ANL-6395

Argonne National Laboratory

U.S. PARTICIPATION IN THE OEEC HALDEN REACTOR PROJECT May 1959 to September 1960

by

Leonard W. Fromm, Jr.

ANL-6395
General, Miscellaneous,
and Progress Reports
(TID-4500, 16th Ed.,
Amended)
AEC Research and
Development Report

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois

U.S. PARTICIPATION IN THE OEEC HALDEN REACTOR PROJECT May 1959 to September 1960

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Reactor Engineering Division

Written
December 1960

Operated by The University of Chicago
under
Contract W-31-109-eng-38

TABLE OF CONTENTS

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

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ABSTRACT

The Halden Boiling Water Reactor, located in Halden, Norway, is an experimental process steam producer using D_2O as moderator and coolant. It is operated under an international agreement through which thirteen western European countries contribute funds and professional staff. The United States participates in the project by assigning USAEC laboratory personnel to work with the project staff for extended periods. From May 21, 1959 to September 22, 1960, the author served as General Consultant to the Halden Project Management, on assignment from Argonne National Laboratory.

The reactor was first brought to criticality on June 29, 1959. A low-power, low-temperature physics program with two core loadings has been completed. Mathematical models of the reactor and plant have been developed and used in an analog computer study of dynamic behavior. Engineering activities have included design and procurement of a second core of Zircaloy-2-clad, 1.5% enriched UO_2 to replace the first core of aluminum-clad, natural uranium metal; system modifications for 20 Mw operation; a proposed replacement of the reactor vessel with a design better suited for future programs; correction of low-temperature pitting corrosion of first core fuel elements; and improvement of the cavern containment. An initial first-core power run with boiling at 2 Mw was made on October 5, 1960.

The benefits of U.S. participation in the Halden Project include: (1) the obtaining of detailed advance information on the boiling D_2O reactor, thus avoiding the need for building an experimental reactor of this type in the U.S. or diverting a facility such as EBWR from other important programs; (2) the opportunity to maintain close contact with atomic energy programs in other European countries through association with foreign staff members and with foreign representatives to the Project's administering committees; and

(3) the political benefit of symbolizing an American attitude of active cooperation with other countries in matters of common interest.

The author strongly recommends further participation in the Halden Project at both the staff and administering committee levels, as well as exploration of other European projects to determine further areas for mutually profitable cooperative efforts.

I. INTRODUCTION

This report summarizes the activities of the OEEC Halden Reactor Project, Halden, Norway, during the period of participation in the project by the author, from May 21, 1959 to September 22, 1960. Previous U.S. participants in the project have been H. D. Brown of E. I. DuPont de Nemours & Company Savannah River Laboratory, and J. A. DeShong, Jr., of Argonne National Laboratory.

The author was requested by the Norwegian Institutt for Atomenergi, operators of the project, to serve as General Consultant to the Project Management. The assignment covered a very wide range of activities, and consequently no attempt is made in this report to identify all of the specific contributions made. Rather, a general description of the accomplishments of the entire project is given, and the author's work is pointed out only in areas of major effort or special significance.

The Halden reactor facility is currently operated under an international agreement involving thirteen European countries and the United States. The relationships between the countries under this arrangement are most important to a proper understanding of project accomplishments and the significance of U.S. participation. For this reason, a discussion of the history and organization of the project, as well as a brief description of the reactor facility, are included.

The review of the low-power, low-temperature physics program, presented in Section IV-B, is extracted from a paper, "The Halden Project Experimental Program," by H. Ager-Hanssen and V. O. Eriksen, presented at the European Atomic Energy Society Enlarged Symposium on Heterogeneous Heavy Water Power Reactors, Mallorca, Spain, October 10-15, 1960. The paper is to be published as Halden Project Report HPR-7. Details of all results reported have been published in Halden Project Reports HPR-3, -4, and -6. The review is presented here in the interest of completeness of this report, in an effort to provide an impression of the character of research being carried out at Halden.

References are made throughout the report to Halden Project Reports (HPR-) and Halden Internal Reports (HIR-), by means of footnotes. A complete list of these reports is provided in Appendix B. Other references are numbered in the conventional manner. Most of the figures used in this report are either traced or reproduced directly from Halden Project documents.

The metric units used by the Halden Project are generally familiar, and in most cases equivalents are given in English units. The abbreviations for metric units of pressure are perhaps not so widely known, and are explained below for clarity.

Metric Abbreviation	Meaning	Corresponding English Unit Abbreviation
ata	atmospheres absolute	psia
ato	atmospheres gage	psig
atm	atmospheres without reference, as for pressure-drop data	psi

II. HISTORY AND ORGANIZATION OF THE OEEC HALDEN REACTOR PROJECT

A. Early Norwegian Atomic Energy Research

Research in atomic energy began in Norway in the late 1940's with the establishment of the Institutt for Atomenergi as an outgrowth of the Norwegian Defense Research Establishment. The Institutt is located adjacent to the Defense Establishment at Kjeller, near the town of Lillestrøm, about 25 km (15 miles) east of Oslo. It is organized as a private company funded largely by the government but also supported to some extent by Norwegian industry and educational institutions.

The Institutt took on an international character early in its life, since at that time it was perhaps the only establishment in the world carrying on experimental work in atomic energy on a completely unclassified basis. As such, it attracted staff members from all corners of the globe.

The early Norwegian interest in atomic energy was based largely upon the attitude of the government that the country should keep abreast of new technological developments to as large an extent as possible within its limited means. After an initial period of orientation, the program of the Institutt was directed more specifically toward the development of heavy water-moderated, natural uranium-fueled reactors, primarily because of the availability of heavy water in Norway, the general availability of natural uranium, and the unavailability of enriched uranium. The Institutt joined with the Dutch in this interest, with the formation of a Joint Establishment for Nuclear Energy Research (JENER). The first major accomplishment of this group was the completion, in 1951, of the JEEP (Joint Establishment Experimantal Pile) research reactor at Kjeller. The JEEP is a 450-kw tank-type reactor similar, in many respects, to the CP-3 reactor which was operated at Argonne National Laboratory until 1954.

B. Origin of the Halden Reactor

Since Norway has a very rich potential for hydroelectric power (less than 10% of the potential is currently exploited), long-range planning of the Institutt for utilization of atomic energy has been directed toward steam production for paper mills and other process industries, and toward propulsion of merchant shipping. By 1956, sufficient technology had been developed from the JEEP reactor and other activities to warrant undertaking a larger project. The success of the BORAX experiments in the United States, coupled with the JEEP experience with heavy water lattices, led to a decision to design and construct an experimental boiling heavy water reactor in the 10 to 20-Mw range of thermal power. It was judged that this project would further the national program and would constitute a fundamental contribution to world knowledge.

The site for the reactor was chosen in Halden, an industrial town on the Norwegian-Swedish border about 120 km (75 miles) southeast of Oslo and Lillestrøm. The selection was made on the basis of the availability of water and electric power; the opportunity for convenient disposal of the steam generated; the ability of the town to provide housing for the scientific staff and to supply the nonprofessional working staff required; and the fact that Halden is near enough to the Institutt's main Lillestrøm establishment and supporting facilities to permit operation of the reactor plant as an outstation. An agreement was reached with a Halden paper mill, Saugbrugsforeningen, whereby the latter would provide a site for the reactor (a small mountain in which the reactor could be placed in a man-made cavern, with space for associated buildings outside) and would accept the steam produced by the reactor for use in their processing plant.

The design and construction of the reactor was carried out by the Institutt in cooperation with the Christian Michelsens Institute of Bergen and Norwegian heavy industry. A primary design philosophy was economy in every possible aspect in order to remain within Norwegian resources.

C. The OEEC Halden Reactor Project

1. The Halden Agreement

Concurrent with the construction of the Halden reactor, a study group sponsored by the Organization for European Economic Cooperation (OEEC) was engaged in carrying out an independent examination of the possibilities for constructing and operating research, testing, and prototype reactors as joint European undertakings. In March, 1957, the group recommended several reactor types, including the boiling water type, as promising, but stressed that such a project should be started immediately if maximum value were to be obtained. At the suggestion of the Norwegian government, consideration was given to operating the Halden reactor under such an arrangement. Discussions and negotiations were carried out under the auspices of the European Nuclear Energy Agency, and an agreement was signed in Oslo on June 11, 1958.

The Halden agreement originally provided for joint operation of the Halden reactor for the period from July 1, 1958, through June 30, 1961, and this has since been extended through December 31, 1962. Under the agreement, the capital costs and investments for initial completion of the reactor plant are borne by Norway. The costs of operating the reactor and carrying out the outlined research program, including necessary plant modifications and provision of two fuel charges, are borne by the signatories to the agreement, acting as a group. Except for the liquidation of the first fuel charge and certain housing properties acquired for accommodating the staff, all property remaining at the end of the agreement period will revert to the Norwegian Institutt. Table I lists the countries participating

in the agreement and the responsible orgainizations which are the actual signatories representing those countries. Also listed are the approximate percentages of the total $6,285,000 budget for the extended $4\frac{1}{2}$ -year agreement to be contributed by each organization.

