to this system.

288. The helium coolant is expected to contain a fairly large concentration of fission products which will tend to settle on various components of the helium system. To permit access to these components for care and maintenance, adequate shielding has to be provided. The steam generator will be safe for approach 24 hours after reactor shut-down. The main helium circulator requires shielding because of deposition of fission products and need for occasional check-up. The helium purification system also has a high concentration of fission products.

289. The plant will be separated into five zones to meet the various accessibility and isolation requirements. The maximum permissible yearly radiation dose is 12 rem. The limiting values for radiation exposure in various zones are as follows:

		mrem/hr
Zone I	Unlimited access	not more than 1.0
Zone II	Limited access	not more than 2.5
Zone III	Controlled access	not more than 40
Zone IV	Area inaccessible during normal plant operation	over 40

The control room will be fully shielded for unlimited access.

290. Fission product control system. In dealing with the fission products in HTGR no attempt is made at their complete retention as would be the case with a clad fuel system. Instead, their release is minimized and those which do manage to escape are continuously swept away and prevented from contaminating the main coolant stream. The fuel elements of HTGR offer four barriers to the escape of fission products, namely fuel particle coating, fuel compact matrix, purge gas stream and low permeability graphite sleeve.

291. The carbide fuel particles are surrounded by 50 micron pyrolytic carbon which retains 99% of the volatile short half-life fission products while the others are partly retained in the high permeability graphite in which the fuel is dispersed. Thus about half of the non-volatile products become fixed in the graphite and the diffusion of others is greatly delayed. Those which escape are removed by helium current. The helium in the purge circuit enters through a porous graphite plug in the top of the fuel elements at the rate of 900 lb/hr or about 1.1lb/hr/fuel element, and goes vertically downward, passing through the recesses between the fuel compacts and the tube. The fission product bearing helium then passes through the internal trapping system where suitable adsorbers consisting of activated charcoal and silver retain most of the condensable fission products. Then it goes to the external trapping system where the remaining products are removed and the purified helium returns to the primary cooling circuit. The downward flow of the helium between the fuel compact and outer tube reduces the amount of fission products in this area, and thereby lowers their concentration and partial differential pressure, thus preventing their diffusion through the outer tube to the main coolant.

292. Helium purification system. The purification system has two main functions, namely to remove the non-radioactive impurities and the fission products.

293. The non-radioactive impurities result primarily from the in-leakage of water from the steam generator to the helium circuit. This in-leakage will be limited to 0.005 to 0.001 lb/hr. This water changes into CO and H_2 in passing over the carbon in the reactor. These gases are first oxidized and then CO_2 and water are removed from the system.

294. Most of the fission products which diffuse through the graphite fuel matrix are removed by the purge trapping system. The fission products which are to be removed by the purification system are volatile or noble gases, namely krypton and xenon. This system uses three stages of adsorbers which operate at successively lower temperatures so that as delay is provided the decay heat is reduced and it is possible to use the more expensive lower temperature refrigeration at the last stage and thus reduce the trap size and cost.

295. The purge stream containing 200 lb/hr of helium leaves the reactor at about 850°F and is fed to the first bed which cools the gas to 100°F, removes iodine which may pass through the internal trap and delays xenon-137. This bed is the smallest and most heavily shielded. It will be cooled by water under forced circulation. The second bed provides primary delay of krypton of 12 to 24 hours. The secondary delay of krypton of about three days is achieved in the third bed which is at a very low temperature. This reduces the activity in the primary system to about 10 curies.

296. Helium handling and storage system. The equipment and associated controls in the helium handling and storage system has five principal functions and accordingly provision is made for:

(a) Storing of helium in the system during refuelling or maintenance;

(b) Keeping the system pressure of 350 psi during transience or thermal cycling;

(c) Taking pressure surges after load swings or other unusual situation;

(d) Supplying the high pressure surge for the purified helium used in the purge gas stream; and

(e) Housing the system vent and purge lines.

297. Control of effluents. The reactor design assures that very little radioactive liquids are generated in the course of normal operation. The solid wastes generated are trapped in filters, spent demineralizers, resins, fission product adsorbers and are more easily dealt with. The gaseous effluents are the only ones which are of major consequence. Under worst conditions, the total activity in the coolant system is about 200 curies, which is not very much higher than the normal permissible dose of 35 curies. Assuming the system leak rate at 0.01 of 1/100 of the total volume per day, the resulting maximum dose from the activity release through the stack will be well below the safety limits.

298. Fuel handling. The refuelling is done 24 hours after shut-down when the core has cooled down appreciably and activity is reduced. The system pressure is lowered to below atmospheric so that all leaks are into the system. During refuelling, shut-down heat is removed by helium entering at 250°F and leaving at 450°F. The operation is remote controlled and the principal equipment consists of a charge machine, a fuel transfer machine, a spent-fuel canning machine and a fuel pick-up cell. Shielding plugs are removed from the vessel top and the charge and fuel transfer machines are secured tightly and sealed to the access ports. The fuel transfer machine is then lowered into the vessel. It can be rotated to any position in azimuth and can interchange fuel elements between positions in the core and the reflector. The fuel element to be removed is lifted by the fuel transfer machine by grasping its top knob. It is then parked just below the unloading port; the element is raised by a cable in the charge machine cask and rolled to a position over the spent-fuel canning machine. During this operation it is cooled only by natural convection and radiation. Due to its large heat capacity, its temperature is not expected to increase by more than 50°F. The cask is well shielded by 25 tons of lead. The canning machine encloses the spent-fuel element in a loosely fitting metal can before it is discharged into the spent-fuel storage area for cooling off. The new element is put into the pick-up cell and lowered into the vessel by the charge machine.

299. Reactor control. The reactor is controlled by 36 cylindrical control rods, each 80" long, inserted from the bottom and run in guide tubes. They are driven downwards to add and upwards to subtract reactivity. The 17" poison section is made of graphite

lead with boron carbide. Each rod is hydraulically operated and has a self-contained independent scram system. The specifications for each rod call for a duty life of 5×10^6 starts, 5×10^3 scrams, an initial upward acceleration of 100 ft per sec^2, maximum rod velocity of 10 ft/sec, and maximum response delay time of 0.075 sec. The maximum rod worth is $0.01 \frac{\Delta k}{k}$.

300. The main features of the control system are as follows:

(a) The k_{eff} of clean core at operating temperature with all rods out will not exceed 1.08. With equilibrium xenon and samarium, it will drop to 1.05 at the beginning of life;

(b) To avoid any troubles with control rods stuck in mid-position, only three control rods will be in partially inserted positions at any time, two of which will be automatically operated and one manually. All other rods will be completely in or out;

(c) The k_{eff} with all 36 rods in during shut-down conditions will be less than $0.95 \frac{\Delta k}{k}$ and, with one rod removed, less than $0.97 \frac{\Delta k}{k}$;

(d) Boron carbide burnable poison will be used for reactivity control. In addition, rhodium will also be used to give a strong negative temperature coefficient;

(e) 19 back-up shut-down poison rods will be available for emergency shut-down to make the reactor sub-critical under all conditions; and

(f) Another 19 fuse operated poison rods will be installed to drop automatically if the core temperature rises excessively.