Table I

HALDEN AGREEMENT PARTICIPANTS

Country	Representing Organization	Approximate Percentage of Total Budget Contributed
Signatories		
Austria	The Federal Chancellery, Republic of Austria	3.8
Denmark	The Danish Atomic Energy Commission	3.8
Belgium, France, Germany, Italy, Luxembourg, The Netherlands	The Commission of the European Atomic Energy Community (EURATOM)	25.0
Norway	Institutt for Atomenergi	25.0
Sweden	AB Atomenergi, Stockholm	8.8
Switzerland	The Government of the Swiss Confederation	8.8
United Kingdom	The United Kingdom Atomic Energy Authority	17.0
Associated Parties		
Finland	The Finnish Atomic Energy Commission	3.8
United States	The United States Atomic Energy Commission	-
—	The European Nuclear Energy Agency	-
(Receipts from liquidation of first fuel charge and housing property)		4.0
		100.0

It will be noted that the United States, Finland, and the European Nuclear Energy Agency (ENEA) are considered "associated parties" to the agreement, but are not signatories. The United States is so considered because representatives of the USAEC participated in the agreement negotiations

on an advisory basis and expressed interest in informal participation. Finland lies in this category because of late entry and because of a preference on the part of Finnish government to enter into a separate agreement directly with the Norwegian Institutt, with the approval of the signatories. The ENEA is associated because of its role in fostering the agreement originally.

2. Administering Committees

Under the Halden Agreement, the participation of signatory countries in the project is administered by the Halden Committee, which consists of one member designated by each of the signatory organizations. This committee formally approves the research and experimental program and the budget, and generally concerns itself with patent matters, staff procurement, insurance problems, and relationships with other organizations, such as the International Atomic Energy Agency. In technical matters, the Halden Committee relies heavily on the advice of a second body, the Halden Technical Group, which consists of one senior technical specialist designated by each of the signatory organizations. This group acts in a supervisory capacity over the operation of the project by the Project Management, giving prior approval of programs, budgets, technical progress reports, etc., before they are presented to the Halden Committee.

It is interesting to note, however, that the full responsibility for safe operation of the reactor, for selection and actions of staff members proposed by the signatories, and for appointment of the Project Manager, rests solely with the Norwegian Institutt for Atomenergi. The Institutt has veto power in matters of insurance, safety risks, and actions which might prejudice future use of the reactor. The liability of the signatories is limited to the amounts of their respective financial contributions, and signatories are indemnified by the Institutt against any further claims.

With the reactor operational and the experimental program under way, an increasingly important function of the Halden Technical Group is that of providing active liaison with technical specialists in the signatory countries in an effort to apply the experience existing in those countries to the solution of problems arising at Halden. Under a recent procedural revision, the Project Management, prior to each meeting, issues an agenda which includes descriptions of current technical problems to be resolved, in addition to the normal list of budgets, reports, etc., to be approved. The Technical Group members then brief themselves on the information available in their own countries on these subjects and attend the meeting equipped to discuss the problems. Occasionally, the members bring with them to the meetings specialists who can lend more detailed assistance.

The Halden Committee meets twice each year, usually in Paris. The Halden Technical Group meets four times a year, usually in Halden. Representatives of associated parties are invited to attend these meetings in an advisory capacity.

The Halden Agreement states that decisions of the Halden Committee and of the Halden Technical Group are to be made by a two-thirds majority of votes cast, with Norway and EURATOM having three votes each, the United Kingdom having two votes, and the other signatories having one vote each. It is refreshing to note, however, considering the normal course of international diplomacy, that there has not been a single instance to date where it has been necessary to put an issue to an actual vote; all decisions have been unanimous after initial resolution of differences by discussion.

III. BRIEF DESCRIPTION OF THE HALDEN BOILING WATER REACTOR

A. General Design

The Halden Boiling Water Reactor (HBWR)* is an experimental, indirect-cycle boiling water reactor utilizing D_2O as moderator and coolant, and designed to produce (ultimately) approximately 20 Mw of reactor heat in the form of 230°C (446°F) D_2O steam, corresponding to about 30 ata (440 psia) pressure. This D_2O steam is condensed in a steam transformer or reboiler, and the condensate returns to the reactor by gravity. The heat of condensation is consumed in generating secondary light water steam, which is either discharged to a nearby river or delivered to an adjacent paper mill for use as process steam. A moderator subcooling loop with a heat removal capacity of 1.5 Mw is provided for experimental purposes.

The entire reactor and primary and secondary systems, except feedwater-treatment equipment, are contained in a cavern blasted into a steep mountainside. The tunnel leading to the cavern is 50 m (164 ft) long, and there are about 60 m (197 ft) of rock vertically upward from the cavern to the surface. Both cavern and tunnel are lined with concrete, although this lining is not sealed to the rock at all points, and penetrations exist through the concrete to the rock. The overall layout of the plant is shown in Fig. 1.

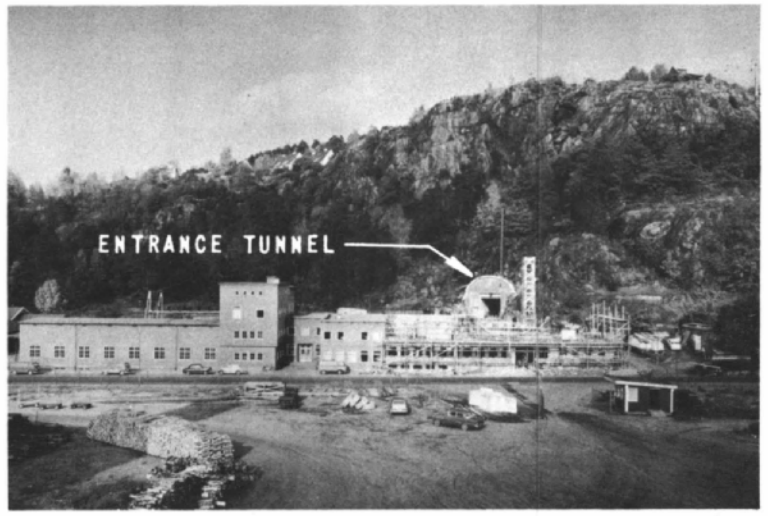
The cavern entrance tunnel is closed at two points by pressure doors, each capable of withstanding an internal pressure of 3 atm. The total containment volume is about 4200 m^3 (148,000 ft^3).

The reactor control room and associated electrical equipment are located in a building near the mountain. All power and instrumentation cables pass from the basement under the control room through a cable duct to the reactor hall entrance tunnel, down the tunnel, through seals in the air lock walls, and into the hall proper. The entrance tunnel also contains all utility, steam and waste water piping to the reactor. The secondary feedwater treatment facilities are located in an enlarged portion at the outer entrance of the tunnel.

B. Reactor Proper

The reactor pressure vessel is designed for a maximum working pressure of 40 ata (590 psia), and has an inside diameter of 270 cm (8.86 ft). The cylindrical portion and lower ellipsoidal head are fabricated of roll-clad plate consisting of a 60-mm (2.36 in.) thickness of ASTM A-212B carbon steel with a 5-mm (0.197 in.) thick layer of Type 304 stainless steel cladding added by arc-deposition.

*HPR-2.



ELECTRICAL BUILDING | REACTOR CONTROL, OFFICE, AND LAB. BLDG.

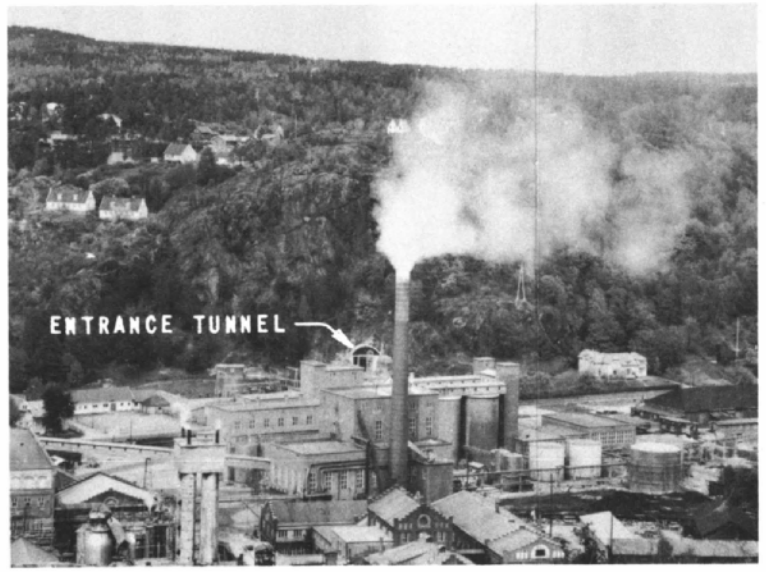


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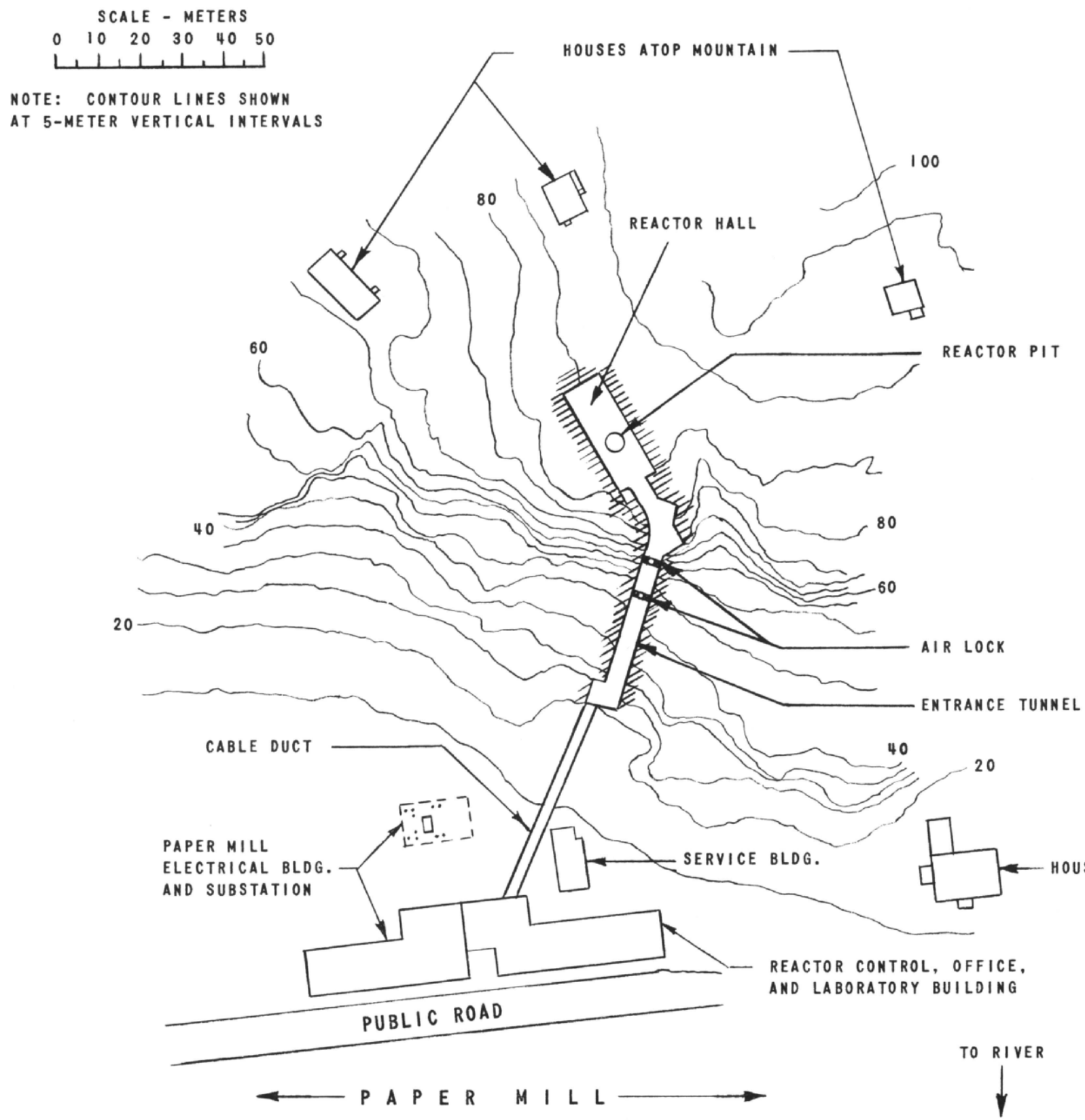
FIG. 1
OVER-ALL PLANT LAYOUT



The flat lid is 70-cm (27.5 in.) thick carbon steel and is penetrated by 316 holes, each 76 mm (3 in.) in diameter, for fuel elements, 19 holes for control rod drives, and 30 holes for steam outlet piping, condensate and subcooled water return, etc. A total of 32 bolts, each approximately 127 mm (5 in.) in diameter, holds the lid in place. Two flat aluminum lid gaskets are used, one on either side of the bolt circle, and leak-off connections are provided from the space between the gaskets. Figure 2 is a schematic view of the reactor vessel and surrounding shielding.

Immediately below and suspended from the lid are two carbon steel plates and one stainless steel plate which serve as shields against vertical radiation and as crude steam separators. These plates are perforated by holes in the same manner as the top lid. The two carbon steel plates and the inner surface of the lid are coated with an electroless "Kanigen" nickel plating.

Also suspended from the lid is a stainless steel plenum chamber which serves as a horizontal guide for the lower ends of the fuel elements. Subcooled water may be pumped into this plenum through the pipes which act as suspension members. The water then flows upward into the lower ends of the fuel elements through orificed openings.

The ellipsoidal lower head of the vessel below the lower plenum is filled with a stainless steel mass which rests on the bottom and serves to displace heavy water to reduce the required inventory for operation.

The first fuel charge consists of seven tons of natural uranium metal canned in 2S aluminum, in the form of rods 25.4 mm (1 in.) in diameter and 2364 mm (7.75 ft) long, disposed in the reactor in a 130-millimeter (5.13-in.) pitch hexagonal lattice. The elements are inserted individually through the holes in the top lid and are suspended from the lid. The lower ends are guided by the lower plenum. At the upper end, each fuel element ends in a stainless steel shield plug, and a pressure seal is made to the lid just below the upper surface of the lid by metal-to-metal contact maintained by a pressure nut. Each fuel rod is equipped with fins extruded on the cladding tubes. A removable shroud tube with holes at top and bottom for natural coolant circulation is fitted to each rod. Figure 3 shows the configuration of a first-charge fuel element, with shroud attached.

With the first fuel charge, operation is limited to a power level of 5 Mw at a moderator temperature of 150° C (302°F), corresponding to about 5 ata (74 psia), in order to retard the corrosion of the aluminum cladding, and to reduce the consequences of a uranium-water reaction in the event of fuel cladding rupture. A number of spike elements, each consisting of six stringers of 1.5% enriched uranium oxide pellets clad in stainless steel, are available for increasing reactivity when desired.