301. The calculated effectiveness of the control rods has been checked by critical assembly measurements.

302. The average rate of reactivity change per rod at a speed of 0.06 ft/sec will be $0.8 \times 10^{-4} \frac{\Delta k}{k}$/sec with a maximum of $1.1 \times 10^{-4} \frac{\Delta k}{k}$/sec. The maximum for three rods will be $2.2 \times 10^{-4} \frac{\Delta k}{k}$/sec.

Safety

303. The safety of HTGR has raised many interesting technical questions and the reactor designers have been devoting special attention to this.

304. HTGR possesses several basic characteristics which make the system inherently safe. These are:

(a) A high prompt negative temperature coefficient of $2 \times 10^{-5} \frac{\Delta k}{k}$ per degree C, which, together with the high heat capacity of the core, serves to protect the reactor against reactivity and power transients;

(b) The stored energy of helium at 1380°F and 350 psi is low and a sudden break in the primary piping would not lead to release of excessive energy from the coolant as, for instance, in the case of water-cooled reactors;

(c) The fact that the side and bottom graphite reflectors and the pressure vessel itself are kept at essentially the temperature of the cold helium (634°F), these act as a heat sink for the dissipation of shut-down heat in case of loss of coolant;

(d) The choice of helium as the coolant makes the system chemically safe because it is an inert gas and fully compatible with graphite and steam;

(e) The Wigner stored energy and growth in graphite are expected to be insignificant because the temperature is high enough even in the reflector;

(f) The fuel compacts consisting of uranium and thorium carbides and graphite are very stable and, like ceramics, capable of withstanding high temperatures; and

(g) Xenon instability is not expected to be a problem in the operation of this reactor.

305. In addition to the aforementioned basic safety features which are inherent in the concept, the actual design of HTGR incorporates several provisions which give maximum degree of protection against any accidents resulting from equipment malfunction or failure. These include the following:

(a) The heat removal system has two independent loops so that in case of trouble in one the other will serve to cool the reactor. A separate shut-down feed pump is used to remove after-glow heat and this pump can also be used as a back-up at low power operation;

(b) Several independent sources of power supply are available including main turbogenerator, a 220 kV line, a 33 kV line, a diesel-driven generator, and a battery system. These assure operation of auxiliary systems in case of power failures;

(c) The helium circulators are electrically driven but have a back-up diesel-driven motor in case of emergency;

(d) Each control rod unit has its independent hydraulic drive and scram accumulator. The failure or leakage in one will not affect the other;

(e) 19 fuse operated safety rods have been provided which automatically drop into the core when the temperature becomes excessive;

(f) Reactivity additions due to control rod withdrawal will be limited to safe values;

(g) To protect the reactor vessel and the primary coolant system against overpressure, a system of relief valves and dump tanks has been provided;

(h) The fission product venting system minimizes the amount of activity in the main coolant stream under normal conditions. In case of any fuel element failure, contaminated helium can be bypassed through the fission product trap system to prevent build-up of activity in the primary loop; and

(i) Leakage of steam into the helium system in the steam generator will be kept to 0.001 lb/hr. Sensitive moisture detectors will be used to register water concentrations in helium of 10-100 ppm.

306. Analysis of maximum credible accident. The most severe accident postulated in the Preliminary Hazards Summary Report is not based upon any single incident but involves a series of failures. They are a primary system rupture with simultaneous loss of primary coolant and failure of the fission product purge line check valve to close, thus permitting back flow from the first three external fission product traps. The primary system rupture will cause a pressure build-up of 4 psig in the containment shell. This alone would, under inverse conditions, raise radiation doses at the boundary site during the first 24 hours as follows:

	rem
Whole body gamma dose from immersion	5×10^{-3}
Thyroid dose	5×10^{-2}
Bone dose	3×10^{-3}

307. Following this rupture, consequent depressurization and failure of the purge line check valve, the flow is reversed in the fuel element venting system and an additional release of activity takes place from the fuel element, internal traps and even external fission product traps. This activity can amount to about 2×10^{6} curies.

308. If, at this stage, the main coolant pumps also fail and only the emergency cooling is available, the core will heat up releasing more fission products from the fuel compacts

at a maximum temperature of 3600°F. Various fission products will be released at different rates. For instance, strontium and barium will have lowest retention; bromide, krypton, xenon and samarium, etc. will come out next; yttrium, ruthenium, lanthanum, etc. will be released slowly, and zirconium, niobium, molybdenum, etc. will not be released at all. The peak build-up will be reached after about 16 hours, when the activity reaches 5.2×10^6 curies and starts dropping gradually.

309. This release of activity inside the containment shell will, under inverse conditions, give rise to the following radiation doses in the first 24 hours:

	rem
Whole body gamma dose from immersion	4×10^{-1}
Thyroid dose	100
Bone dose	20

310. Appropriate evacuation procedures will be developed and suitable communication facilities will be installed in the plant so that speedy evacuation of all persons around the site can be carried out.

Containment

311. The containment shell is capsule-shaped with dished top and bottom and has a diameter of 100 ft and a height of 150 ft. It is made of steel. Since helium does not have much stored energy, it is designed for low internal pressure of 8 psig at about 150°F. The design pressure for the containment was calculated on the basis of the most severe postulated accident when the primary helium pipe break is followed by rupture of one steam generator tube, failure of helium loop valves to isolate the ruptured steam generator from the loop and loss of all forced circulation to the core. Immediate chemical reaction of 1100 lbs of water and steam with the reactor core is assumed. The highest peak containment pressure resulting from this multiple accident is 8 psig. The leak rate is less than 0.2% of volume per day. Considering the local conditions at the site, it is designed to withstand 100 mph winds and 30 lb/ft^2 of snow.

312. During reactor operation the containment shell will be provided with reduced oxygen atmosphere to limit the potential consequence of any accident involving the integrity of the primary loop. All penetrations will be leak-tight and automatically closing valves will be used. Personnel will have access through an air lock to the air room and an autoclave door at the operating floor elevation. The air compartment will be located inside the containment for the equipment requiring access during operation.

Fuel cycle

313. Objectives. The fuel element for HTGR must obviously be capable of operating at high temperature; i.e. higher temperatures than are possible with the fuel used in water-cooled reactors. Metals presently available for use as fuel cladding material are not well suited for temperatures of the order of 1400°F. The material chosen for fuel matrix and cladding as well as moderator was graphite. In order to avoid concentrated zones of heat production, the fissionable material was dispersed in a graphite matrix. This permitted the fuel elements to be relatively large in diameter, which provided added structural strength as well as large heat transfer surface. To avoid high central fuel temperatures, the core of the fuel cylinder contains only graphite, and no fissionable or fertile material. The fuel rods are the only materials in the active zone of the reactor core; neither additional moderator nor supporting structural members are required.

314. Fuel element design. The HTGR fuel elements are essentially graphite rods, with fuel semi-homogeneously dispersed in the rods. Graphite serves as the fuel cladding, matrix and moderator. Outwardly, these fuel elements have the appearance of solid graphite cylinders 3.5" in diameter with a total length of 12 ft.

315. The composition of the fuel cylinder might be described beginning with the smallest members. Fine particles of uranium carbide (highly enriched in U^{235}) and thorium carbide, 100 to 400 microns in diameter, are individually coated with pyrolytic graphite. This graphite coating, 50 to 60 microns in thickness, is essentially a primary cladding of the fuel, protecting the carbides from oxidation during fabrication and partially retaining fission products during reactor operation.

316. The coated fuel particles are uniformly dispersed in a graphite matrix, and formed into hollow cylinders of 2.75" outside diameter, 1.75" inside diameter and 1.5" long. These fuel pieces are slipped on to a 90" long solid graphite rod which serves as a central core. The above are then slipped into a graphite tube or sleeve, which fits rather snugly around the fuel pieces. The gap between the outer wall of the fuel and the inner wall of the sleeve is about 0.005" (or 0.01"). A number of longitudinal slots are provided in the fuel discs which permit flow of helium between fuel body and sleeve, downward to the bleed-off system. The sleeve is about 10 ft long. Graphite cylinders are joined to the top and bottom of the sleeve to serve as reflector zones. Also, in the lower end of the assembled fuel rod there is provided an internal fission product trap. This contains silver coated charcoal granules, which retain some of the fission products escaping from the fuel region.

317. The fuelled hollow cylinders alone contain about 20% by weight of thorium plus uranium, 80% by weight carbon. The graphite in the central core and in the sleeve contain no uranium or thorium. The uranium is 93% U^{235}. A full reactor loading of 804 fuel elements contains 184.8 kg of uranium (173.3 kg U^{235}) and 1987 kg of thorium. However, the fuelling is not identical in each of the 804 elements. In order to improve the flatness of the radial power distribution, the U^{235} content of the outer ring of fuel assemblies will be diluted to 60% of the concentration of the rest of the fuel assemblies, and the thorium load will be increased to about 130% in this outer ring. The decrease in fuel concentration at the edge of the core removes the power peak which would otherwise occur at the fuel-reflector interface due to the large absorption in U^{235} of cold neutrons which return to the core from the reflector. Hence, for the 108 fuel elements in the outer ring, the C/Th/U atom ratio is 3511/24.46/1 while for the remaining 696 fuel elements this ratio is 2126/9.57/1.

318. The initial fuel loading also contains poisons, namely 0.95 kg of boron and 5.0 kg of rhodium.

319. The fuel elements are free-standing in the reactor, with the lower end of the graphite element set into a stainless steel stand-off pin. The outside diameter of the graphite assembly is 3.5" and the assemblies are spaced on a 3.57" equilateral triangular spacing in the reactor. Therefore the fuel tubes are closely packed but do not touch each other over most of their length. However, the top section of each fuel tube is 3.57" outside diameter, so that the elements rest against each other. The outer fuel elements rest against the graphite reflector.

320. An extensive research development programme is under way for fuel element design. The results of tests conducted so far on fuel specimens have been very encouraging. One fuel element was exposed in the GAIL loop in GETR at San José, California, from September 1961 to February 1962. It achieved a burn-up of 8600 MWd/t. The fission product leakage characteristics were good. In the purge stream the activity was 500 mc/cm^2 under equilibrium condition. In the main stream it was 1 mc/cm^2 or lower than sensitivity of the instruments.

321. Another element was put in the loop in February 1962 and is expected to achieve 16 000 MWd/t exposure by September 1962. In this case fuel particles were coated and the permeability was increased from 10^{-5} to 10^{-3}. The activity observed in the purge stream amounted to 100 mc/cm^2 indicating the effectiveness of particle coating.

322. In fuel fabrication hot press technique (2000^{o}F) has been given up in favour of warm press (600^{o}F) method.

323. Fuel element fabrication. The fine particles of $U^{235} C_2$ and ThC_2, after being coated with pyrolytic graphite, are intimately mixed with graphite flour and a binder. The mixture is moulded to shape and graphitized at temperatures in excess of 2000^{o}C. The density of the product is about 1.9 g/cm^2.

324. The outer sleeve is fabricated of low permeability graphite, and is then further treated by impregnating with certain hydrocarbons followed by a regraphitizing. Dimensional tolerances are not very exacting, and machining is not required. For example, the outside diameter of the sleeve is $3.485 \pm 0.005''$.

325. The upper graphite reflector and the top of the sleeve are fitted together by a screwed and cemented joint. The lower graphite-to-graphite joint is screwed and brazed.

326. Fuel cost. The estimated fuel cost is as follows:

Table 17

The high temperature gas-cooled reactor: Fuel costs

Item		Mills/kWh
Fabrication		1.10
Transportation		0.03
Chemical processing and losses		0.27
Burn-up, less credit for U^{233}		1.22
Use charge		0.41
	TOTAL	3.03

327. In arriving at the above-mentioned figures, the cost of fabrication of the initial core is assumed to be $950 000. The fuel is assumed to have a life equivalent to 900 days at full power, which would mean an irradiation level of approximately 48 000 MWd/t plus thorium. U^{233} is credited at $11.88/g. The potential fuel cycle costs for a large (300 MWe) plant are assumed to be below 1 mill/kWh.

328. Fuel management. The present thinking calls for irradiating the initial fuel of the Peach Bottom reactor for a given period (perhaps three years) and then replacing the entire core, with no intervening shuffling or rearranging of fuel positions. For the second or later loadings, it may be decided to discharge and refuel a portion of the reactor at a time.

Plant cost estimate

329. The plant is being built under a fixed price contract by the Bechtel Engineering Corporation for $24.50 million. This does not include the fuel, fuel loading and start-up. The design of the nuclear steam portion is being done by the General Atomics under a sub-contract to Bechtel. USAEC is supporting research and development work by the General Atomics to the extent of $14.50 million. An additional $2.50 million will be allowed in fuel waiver charges. The High Temperature Reactor Development Associates will contribute $16.5 million towards the project with the idea of obtaining information on the development of HTGR concept. The Philadelphia Electric Power Company will pay $8.0 million, which is considered to be the equivalent cost of a conventional plant of 40 MWe size. It will also own and operate the plant.