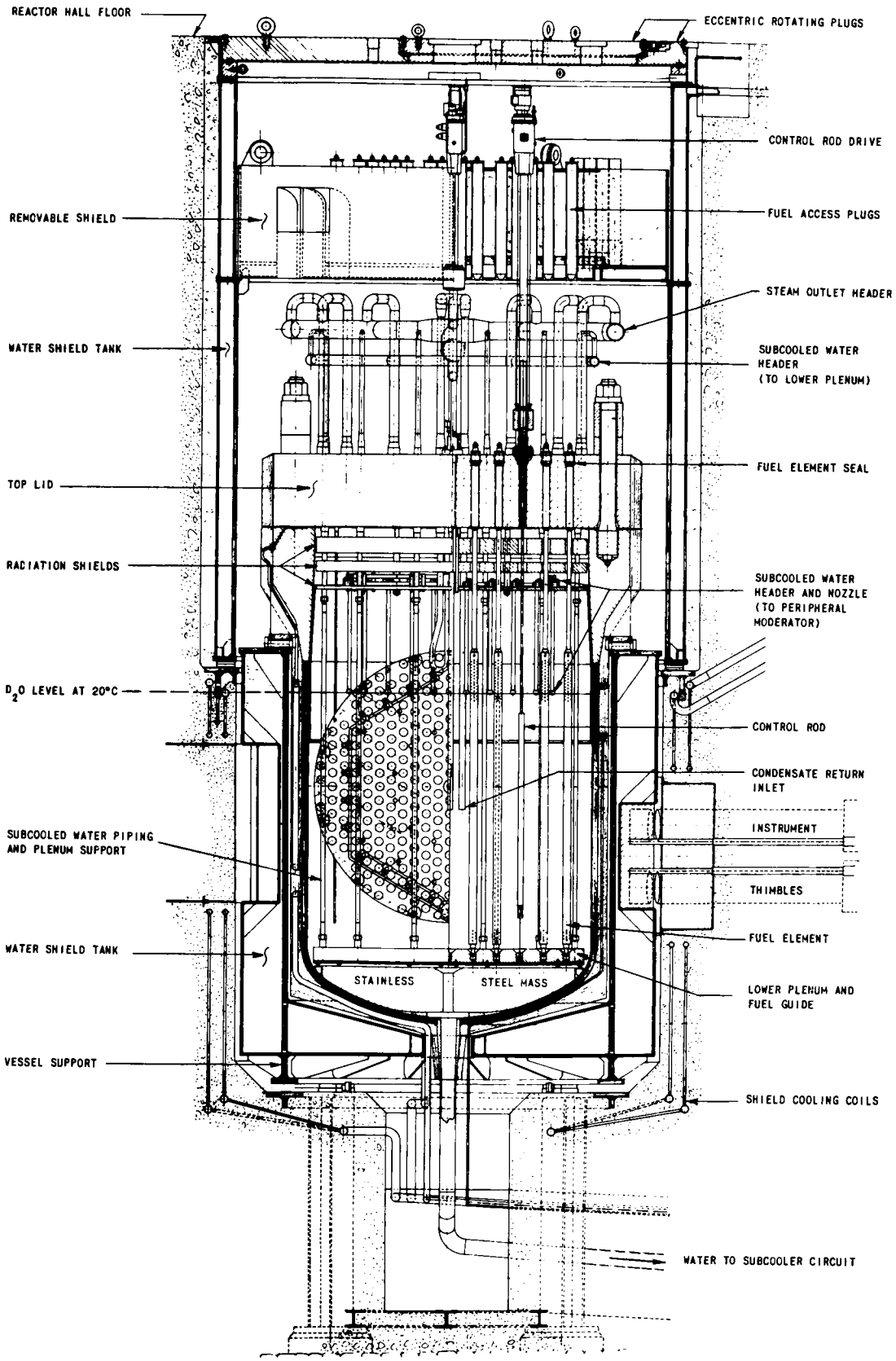


FIG. 2
REACTOR VESSEL AND SURROUNDING CONCRETE SHIELDING

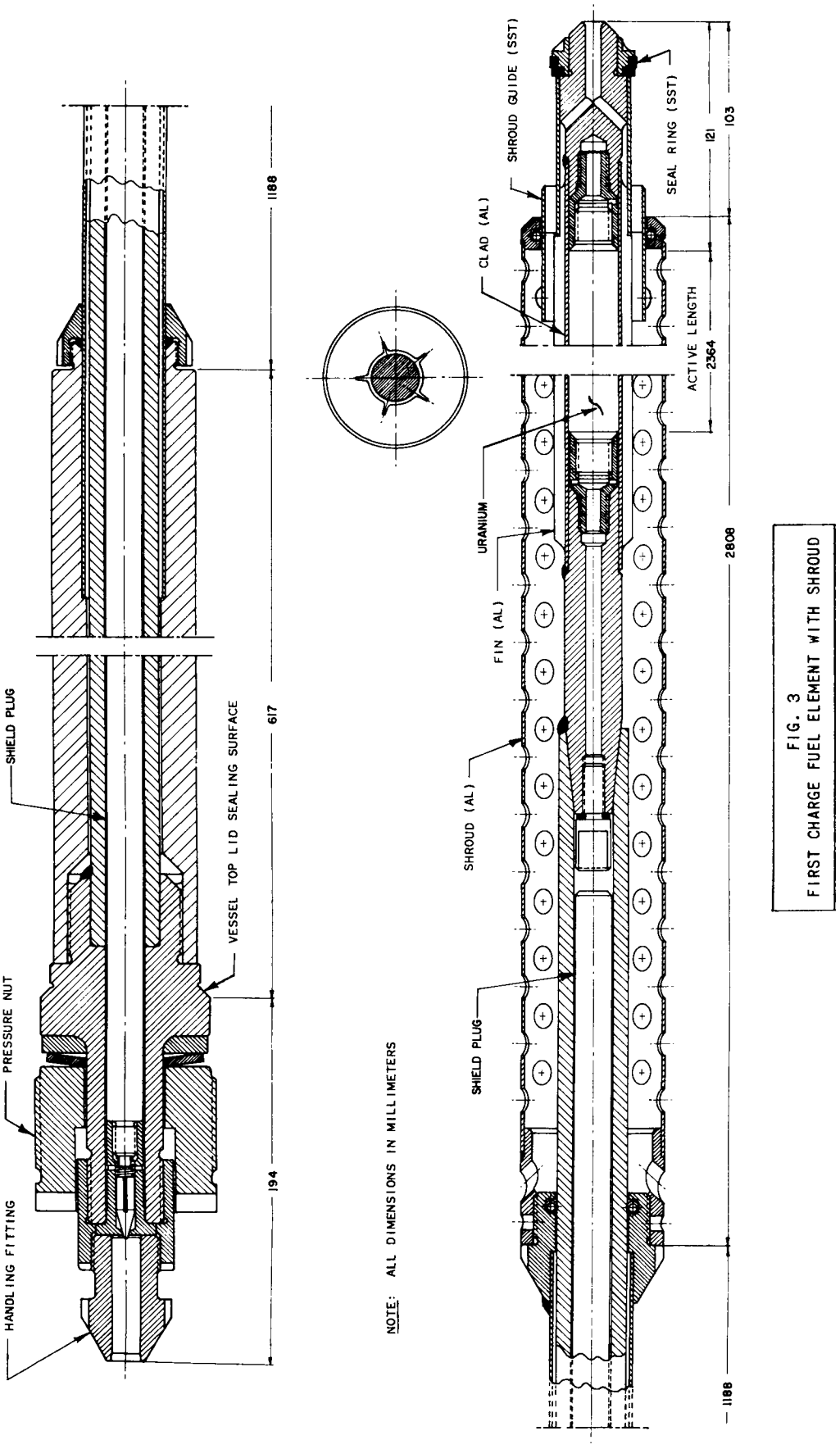


FIG. 3
FIRST CHARGE FUEL ELEMENT WITH SHROUD

A second core, consisting of approximately two tons of 1.5% enriched uranium oxide pellets clad in Zircaloy-2, is currently in manufacture. This core will permit operation up to the design power of 20 Mw. Details of this core, as well as the modifications to the plant to permit disposal of the full design power, are described in Section IV.

For the first core, ten control rods and drives are employed. The rods are cadmium tubes which are clad inside and out with stainless steel. Each rod is guided in its movement by a heavy wire extending from the lower plenum to the top of the drive mechanism. The upper end of each control rod terminates in a plunger of magnetic material which moves within a pressure tube mounted above the top lid. The rod is moved vertically by an electromagnet which surrounds the tube. The electromagnet, in turn, is raised and lowered by a motor mounted at the top. Scram is achieved by de-energizing the electromagnet.

After the completion of first charge operation, these units will be discarded. Thirty new drives of the magnetic jack type will be installed to control the higher excess reactivity of the second charge.

The reactor vessel is located below floor level in approximately the center of the reactor hall. It is surrounded by a light water shield tank. Two toroidal headers, one for steam collection and one for subcooled water injection to the lower plenum, are located above the vessel. A concrete and steel shield plug, containing individual fuel element plugs, is located above these headers. A steel rotating eccentric plug arrangement for fuel handling indexing surmounts the shield at the floor level.

C. Primary and Secondary Systems

Figure 4 is a simplified flow diagram of the plant. The D_2O steam generated in the reactor is collected in the toroidal header above the reactor and passes to a 10-Mw capacity steam transformer, or reboiler. Here the steam is condensed, and the condensate returns by gravity directly to the reactor vessel. The capacity of the present steam transformer is sufficient for operation with the first fuel charge. A second 10-Mw transformer is currently in design and manufacture (see Section IV), and will be installed for full-power operation with the second fuel charge.

A subcooler loop is provided wherein up to 80 m^3/hr (350 gpm) of moderator D_2O may be withdrawn from the reactor vessel at the bottom, cooled in a heat exchanger by removal of up to 1.5 Mw, and re-injected either into the top of the moderator at the periphery of the reactor vessel, or directly into the fuel element shrouds through the lower plenum and guide.

The reactor and associated primary circuits, including purification loops (see Section IV), contain about 16 tons of D_2O.

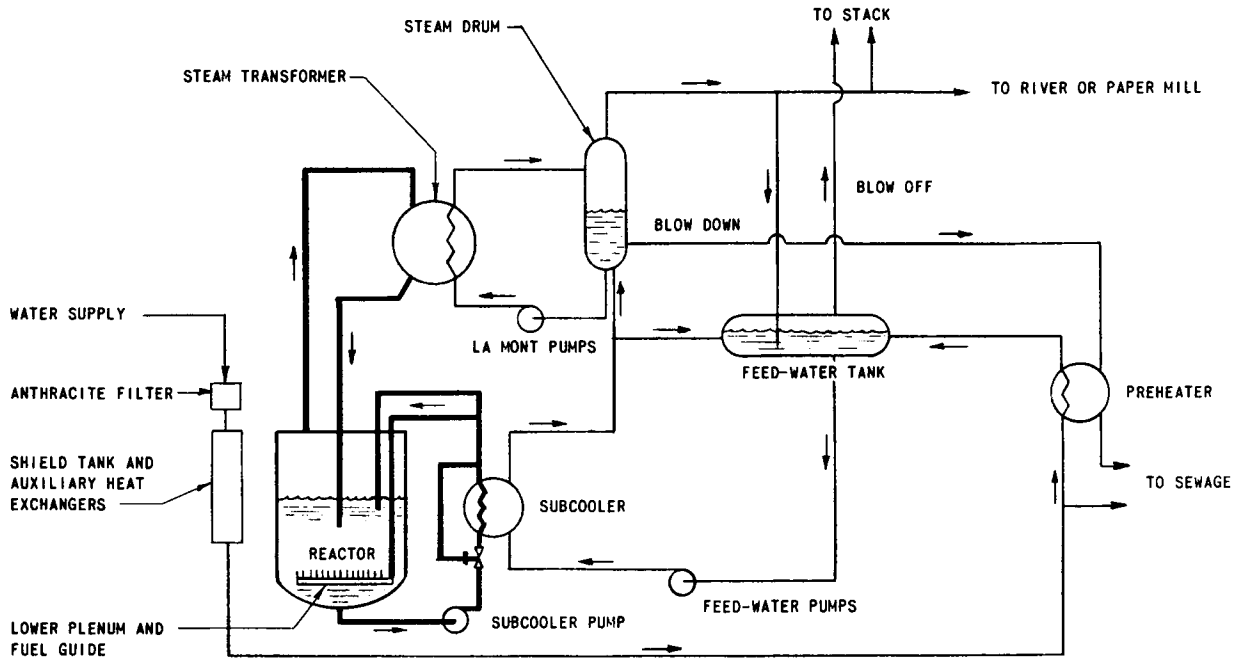


FIG. 4
SIMPLIFIED FLOW DIAGRAM FOR FIRST FUEL CHARGE

In the present plant, secondary light water is preheated first in auxiliary heat exchangers, then in a feedwater and degassing tank, and finally in passing through the secondary side of the moderator subcooling heat exchanger. The preheated water is then fed to a secondary steam drum. The water in the steam drum is circulated by "La Mont" pumps through the secondary side of the D_2O steam transformer, wherein about 10% of the secondary water flow is converted to steam. The steam is separated when the mixture returns to the steam drum, and is discharged to the river nearby. Steam can also be delivered to the Saugbrugsforeningen paper mill, next to the river, for use as process steam. This will not be done, however, until the second fuel charge has been installed, since the pressure of the steam developed in first fuel charge operation will not be high enough for paper mill use.

For second charge operation, the secondary system will be revised to include a closed intermediate circuit. These revisions are discussed in Section IV.

IV. PROJECT ACTIVITIES DURING PERIOD OF PARTICIPATION

A. General Program

The Halden Project program is divided basically into two main categories: research and engineering. During the early period of the project, the international staff was assembled, and most of the effort was directed toward preparing the reactor for criticality. With a loading of 124 first-charge fuel elements (aluminum-clad, metallic natural uranium rods), criticality was achieved on June 29, 1959.