Project schedule

330. The prime contractor for the HTGR plant, the Bechtel Corporation, is planning to complete construction in 27 months. This appears to be rather short but the company feels confident because of its extensive experience in the field and cites examples of its on-time completion of such reactors as Dresden. Since the plant is being built by the Bechtel Corporation on a fixed price ($24.5 million) contract, there appears to be strong incentive for the company to finish the work in as short a time as possible because each extra month of construction will cost it over $100 000 in overhead expenses only.

331. Prior to receiving the construction permit, the Preliminary Hazards Summary Report was submitted in August 1961. The public hearings took place in December 1961. The hearing examiner gave his approval for the plant in February 1962 on the basis of which USAEC issued the permit to commence construction.

332. According to the Bechtel Corporation, the proposed timetable for the HTGR project is as follows:

Table 18

The high temperature gas-cooled reactor: Time schedule for the project

Item	Proposed
Start of construction	March 1962
Containment shell erection	
Start	July 1963
Finish	November 1963
Turbogenerator installed	November 1963
Construction complete	June 1964
Pre-operational tests	June-September 1964
Initial criticality	September-October 1964
Power operation	December 1964

Selected references

333. A list of selected references concerning the high temperature gas-cooled reactor is given below:

Peach Bottom Atomic Power Station, Preliminary Hazards Summary Report, Philadelphia Electric Co., Philadelphia, Pa. (August 1961)

BOSWORTH, G.H., "HTGR to have Carbon-Steel Vessel", Electrical World, v. 156, New York, N.Y. (4 December 1961) p. 75-78

Proceedings of the High Temperature Gas Cooled Civilian Power Reactor Conference, TID-7611, USAEC, Washington, D.C. (January 1961)

HOFFMANN, F. and FORTESCUE, P., "Application of high-temperature gas cooling to nuclear power plants: the HTGR", Small and Medium Power Reactors, Vol. I, STI/PUB/30, IAEA, Vienna (1961) p. 441

C. CANADA

Background information

334. Ever since the start of their nuclear power programme the Canadian authorities have consistently pursued the development of the heavy-water-moderated, natural-uranium-fuelled reactors. The first critical facility, ZEEP, was started in 1945, and was followed by two high flux reactors NRX (1947) and NRU (1957). These research and test reactors logically led to the first small-scale (20 MWe) reactor power station NPD, which was completed in June 1962, thus being the first D_2O-natural uranium reactor to produce electric power for a grid. Experience gained from the design and construction of NPD is now being directly used in the first full-scale power station project, the 200 MWe CANDU, which will be completed in 1964 at Douglas Point on Lake Huron, Ontario.

335. This line of approach has been taken because abundant ore deposits make natural uranium a cheap fuel in Canada, a situation shared by several countries, also lacking enrichment facilities. The fuel cycle appears to be simple with "once-through" fuel elements that are used to a high burn-up and then discarded and stored indefinitely in simple and cheap facilities. This fuel cycle concept has shown economic promise with a possibility for very low share of the generating cost attributable to the fuel, when mass production techniques can be used in the element manufacturing.

336. The fuel cycle concept is one of the great advantages of this reactor type as it avoids the complications of hot-fuel transport and reprocessing. The D_2O moderator, of course, in itself offers a system with excellent neutron economy. Detailed economic studies for large reactor station units have also indicated low generating costs in spite of the high capital cost of the D_2O moderator and primary coolant.

337. The NPD as well as the CANDU design is based on a pressure tube reactor instead of a pressure vessel. This has certain obvious advantages. On-load refuelling can be accomplished from both sides of the core in a straightforward way. Fuel cladding failures can be located rapidly. The pressure tube design would avoid penetration through a pressure vessel wall for instance by control rods.

338. These advantages for the NPD reactor are, naturally, offset by some disadvantages. The high cost of heavy water necessitates expensive safeguards to avoid excessive D_2O leakage and deterioration. In a pressure tube reactor, neutron economy also requires very careful design and places high demands on engineering experience as well as supporting industrial facilities for development and construction. Coolant void and temperature coefficients may be positive under transient operating conditions which makes safety studies of the reactor system necessary and may pose special requirements on the instrumentation and shut-down systems.

339. D_2O-moderated power stations of small size may be at an economic disadvantage as compared to some other reactor types. However, future developments may bring forth considerable cost reductions and studies to this end are being performed in Canada. Use of organic coolant instead of heavy water may, for instance, result in lower costs because of the lower D_2O inventory and a more conventional primary coolant loop in spite of the relatively poorer neutron economy. For units smaller than 25 MWe a slight enrichment to decrease the core size may also improve the economy of the plant.

VIII. THE NUCLEAR POWER DEMONSTRATION PROJECT

General

340. The nuclear power demonstration (NPD) is a heavy-water-moderated and -cooled reactor with a net station electrical output of 19.5 MW(e). Its purpose is to prove the technical feasibility of the heavy water power station concept and to serve as a guide in the design of the 200 MW(e) Douglas Point power station. It has never been anticipated that NPD generated power would be competitive or even of low cost. Detailed cost estimates have thus not been made and these figures are generally not available. Total capital cost of the station is approximately $33 000 000.

341. The NPD station is a joint project between AECL, the Hydro-Electric Power Commission of Ontario and Canadian General Electric Co. Ltd. The plant is situated on the Ottawa River about 15 miles northwest of Chalk River, half a mile from the hydro plant of Des Joachims and half a mile from the town of Rolphton.

342. The time schedule for the project is given in Table 19.

Table 19

The nuclear power demonstration reactor: Time schedule for the project

Item	Actual
Start of construction	1956
Switch to pressure tube design	1957[a]
Start of operator on-site training	1960
Reactor vessel arrival	June 1961
Start of cold testing	Sept. 1961
Construction complete	Dec. 1961
Fuel loading complete	27 March 1962
Initial criticality	11 April 1962
3 MW distributed to grid	4 June 1962
10 MW distributed to grid	10 June 1962
Full power operation	28 June 1962

a/ Switch from pressure vessel to pressure tube design caused a delay between 1956 and late 1958.

Special design features

343. A summary of important design data is given in Annex IV.

344. The original design (now often referred to as NPD-1) was based on a pressure-vessel-type reactor. A study of a full-scale project in 1956/57 features a pressure tube design based on the experience then already gained with zirconium alloys. In 1957 it was decided to adopt this concept also for the NPD station to try the full-scale station design ideas in as much detail as possible.

345. Core. One hundred and thirty-two horizontal pressure tubes made from Zircaloy-2 contain the fuel elements and the pressurized heavy water coolant. The fuel is natural uranium dioxide contained in 1188 fuel elements. The moderator is unpressurized heavy water with a nominal temperature range of 49-82^{o}C. The core has a tapered (55-14 cm) heavy water reflector, in its turn surrounded by an additional light water layer, acting as reflector, and fast neutron shield.