Following criticality, a series of physics experiments was carried out to determine the cold clean characteristics of cores containing 124 and 296 fuel elements. Experiments with the 124-element core were directed principally toward obtaining data of interest for operation of the second core, which is to contain 100 elements of 1.5% enriched uranium oxide pellets clad in Zircaloy-2, and 30 control rod positions. The 296-element core was fixed as the reference for first-charge power operations. These experiments, which included determination of water level and temperature coefficients of reactivity (at 20°C), thermal and fast flux measurements, control rod calibrations, and void-reactivity experiments, were completed in January, 1960.*

Although the reactor was made ready for low-temperature operation by mid-1959, a considerable amount of engineering work was necessary before operation was possible at elevated temperature and pressure. After completion of the low-power, low-temperature physics experiments, all fuel was removed from the reactor, and over 300 new stainless steel insert seats were welded in place in the fuel element holes in the reactor lid. The design of the new seals was the result of continued development work at the Norwegian Institutt's main establishment at Kjeller.

Inspection of the fuel elements revealed serious pitting corrosion at the lower ends. The fins were machined off the lowermost 27 cm (10.6 in.) of each element, and an aluminum sleeve, closed at the lower end, was installed to cover the pitted area and welded to the cladding at the upper end of the sleeve. At the same time, changes were made in the upper ends of the elements to make the sealing surfaces compatible with the new sealing seat inserts in the reactor vessel lid.

The primary and secondary systems were also modified at this time. The changes in the primary system included installation of a new condensate return line to the subcooler pump inlet, a high-capacity filter unit to remove solid corrosion products, and a D_2O filling pump of higher capacity.

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*HPR-3, -4, and -5.

The D_2O ion exchange system was changed from a gravity-feed design to a forced circulation arrangement giving higher flow. Secondary-system modifications included installation of an improved separator unit in the steam drum and provision of motorized drives on several valves.

Some of the engineering modifications necessary for operation with the second fuel charge were also carried out during this period, primarily because of the anticipated difficulty of working near the reactor at a later date after activation by first-charge operation. Notable among these modifications was the machining of gasket surfaces and provision of bolt holes on the top surface of the reactor lid for the 30 new control drive mechanisms.

Other improvements to the reactor plant prior to power operation were made in the process instrumentation, and in the control room instrument and control panel arrangement. The containment of the reactor was improved by sealing the air lock barrier walls in the cavern entrance tunnel to the surrounding rock, utilizing concrete and latex-injection methods.

Following this modification period, the reactor was reloaded and again brought to criticality on August 26, 1960. After a short period of a low-power operation for confirmation of critical data, test runs on active handling equipment, and a pressure leakage rate test on the reactor plant containment, an approach to power operation was made, culminating in boiling operation at 2 Mw on October 5, 1960. Primary D_2O steam conditions were 142°C (288°F) and 3.9 ata (57 psia) during this run.

A number of activities, both in the research and engineering categories, have been pursued concurrently with the above program. A portion of the research effort has been devoted to the installation and operation of an analog computer, and to the development of mathematical models of the Halden reactor to be set up on the computer for use in predicting the dynamic behavior of the reactor.* Both research and engineering efforts have been directed toward development of experimental equipment for use in the reactor, particularly reactivity oscillation devices and in-core instrumentation.

The design and procurement of the second fuel charge, new magnetic jack-type control drive mechanisms, a second steam transformer to be installed parallel to the first to permit 20-Mw operation, and other equipment for eventual high-power operation, have occupied a major fraction of engineering effort throughout the project's history. A continuing program of improvement of containment, waste disposal, operating and experimental procedures, and reactor safeguard measures in general, has been carried out.

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*HPR-5.

B. Research Activities

1. Low-power, Low-temperature Physics Program

The low-power, low-temperature physics program carried out on the HBWR from June, 1959, until January, 1960, included experiments on two core assemblies consisting of 124 and 296 fuel elements, respectively. The 124-element loading corresponded to an effective core radius of 80.5 cm, giving a radial D_2O reflector of 54.5 cm. The final power core of 296 fuel elements corresponded to an effective core radius of 120 cm, and a radial D_2O reflector of 15 cm. The experiments on the 124-element loading were designed to give data of importance for the operation of the second fuel charge. The second charge will have a heavy water reflector of approximately the same size as the 124-element loading.

The reactor physics calculations on the HBWR lattice benefited to a great extent from the large number of experiments which have been reported on the physics of heavy water, natural metallic uranium lattices. The lattice parameters were derived from basic buckling measurements; no attempt was made to develop involved theoretical models. It is interesting to note that good agreement was obtained using standard techniques in the extrapolation from the reported measurements.

The extrapolated critical water height as a function of loading obtained from the HBWR experiments is presented in Fig. 5.

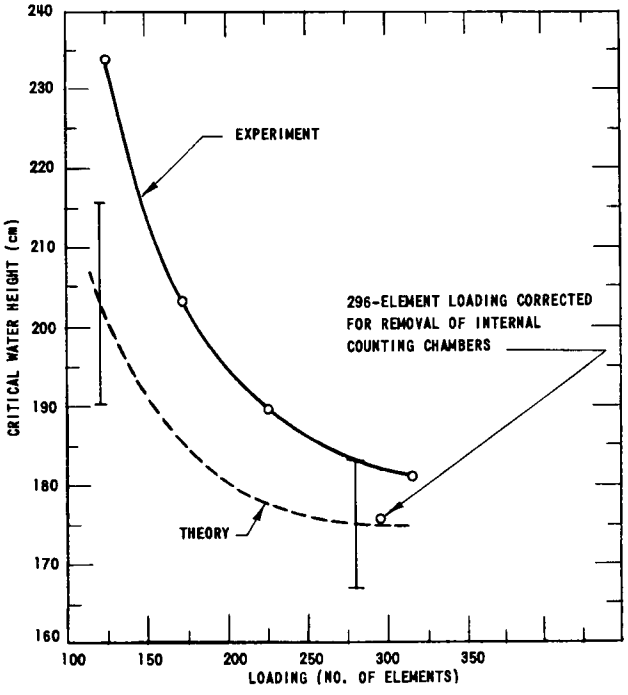

FIG. 5
EXTRAPOLATED CRITICAL WATER HEIGHT AS A FUNCTION OF LOADING.

The results of two-group calculations are presented in the same figure. The errors indicated on the figure correspond to the probable uncertainty in the material buckling of $\pm$ 0.3 m^{-2}. The theoretical results, however, cannot be compared directly with the experimental results, since the influence of empty fuel positions, counting tubes, and control rod guide stringers which penetrated the core axially have been neglected in the calculations. It is estimated that the presence of these items can account for a decrease of approximately 0.3 m^{-2} in the axial geometric buckling (corresponding to an approximately 12-cm increase in the extrapolated height of the 124-element loading). This was verified to a certain extent for the 296-element loading when all counting tubes in the core used for the approach to critical experiments were moved into the reflector, resulting in a decrease of 7 cm in the extrapolated critical height. The new extrapolated critical height established is included in Fig. 5.

The macroscopic thermal neutron flux distribution was experimentally determined in both the 124- and 296-element loadings. A number of Mn-Ni wires extending axially through the core were irradiated at different radial positions. The activities of these wires were measured by a continuous scanning device, enabling a direct recording of the axial neutron flux distribution. The radial distribution within the core, obtained from the axial curves, was fitted to the corresponding reflector distribution, the latter being determined from the induced activity of a Mn-Ni wire extending horizontally through the reflector.

The radial flux distributions obtained for the 124- and 296-element loadings are presented in Fig. 6, along with the distributions obtained from the two-group calculations. The axial flux measurements have established a total axial reflector saving of 14 $\pm$ 1 cm. The major part of this reflector saving is due to the steel bottom plate. The radial reflector saving for the 296-element loading was found to be 22 $\pm$ 2 cm.

From these results, the material buckling was found to be 5.97 m^{-2}, compared with the theoretical calculated value of 6.44 m^{-2}. Taking into account the influence of control rod guide stringers, empty fuel element positions, and counting tubes, as mentioned earlier, the experimental buckling should be increased by 0.3 m^{-2} to yield 6.27 m^{-2}, which compares rather well with the theoretical value.

The water level reactivity coefficient for the 296-element loading has been measured as a function of water level. A typical measured value is 71.5 x 10^{-3} $(\%\delta k/k)/cm$ at a water level of 179 cm. Analysis of the data shows the total excess reactivity of the cold, clean core at the maximum water level in the vessel to be 4.2%.

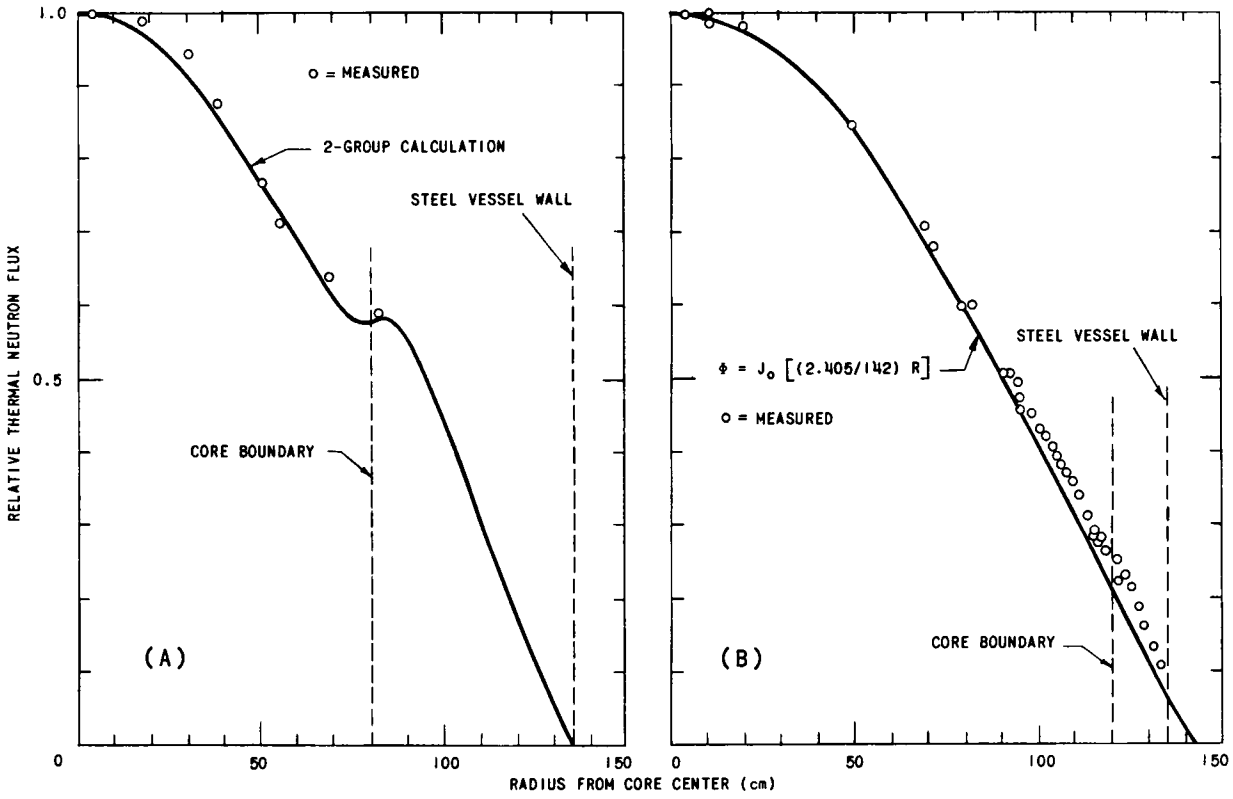

FIG. 6
MEASURED AND CALCULATED RADIAL THERMAL NEUTRON FLUX DISTRIBUTION IN (A) 124-ELEMENT AND (B) 296-ELEMENT LOADING.

The temperature coefficient of reactivity between 20°C and 24°C has been determined by measuring doubling time in the clean reactor. A value of $-(18.1 \pm 0.4) \times 10^{-3}$ (% δ k/k)/°C was obtained.

One of the most important aspects of the low-power, low-temperature physics program was the investigation of void reactivity effects. The measurements were carried out using a method (originally developed for the JEEP reactor) which consisted of pneumatically oscillating the water level confined between two annular channels around a fuel element.(1) The reactor response to these oscillations was analyzed with reactor oscillator electronic equipment, and the results interpreted in terms of reactivity units using a calculated zero power transfer function. The water level could be oscillated between any depths in the reactor. An analysis of the experimental results has shown that it is possible to construct an integral void coefficient from the differential measurements.

A normal first-charge fuel element was modified to permit installation of up to three concentric shrouds of different diameters, the largest shroud having essentially the same diameter and wall thickness as the normal shrouds. Tests were made with each shroud individually and

with all shrouds simultaneously, the latter in order to estimate the effect of the aluminum. Void oscillations were introduced by alternately pressurizing the shroud space with air and venting. Void reactivity interaction effects were also studied by making comparative measurements with static voids introduced around neighboring fuel elements.

An example of the type of effects observed is presented in Fig. 7. The curves represent reactivity tied up in a void column of the same diameter as the normal shroud (the largest oscillation test shroud was used here), and having a depth extending from the normal reactor water level to the depths indicated. A set of values was obtained with the oscillating void test element at each of five different radial positions in the core.

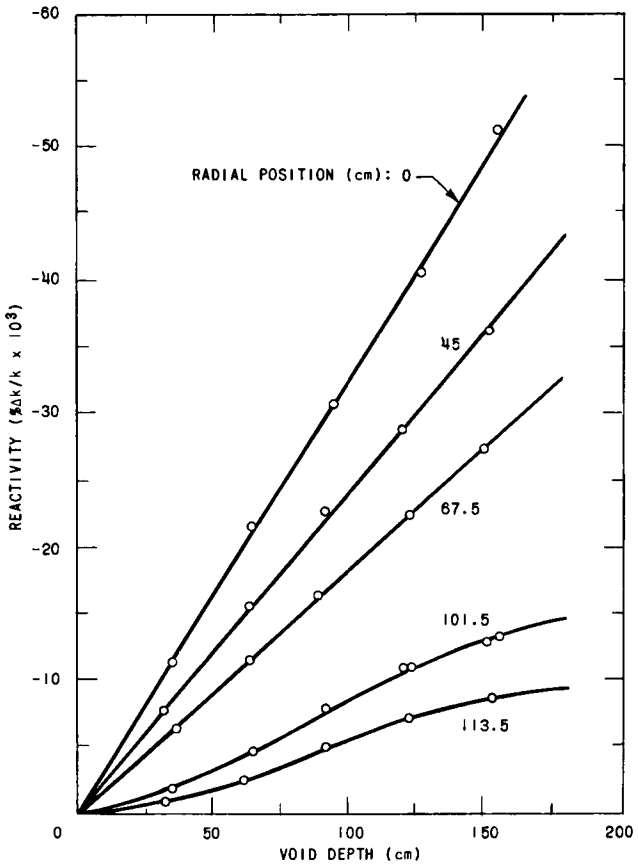

FIG. 7
VOID REACTIVITY AS A FUNCTION OF VOID DEPTH. (NORMAL DIAMETER SHROUD ALONE).

It is interesting to note from the results in Fig. 7 the almost linear behaviour of reactivity as a function of depth of void column, indicating that the reactivity associated with a unit void volume is essentially independent of its axial position at least in the central part of the lattice.

Based on the measured void coefficients and on void distributions calculated from the results of hydrodynamic experimentation,* the reactivity effects of voids have been calculated for different degrees of subcooling and reactor pressures. The results of these calculations are presented in Fig. 8.

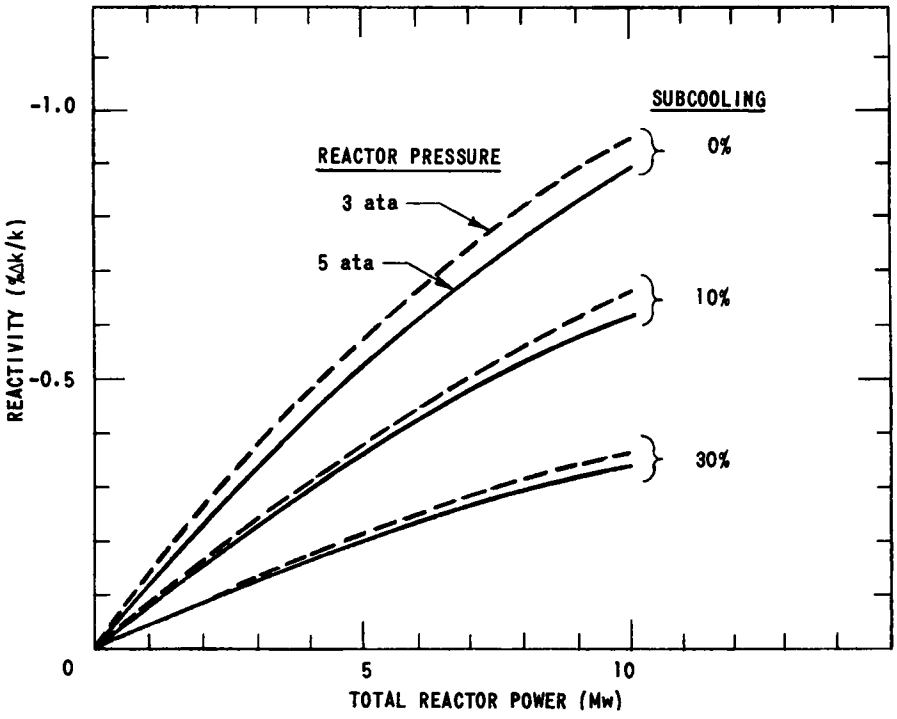

FIG. 8
REACTIVITY EFFECT OF VOIDS AT VARIOUS DEGREES OF SUBCOOLING AND REACTOR PRESSURES

Considerations of water level reactivity coefficient have indicated that 4.2% excess reactivity is available in the cold and unpoisoned natural uranium reactor. A limited number of spike elements, consisting of 1.5% enriched uranium oxide pellets clad in stainless steel, will be available, and critical experiments in the Swedish RO facility indicate that the incorporation of these elements will increase the total available excess reactivity to about 5.2%. Based upon the void reactivity of Fig. 8 for the assumed operating conditions of 5-Mw power, 150°C temperature, 5 ata pressure and 10% subcooling, and based upon a calculated temperature coefficient normalized to the measured room temperature value, the reactivity budget for full-power first-charge operation appears as follows:

Total excess reactivity with spikes, cold and clean		5.2%
Temperature (20-150°C)	-2.54%	
Voids	-0.35%	
Xenon	-1.35%	
Samarium	-0.6%	
		-4.84%
Excess reactivity available at operating conditions		0.36%

*HIR-11.

This excess will not be sufficient for running some of the most reactivity-consuming experiments, e.g., transfer function measurements with approximately 1% reactivity tied up in oscillating equipment in the core. However, it will take some time for samarium poisoning to reach equilibrium, and some of the experiments can be carried out at reduced temperatures and/or before xenon poisoning reaches equilibrium. The time required to reach one-half the equilibrium level is about 15 hours.

2. Initial Boiling Experiment

Although the results of the initial power boiling experiment in the HBWR have not yet been fully analyzed, a preliminary statement of the procedure followed and observations made during the experiment has been issued.