346. Reactor vessel and pressure tubes. The vessel is a horizontal aluminium cylinder, double-walled to separate the light and heavy water reflectors. The Zircaloy-2 pressure tubes are expansion rolled in grooved stainless steel end fittings.

347. Reactor process system. A closed primary loop transfers heat to a steam generator from which light water steam in the secondary system loop drives the turbine. The piping in the primary system is made from carbon steel. The heavy water pH is maintained high and inhibitors are used to minimize corrosion.

348. Three primary pumps with double shaft seals are installed, but full flow can be given by two pumps.

349. In the secondary systems dry, saturated steam of 232°C and 28 kg/cm^2 is carried to the turbine, which has a maximum rating of 22 MW.

350. The main condenser is bolted directly to the turbine exhaust. Half of it can be isolated and cleaned when the plant is operating on a reduced load.

351. A reject condenser permits running the reactor on a reduced power level even if the turbine system has had to be shut down completely.

352. Heavy water losses. The NPD operating budget allows for a loss of heavy water of 2-4% of the total inventory per year (5-10 kg/day). The NPD reactor vault is designed as a sealed-vapour barrier with no outward flow of liquid or vapour and with the air maintained dry. Liquid leakages are collected in sumps and an air cooler condenses vapour.

353. This system was tried out during the start-up operations and it was actually found that the losses could be kept down to about one quarter of the first year's budget allowance of 4% of the total. Losses generally occurred in specific events rather than in the form of small continuous leakages.

Safety and control

354. The reactor power level is controlled exclusively by the moderator level and there are no trim or safety rods. A helium gas pressure balance system between the reactor vessel and a moderator dump tank is used both for fine adjustment of the moderator level and for dumping all heavy water in a scram.

355. In addition to the control provided by changes in the moderator level, change of the moderator temperature between the normal operating limits of 49°C and 82°C will control another 0.36% $\frac{\Delta k}{k}$ to override xenon poisoning or fuel burn-up reactivity changes. With equilibrium fuel the moderator temperature coefficient is essentially zero.

356. A booster rod, consisting of an enriched fuel element, can be inserted into the core to add another 0.25% $\frac{\Delta k}{k}$ for xenon override.

357. With these contingencies the reactor can be brought to power against xenon poisoning within 45 minutes after a scram.

358. The safety system operates on six valves in three helium dump-lines connecting the reactor vessel with the moderator dump tank. The measuring channels are always triplicated and connected so that two signals out of three are needed for a scram. When the dump-line valves are opened the gas pressure difference between the reactor vessel and the tank is removed and the moderating heavy water is dumped from the vessel. Within one second from a scram signal reactivity has been reduced by 0.5% $\frac{\Delta k}{k}$. After four seconds reactivity decreases by 4% $\frac{\Delta k}{k}$/sec.

359. There are several safety features inherent in the design, such as a strong negative fuel temperature coefficient and the sintered UO_2 fuel material, which is reasonably retentive of gaseous fission products, has a high melting point and does not react with high temperature water.

360. The coolant void coefficient is positive but the shut-down rate in a scram is considered large enough to offset this.

Fuel and fuel handling

361. The fuel cycle is distinguished by two especially important features: the on-load fuelling system, and the fundamental concept of not reprocessing any of the fuel elements. The natural uranium fuel is considered as a cheap material and the only aim is to push the burn-up to the highest possible figure, after which the elements are discarded and left in storage on the reactor site.

362. Fuel elements. The elements are of the bundle type made up of rods. Each rod consists of UO_2 sintered pellets clad in Zircaloy-2. There are two types of elements, one 7-rod bundle used in the periphery and one 19-rod bundle used in the central parts, to avoid thermal overloading of the elements.

363. The maximum fuel rating of 4 kW/cm length of 19-rod bundle is conservative and an increase up to 8.9 kW/cm is considered permissible.

364. It is expected to run the NPD elements to an average burn-up of 6000 MWd/t, reactivity being the limitation. Tests have shown 9000 - 10 000 MWd/t not to give any fuel deterioration.

365. Refuelling. In normal operation three elements will be exchanged every second day. This is performed by two identical fuelling machines working on each side of a pressure tube, controlled by push-button manoeuvres in the control room. The machine picks out the appropriate tube, aligns automatically with the end fitting and seals against it. It then pressurizes itself, removes the end cap, exchanges a fuel element bundle, replaces the end cap, depressurizes and drains. Fuel exchange is made simply by pushing in a new element from one side, removing a spent one from the other.

366. This feature permits fuelling adjacent channels from the opposite sides to increase the possibilities for high and even burn-up in individual elements.

367. Spent fuel is carried away for storage under water in a storage bay close by the reactor.

Operating personnel and training

368. NPD's present staffing plan is shown in Table 20.

Table 20

The nuclear power demonstration reactor: Staffing plan

Category	Number of persons
Administration	
Plant superintendent	1
Assistant plant superintendent	1
Operation	
Shift supervisors[a]	5
Operators	22
Radiation inspector	1
Maintenance	
Supervisor	1
Instrumentation	3
Mechanical and electrical	7
Service	4
Technical	
Fuel engineer	1
Chemical unit	2
Radiation protection officer	1
General	
Training	3
Clerical unit	7
TOTAL	59

a/ Each shift supervisor managing a shift has three operators and five trainee operators.

369. For the key people a requirement of at least one year's training or previous experience in nuclear reactor operation was set up. Thermal power plant training was given by the Hydro-Electric Power Commission of Ontario, and reactor training by AECL at Chalk River. On-site training was given with the assistance of the Canadian General Electric Co. Ltd. consisting of courses, preparation of descriptive, operating and testing manuals and participation in the testing programme on the reactor.

370. There is no radiation protection officer on the shifts. Radiation control is obtained by means of stationary monitors and access interlock systems.

Construction and operating experience

371. Primary D_2O system. With the primary system piping made of carbon steel, crud deposition in the fuel elements must be avoided by careful conditioning of the system surfaces. A certain degree of cleanliness and dryness had been sought for during the installation so the actual system cleaning operations could be kept to a minimum. The conditioning was performed by heavy water.

372. In the pressure tests some leaks were found, for instance, in the pumps, but they were easily corrected. The system temperature was raised to 185°C, using primary pump losses as heat source. A very low crud level in the water of <0.05 ppm was reached within three days and in general cleanliness requirements for the primary system were very well met in short time.

373. Fuel loading. The entire first loading with 1188 elements was performed in 19 days, using the fuelling machines, corresponding to almost three years of refuelling operations. The experience of the fuelling method is generally very positive but the machines have as yet been tried out at full operating pressure and temperatures only on a test loop. The first loading used some elements with depleted uranium to avoid too high excess reactivity in the core.

374. Station start-up. When the station is off the line it is preferable to keep the reactor critical but at low power (0.1%), and the primary system at nominal pressure. The temperature of the system may be either cold or hot, but accordingly start-up will be different.

375. A cold start requires a two hour interval to bring the primary system temperature up to normal values. The limiting rate, 110°C/hr, is imposed by thermal stress considerations of the major components. The reactor can be brought to 6% of full power to establish this heating rate. At the same time steam can be generated to warm the secondary system and the turbine can be rolled on steam for initial warm-up. From a cold start the plant can be on load, presenting 50% power, in about two and a half hours.

376. A warm start with the primary system at 150°C can be effected in less than 100 minutes. From a complete shut-down, the reactor can be brought to 6% of full power in 30 minutes and can raise steam to 27 kg/cm^2 in another 40 minutes. The turbine can then be loaded at its normal rate.

377. The minimum availability factor will be 90%, limited upwards by normal maintenance especially on the D_2O pump shaft seals. For a bigger station with the conventional equipment multiplied to a higher degree, the on-load refuelling should give very high load and availability factors.

378. No major troubles were encountered during the first operational try-outs with NPD. Some heavy water leaks and cavitation in the moderator pumps were found but could be corrected. When the turbine was run at full speed vibrations were caused by bad adjustment of the alternator shafts, but this was easily rectified. The whole start-up programme has been extremely smooth and the reactor produced power a remarkably short time after construction was completed.

Studies of up-rated NPD plants

379. The NPD reactor was designed and built expressly as a pilot plant for future full-scale power stations. Some studies have been performed on only slightly changed but up-rated NPD stations. In the following only a brief summary will be given of these studies together with the cost estimates available for them.

380. To obtain a quick reassessment of possible power costs from an NPD-type station, AECL has performed a study of an NPD, up-rated to 50 MW(e), assuming only such changes as were necessary to operate at high power but with the reactor core otherwise unchanged. The total plant cost was estimated at $29 000 000 with a power cost of 10.05 mills/kWh assuming 6.3% fixed annual charge (approximately 4.5% interest rate).

381. The Canadian General Electric Co. Ltd. has performed a study after the initial start-up of NPD, introducing some definite improvements in the reactor:

(a) Up-rating the fuel to 7.8 kW/cm using 19 rod elements everywhere;

(b) Radial flux flattening by means of a two-zone core;

(c) Increasing the number of fuel channels from 132 to 164; and

(d) Improving the thermal cycle efficiency.

This results in a power station with a net output of 73 MW(e) and an over-all net efficiency of 27.7%. The cost for power from this station would be 8.7 mills/kWh under roughly the same assumption as used for the 50 MW(e) station.

Selected references

382. A list of selected references concerning the nuclear power demonstration reactor is given below:

NPD-2 Design Description - Canada's first nuclear power station, AECL-952, Canadian General Electric Co. Ltd., Toronto (1959)

LEWIS, W.B., "Competitive Nuclear Power for Canada", Nucleonics, v. 18, No. 10, New York, N.Y. (October 1960), p. 54

MELVIN, J.G., "Nuclear power in Canada", Small and Medium Power Reactors, Vol. I, STI/PUB/30, IAEA, Vienna (1961), p. 127

LAURENCE, C.C., "Potential of the use of heavy water in power reactors", Small and Medium Power Reactors, Vol. I, STI/PUB/30, IAEA, Vienna (1961), p. 139

OLSEN, J.L., A Description of NPD - Canada's first nuclear power reactor, AECL-1318 (Paper 1), Canadian General Electric Co. Ltd., Toronto (1962)

A Report on Start-up Operation at the NPD Generating Station and Prospects for an Up-rated NPD Station, Canadian General Electric Co. Ltd., Toronto (June 1962)

ANNEX I

Important design features of the Hallam nuclear power facility

Location	Hallam, Nebraska
Owner/Operator	USAEC/CPPD
Type	graphite-moderated and sodium-cooled reactor
Power	
Gross thermal	240 MW (nominal); 256 MW (design)
Electrical	gross: 82 MW; net: 75 MW
Over-all efficiency	31.6%
Fuel element	
Type	slugs forming a rod
Fuel	enrichment: 3.6%, uranium metal alloyed with 10 wt% Mo
Core	
Dimensions	13 ft diameter; 13.25 ft high
Number of fuel elements	137 (expected initial core loading)
Power density	4.8 kW/1
Reactor vessel	19 ft outside diameter; 33 ft high; stainless steel
Control rods	
Type	tubular rods with gadolinium and samarium oxides
Number	19
Containment	primary systems enclosed in concrete shielded, carbon-steel-lined cells
Turbine steam conditions	
Temperature	825°F (nominal); 843°F (design)
Pressure	800 psig (nominal and design)
Mass flow rate	710 000 lb/hr (nominal); 752 000 lb/hr (design)
Construction schedule	
Start of construction	April 1959
Reactor critical (dry)	January 1962
Full power operation	April 1963
Costs	$55.5 million

ANNEX II

Important design features of the experimental gas-cooled reactor

Location	Oak Ridge, Tennessee
Owner/Operator	USAEC/TVA
Type	graphite-moderated, helium-cooled reactor, using slightly enriched uranium dioxide
Power	
Gross thermal	85 MW
Electrical	gross: 29.5 MW; net: 21 MW
Over-all efficiency	26.3%
Fuel element	
Type	7-rod cluster contained in graphite sleeve
Fuel	UO_2, stainless steel clad; enrichment 2.46%
Core	
Dimensions	11 ft 10 in. diameter; 14 ft 6 in. high
Number of fuel elements	1404 (max.) in 234 channels
Power density	1.97 kW/l
Pressure vessel	20 ft diameter; 46 ft high; carbon steel cylinder with hemispherical ends
Control rods	
Type	tubular rods of boron carbide
Number	21
Containment shell	112 ft diameter; 216 ft over-all height; carbon steel plates
Turbine steam conditions	
Temperature	900°F
Pressure	1300 psig
Mass flow rate	250 000 lb/hr
Construction schedule	
Start of construction	August 1959
Reactor critical	second half 1963

ANNEX III

Important design features of the high temperature gas-cooled reactor

Location	Peach Bottom, Pennsylvania
Owner/Operator	Philadelphia Electric Power Company
Type	graphite-moderated, helium-cooled reactor
Power	
Gross thermal	115 MW
Electrical	gross: 46 MW; net: 40 MW
Over-all efficiency	34.7%
Fuel element	
Type	hollow fuel cylinders slipped on graphite rod and contained in graphite tube
Fuel	dicarbide particles of uranium (93% enriched) and thorium, coated with pyrolytic graphite, dispersed in graphite matrix
Core	
Dimensions	9 ft diameter; 7.5 ft high
Number of fuel elements	about 804
Power density	8.4 kW/l
Pressure vessel	14 ft inside diameter; 35.5 ft over-all height; 2.5" carbon steel
Control rods	
Type	cylindrical rods of boron carbide operating in guide tubes
Number	36
Containment shell	100 ft diameter; 150 ft high; steel plates
Turbine steam conditions	
Temperature	1000°F
Pressure	1450 psig
Mass flow rate	365 500 lb/hr
Construction schedule	
Start of construction	February 1962
Reactor critical	middle of 1964
Full power operation	late 1964
Costs	$24.5 million