The experiment was begun on October 4, 1960, with conditions of 110°C (230°F), 1.4 ata (20.5 psia) in the primary D_2O system, and 137°C (279°F), 3.4 ata (50 psia) in the secondary system, with the reactor balanced at 1-kw power. The higher secondary conditions resulted from the initial heating of the entire system by electric heating of the water in the secondary surge tank.

The conditions of the primary system were first raised to 147°C (297°F), 4.5 ata (66 psia) by increasing the reactor power initially to 500 kw and later to 900 kw. This operation consumed about $1\frac{1}{2}$ hours. The power was then reduced to 150 kw and held constant, while the secondary steam load was increased, stepwise, by opening the steam-regulating valve. Pressure and temperature in both circuits decreased somewhat. When the primary system temperature reached 140°C (284°F), the reactor power was increased to 400 kw and, later, in steps up to 2 Mw. The first indication of D_2O steam flow was obtained at 400 kw, with a temperature difference of 10°C between the primary and secondary circuits. The primary circuit temperature was kept constant by regulating the secondary steam load. The final conditions with the reactor balanced at 2 Mw were 142°C (288°F), 3.9 ata (57 psia) in the primary circuit, and 122°C (253°F), 2.2 ata (32.5 psia) on the secondary side. These conditions were maintained during a 30-min steady run before the reactor was shut down and the experiment ended.

The following observations were made during the run:

a. Only a weak coupling was observed between the steam load on the secondary side and reactor power at each power level. An increase in steam demand produced an immediate small decrease in reactor power, due to the increase in voids. This was followed a few seconds later by a slow increase in power due to the negative temperature coefficient as the temperature of the water decreased. The rather slow response to changes in power demand made it easy to control the reactor power using the control rods.

b. The noise level increased with increasing power. At 2 Mw the noise level was of the order of 1-2%, with dominating frequencies probably in the range of 1 to 5 cycles/sec.

c. The control rods had not been previously calibrated at high temperatures, so that it was difficult to estimate the reactivity tied up in temperature and voids. The approximate value obtained for the total reactivity appears to check well with the calculated value (based partly on extrapolation from low-power, room-temperature measurements). The void reactivity calculated from the control rod settings at different power levels indicates a higher value than was estimated, possibly due to a higher void content in the vessel than was expected.

d. When boiling in the primary system began, a small amount of deuterium gas was detected. This was recombined immediately, and no further gas was observed.

3. Analog Computer Studies

Theoretical studies of boiling reactor dynamics have been directed toward:

a. predicting the behaviour of HBWR within its operating limits so that the safety of the reactor can be evaluated and the experimental program previewed;

b. extrapolation studies for the two cores to operating conditions which cannot be studied experimentally due to physical limitations of the plant; and

c. efforts to solve the much debated problem of whether boiling reactor stability is primarily limited by power feedback effects or by purely hydraulic effects. In all probability, this depends on the design of a particular reactor, so that an improved understanding of this important problem would be of considerable value for future boiling reactor designs.

The first of these points has been covered, insofar as the first core is concerned, by computer studies using an adaptation of the Beckjord analytical model,[2] tested extensively in the Experimental Boiling Water Reactor (EBWR) at Argonne National Laboratory.[3,4] This model gives the relationships between reactivity and neutron flux in linearized form in terms of a transfer function, and is, because of the linearization, only valid for small deviations from a given operating point (in pressure and power) of the reactor. For another operating point, new parameter values must be computed. A full theoretical model of the HBWR plant based on this principle has been developed for these studies.*

*HPR-5.

For predicting behavior of the second core, and for the extrapolation and stability limitation work, a different mathematical model, developed by Fleck,[5] was adapted for use with the analog computer. This model is concerned mainly with the hydrodynamics of the reactor core. Mass, energy, and momentum balance equations are applied to the core, which is divided into a number of sections corresponding to regions where essentially different hydraulic phenomena occur. One and the same set of equations is valid for any region of power and pressure variation for which the thermodynamic and hydraulic parameters can be evaluated. These advantages are, however, compensated by a considerable increase in complexity over the Beckjord model. The latter, even though fairly simple in principle, is already quite complicated for practical computation purposes.

The analog computer model based on the Fleck approach is described in HIR-24 and -29. The model is applied to one representative flow channel, and the division of the core referred to above has been done in five regions:

1. nonboiling section of the coolant channel (with its upper boundary with the boiling section variable in time);
2. boiling section of the coolant channel;
3. top reflector;
4. steam space; and
5. moderator (downcomer).

Application of the various balance conditions produces nonlinear time-dependent differential equations. The hydrodynamic parameters in this first approach are valid between 2 and 10 atmospheres. The power range is limited downwards by increasing inaccuracy in the selected value of slip ratio, and upwards by inaccuracy in the representation of friction losses when the exit void fraction is close to unity. For the external portion of the plant, the same relationships are utilized in the Fleck model as previously developed for the Beckjord model.*

The Fleck model will be used on the computer with a program of computations starting with investigations for 5-atm, 5-Mw operation as in the earlier work, in order to compare the two models with each other and later with the reactor.

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*HPR-5

C. Engineering Activities

1. Second Fuel Charge Design and Procurement

It was recognized very early in the Halden Project that the first fuel charge (metallic natural uranium rods clad in 2S aluminum) would be inadequate to permit operation of the reactor at its ultimate design conditions of 20 Mw thermal at 230°C (446°F) and 30 ata (440 psia). The excess reactivity was considered insufficient to compensate the void and temperature effects at these conditions, and the operation of such elements at temperatures above 150°C (302°F) was considered too hazardous from the standpoint of the consequences of rupture of fuel elements. Therefore, even before the reactor first became critical with the first charge, effort was initiated toward design and procurement of a second charge.

One of the principal decisions as to the nature of the second fuel charge was that the purpose of the charge was to permit experiments aimed at determining the physical and operational characteristics of the boiling heavy water reactor, and not to test the fuel itself. Consequently, a nonexperimental fuel with maximum reliability was desired. In the winter of 1958-59, two proposals were put forth for procurement of a second fuel charge for the Halden Reactor. One was a stainless steel-clad uranium oxide core to be produced within the Institutt at Kjeller, and the other was a Zircaloy-2-clad uranium oxide core, to be produced by an outside supplier, because of the lack of experience in zirconium technology at the Institutt.

Early in 1959, the decision was made in favor of Zircaloy-2 for cladding. It was felt that the use of stainless steel, which requires high uranium enrichment, would be basically incompatible with the philosophy of D_2O-moderated reactors. Such reactors are theoretically capable of operation with natural uranium, at least in larger sizes.

a. Procurement History

A significant fraction of the author's contribution to the Halden Project consisted of assistance in the field of component specification and procurement. Although detailed procurement histories are normally out of place in a technical report, it is felt that in this case the review of one such procurement is of interest because it reflects current European attitudes toward American suppliers and toward such issues as third-party liability protection and transfer of fissionable material over national boundaries.

General Electric was chosen as the probable supplier because of its unique experience in large-scale production of Zircaloy-clad uranium oxide fuel for the Dresden Nuclear Power Station. Moreover,

it was thought that a cost advantage could be obtained by utilizing fuel segments having enrichment, dimensions, etc., identical with those of the Dresden fuel segments which GE had been mass-producing. An initial meeting was held in April 1959 in San Jose, between General Electric and Halden personnel. The author attended this meeting.

Some time later, a preliminary communication from General Electric indicated a price and delivery less favorable than had originally been anticipated, and discussions were initiated with AB Atomenergi * of Sweden, with a view toward obtaining an alternate proposal. At the suggestion of the Halden Committee, two other firms in the signatory countries, CERCA** of France and DEGUSSA*** of Germany, were also invited to submit offers.

In the interim period between the original invitation and the evaluation of the responses from the four bidding organizations, more detailed physics calculations and engineering studies had been carried out by Halden Project personnel, with the result that requirements were more firmly crystallized, but in some areas differed considerably from the original concept. Consequently, a more detailed revised specification was prepared, and new bids were requested from GE, CERCA, and AB Atomenergi. DEGUSSA was eliminated from further consideration at this point because of failure to offer complete fuel assemblies. Their proposal covered only fuel segments, which would have required fabrication of additional parts and assembly by still another organization.

A review of the three revised proposals received indicated that the prices set forth, adjusted for other factors, such as quantity of enriched UF_6 required as starting material and value of UO_2 scrap left after fabrication, were sufficiently close to each other that no selection was possible on the basis of price alone. The situation was similar with regard to delivery time. It was apparent that the selection would have to be based

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*Aktiebolaget Atomenergi, a company established as a joint government-industry enterprise to plan and implement the Swedish nuclear energy program. The Swedish government contributed $\frac{4}{7}$ of the initial capital investment, and private industry and municipal interests the remaining $\frac{3}{7}$. All later investments and operating expenses are fully covered by the government.

**Centre de Richerches pour Combustibles Atomique, an organization formed by five industrial firms to carry out research and production of nuclear fuels. CERCA has a cooperation agreement with Sylvania-Corning Nuclear Corporation (SYLCOR) in the U.S.

***Deutsche Gold und Silber Scheideanstalt, a private company specializing in manufacture of rare metals. DEGUSSA is the European representative of Mallinckrodt Nuclear Corporation in the U.S.

on the more nebulous grounds of technical experience and manufacturing capability.

Another problem that arose was third-party liability indemnification, which all three bidders requested in one form or another. Correspondence and discussions with some of the bidders indicated that GE could be expected to remain firm on this requirement, whereas a somewhat relaxed agreement might be reached by negotiation in the other cases. It was decided to select one of the two European bidders on a purely technical basis, and then to attempt negotiations on the liability issue.