ANNEX IV

Important design features of the nuclear power demonstration project

Location	near Des Joachims, Ontario, Canada
Owner/Operator	AECL/Hydro-Electric Power Commission of Ontario
Type	heavy-water-moderated and -cooled pressure tube reactor
Power	
Gross thermal	83.3 MW
Electrical	gross: 22 MW; net transformer output: 19.5 MW
Over-all efficiency	23.3%
Fuel element	
Type	bundles with 7 or 19 rods per bundle
Fuel	sintered natural uranium dioxide pellets in Zircaloy-2 cladding
Reactor vessel	horizontal double-walled aluminium cylinder
Pressure tubes	Zircaloy-2; 8.25 cm inner diameter
Core	
Number of fuel elements	1188 elements in 132 tubes
Fuel burn-up	6000 MWd/t
Uranium weight	15 800 kg UO_2
Heavy water inventory	61 000 kg (15 700 of which in primary system)
Control	moderator level; one booster rod to help overcome xenon poisoning

FIGURE 1

THE HALLAM NUCLEAR POWER FACILITY (HNPF)

Flow diagram

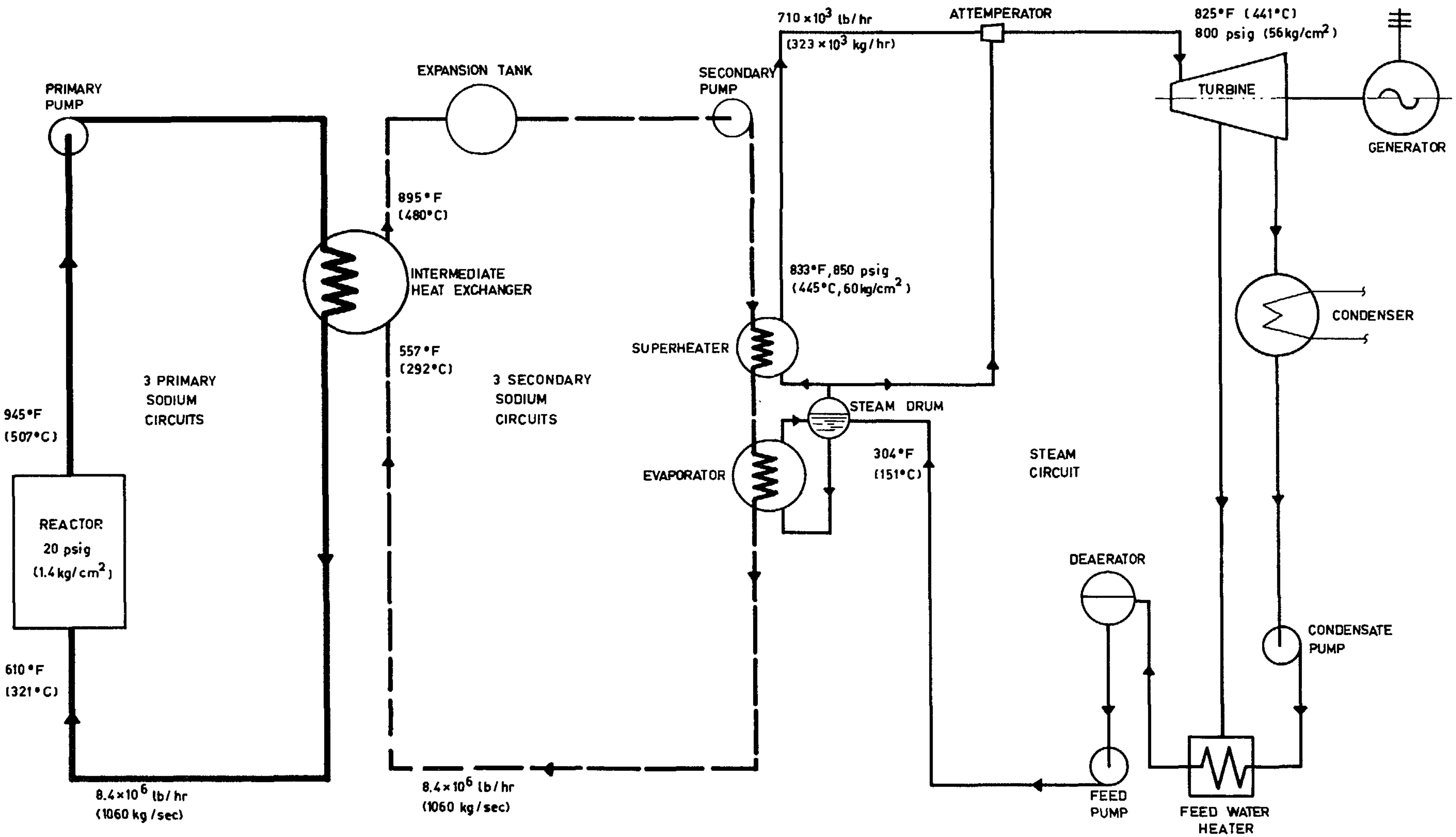

FIGURE 2

THE EXPERIMENTAL GAS-COOLED REACTOR (EGCR)

Flow diagram

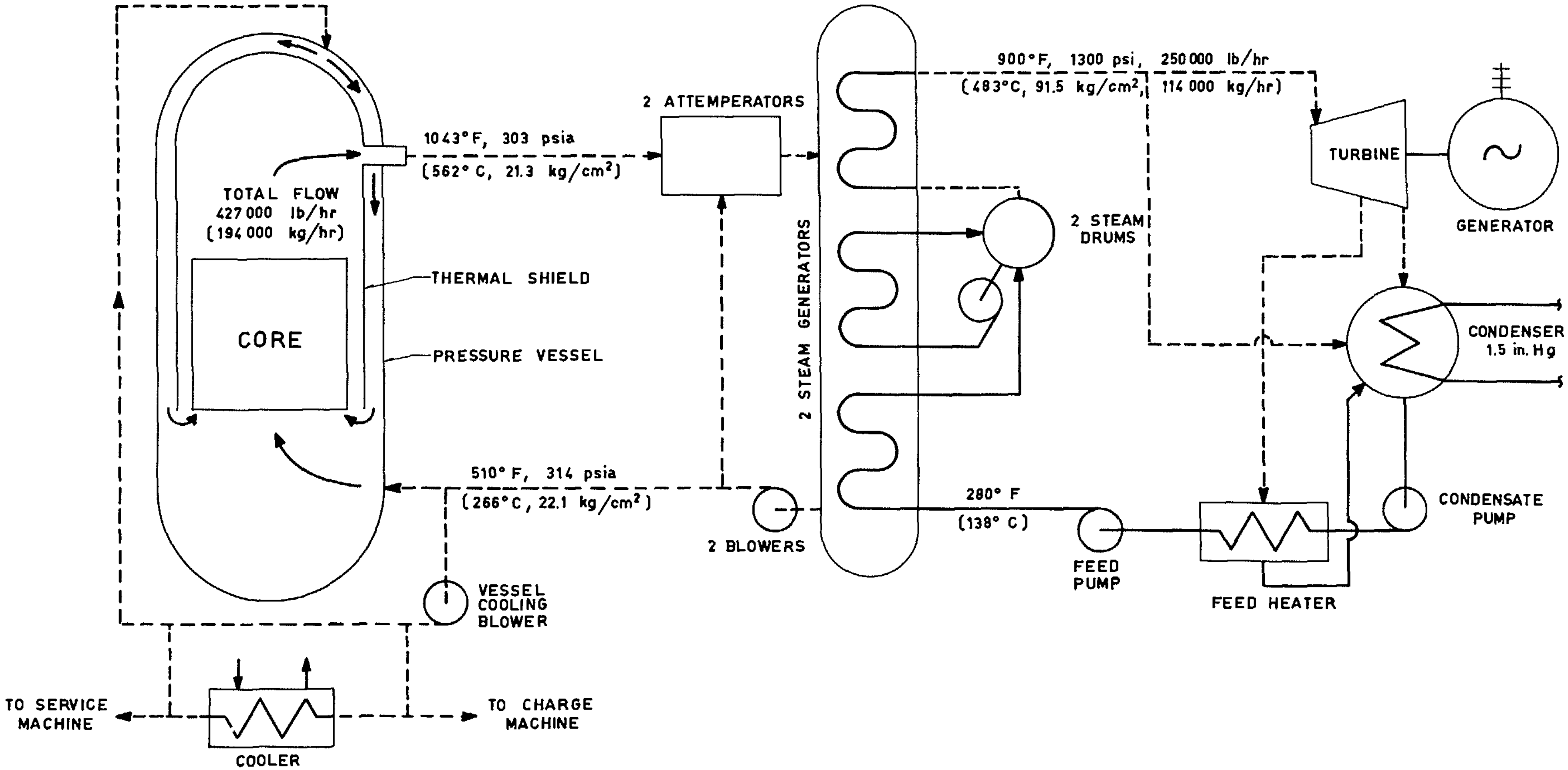

FIGURE 3

THE HIGH TEMPERATURE GAS-COOLED REACTOR (HTGR)

Flow diagram

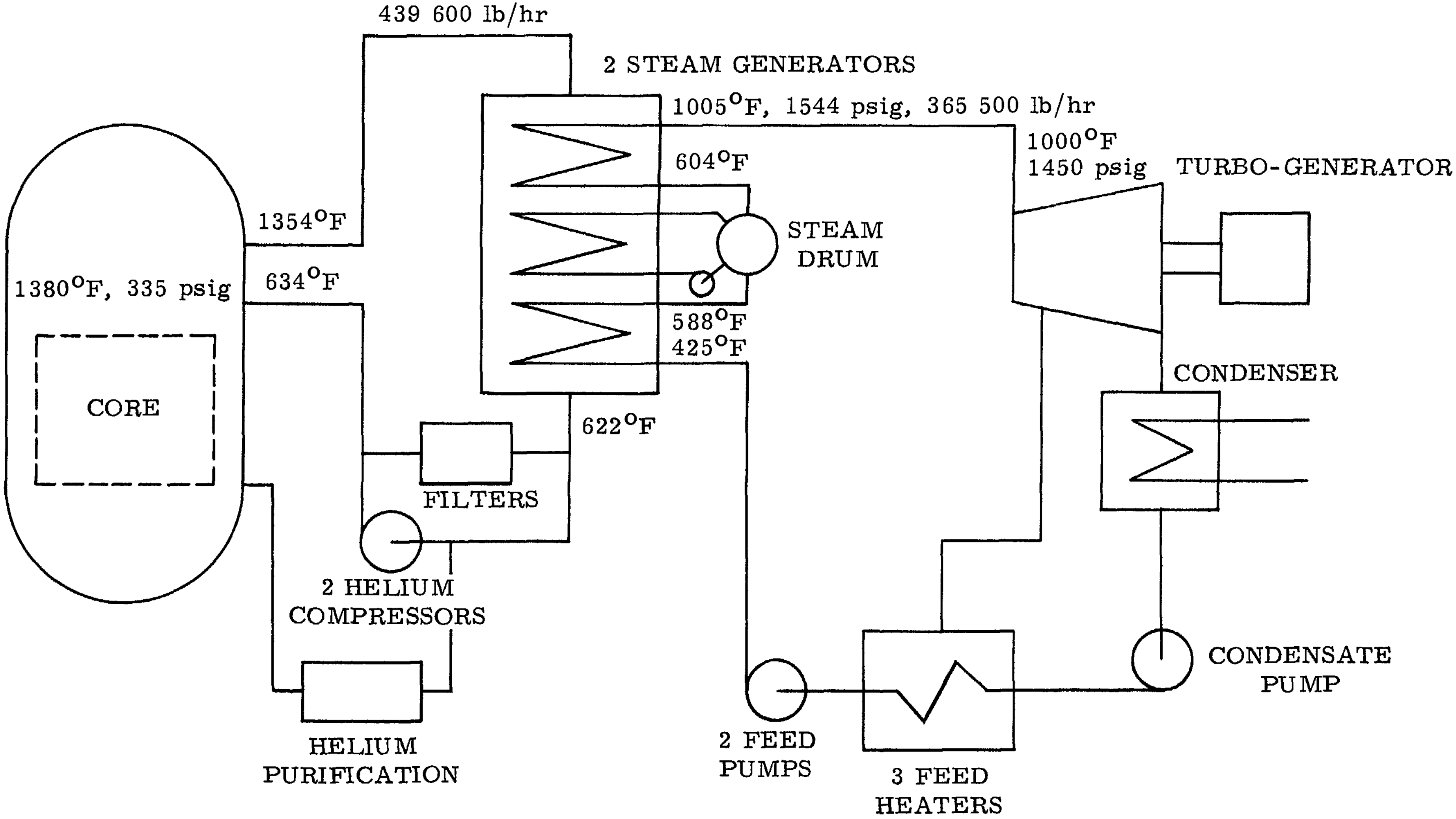

FIGURE 4

THE NUCLEAR POWER DEMONSTRATION PROJECT (NPD)

Flow diagram