The fabrication plants of both CERCA and AB Atomenergi were visited by delegations of the Halden Project staff (including the author), and on the basis of previously determined criteria, AB Atomenergi was selected for further negotiations.

Throughout all preliminary discussions with European suppliers up until the final negotiations with AB Atomenergi, it was understood that the enriched UO_2 required would be purchased by Norway from the United States under the existing power bilateral agreement, and delivered to the fuel charge contractor for fabrication into finished fuel elements. The original Halden research program included trans-shipment of a portion of the completed elements to the RO reactor in Sweden for reactivity measurements in different lattice configurations, as had been done with the first charge UO_2 spike elements.

During these preliminary discussions, neither CERCA nor DEGUSSA gave any indication that any legal obstacles existed as to this plan, as far as fabrication was concerned, and it was known that the transfer of some elements to Sweden for testing purposes was permissible under existing agreements, since the transfer of first-charge spike elements had been approved by the USAEC. However, there was some question as to whether the fabrication plan was permissible under the Swedish-American research bilateral agreement. Sweden investigated this matter with the USAEC and assured the Halden Project that an arrangement would be satisfactory whereby Sweden would purchase the UO_2, fabricate the fuel elements, test them in their RO reactor, and then resell them to Norway at the conclusion of the test program. The final contract was drawn up on this basis.

A contract agreement was reached between AB Atomenergi and the Norwegian Institutt for Atomenergi (representing the Halden Project) on November 2, 1959. In this contract, the vendor is indemnified for third-party liability only to the extent that such indemnification is covered by the existing $8,500,000 insurance policy held by the Halden Project. The contract also holds certain other advantages to the Halden Project, notable among which are the retention of responsibility for the fuel by AB Atomenergi

during shipment to the Swedish RO reactor, testing therein and reshipment to Halden, and the agreement by AB Atomenergi to dispose of all UO_2 scrap and to credit the Halden account with at least the USAEC selling price value of the contained U^{235}, with a guaranteed maximum loss of 2%.

b. Design Description*

The overall configuration of the Halden second core of 100 fuel assemblies was determined primarily from the limitations imposed by the existing pattern and size of fuel and control element holes in the reactor vessel lid, and from the desire to maintain a sufficient D_2O reflector around the core to protect the vessel from radiation damage. A series of experiments was conducted in the Swedish-Dutch-Norwegian ZEBRA exponential facility,[6,7] on 6- and 7-stringer clusters with two different stringer diameters. The final design was based on the results of these measurements.

The element design also reflects two basic decisions with regard to mechanical arrangement. Although the first-charge elements are fitted with removable shrouds for inspection and experimental flexibility, it was judged that very little could be gained from inspection of individual stringers in the second-charge element, and there was little, if any, interest in operating the HBWR with unshrouded second-charge elements. Therefore, permanently attached shrouds were specified in order to achieve the greater structural rigidity and lower cost (through simplicity) possible with this type of design. The second basic decision was to have the second-charge elements rest upon the bottom plenum support in the reactor, with the upper ends guided laterally, but free to expand axially. This permits making the shield and sealing plug in the top lid a separate piece which can be removed independently for reworking of seal surfaces.

The details of the second-charge fuel elements are shown in Fig. 9. The fuel segments consist of Zircaloy-2 tubes (12.76 mm ID, 0.76 mm wall thickness) filled with pellets of sintered $1\frac{1}{2}$% enriched UO_2, 12.54 mm in diameter, and at least of 95% theoretical density. Each segment contains 800 mm enriched UO_2. A natural UO_2 pellet is placed at each end of the enriched fuel in order to avoid excessive heat generation at positions where local flux peaking may be expected. The segment is filled with helium and has an axial free space of 1 to $1\frac{1}{2}$% of the fuel length under cold conditions.

Two fuel segments are screwed together to form a complete fuel stringer. Seven stringers spaced together form a fuel cluster, with a total active length of 160 cm.

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*HIR-20

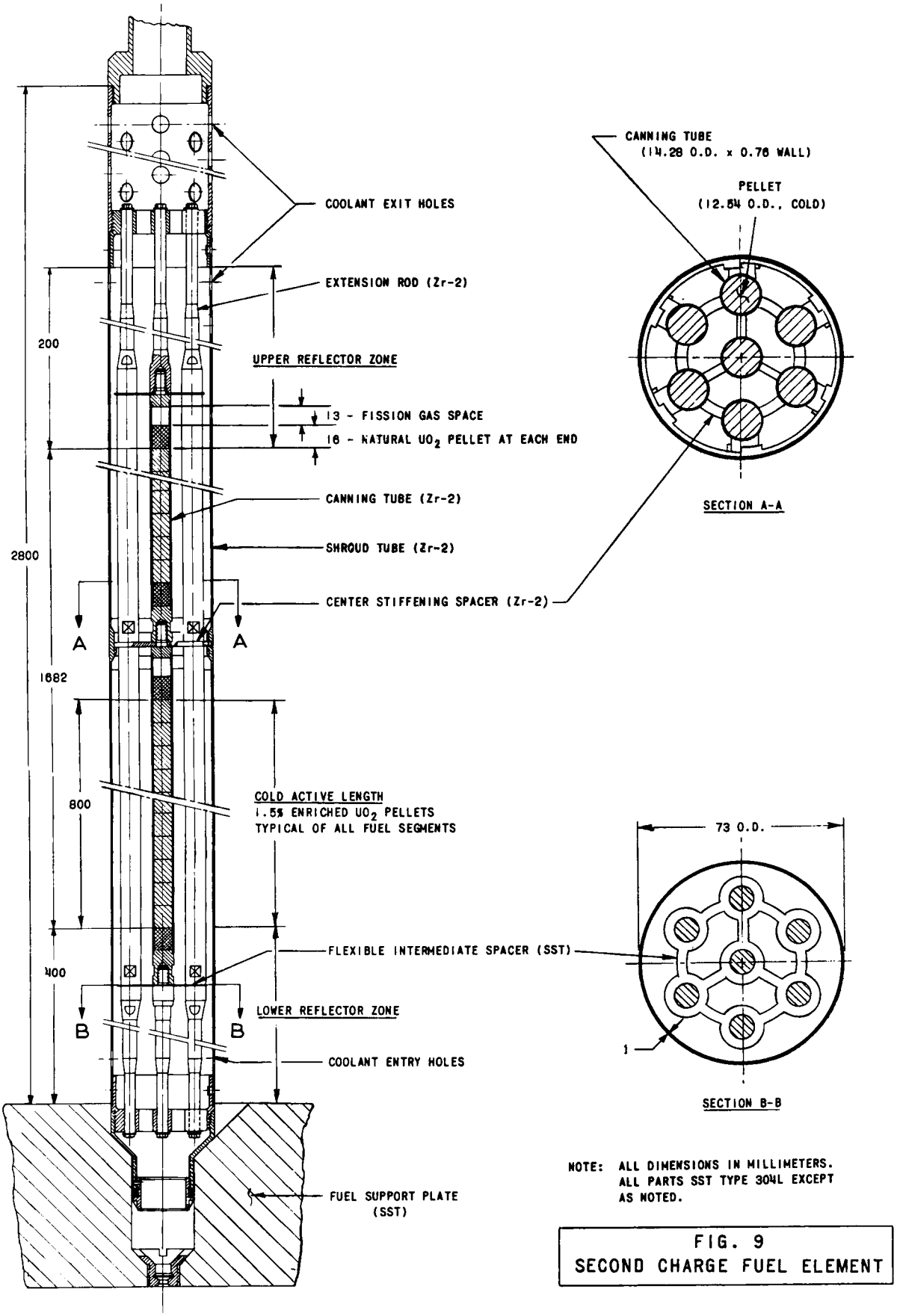

FIG. 9
SECOND CHARGE FUEL ELEMENT

The stringers are connected to top and bottom cluster end grids through Zircaloy-2 extension rods screwed onto the stringers. The Type 304L stainless steel end grids are screwed to the top and bottom extensions of the fuel element.

The lower end of the fuel element rests in the bottom plenum and fuel guide plate in the reactor vessel, and is guided by a stainless steel sealing ring. This ring is held in place by a nose piece which also forms the necessary flow restriction for the subcooled water. The lower assembly also supports the element vertically. The top extension piece proceeds upward to the top lid shield and sealing plug, wherein it is guided laterally.

The 73-mm OD, 1-mm wall thickness Zircaloy-2 shroud tube is riveted at both ends to the stainless steel cluster end grids. A large number of 10-mm diameter holes are provided at top and bottom of the shroud for natural circulation coolant entry and exit.

Primary support of the stringers is obtained by interposing a stiffening spacer of Zircaloy-2 between the upper and lower fuel segments where they are screwed together, and fixing this plate to the shroud tube with a pinned bayonet-type fitting. Axially flexible stainless steel spacers are provided at the upper and lower ends of the complete stringers at the joints between stingers and Zircaloy extension rods, but these spacers are not attached to the shroud. The extreme ends of the extension rods are guided laterally in the stainless steel cluster end pieces, but are free to expand axially. The form of all spacer plates is such as to combine maximum desired rigidity with minimum axial flow restriction.

Table II presents thermal properties calculated to exist in the position of maximum heat load in the second charge with the reactor operating at 20 Mw at a coolant temperature of 230°C.

Table II

THERMAL PROPERTIES OF HBWR SECOND CORE AT POSITION OF MAXIMUM HEAT LOAD

	Full Power	Full Power + 25% Overshoot	Full Power + 25% Overshoot + Hot Spot Factors
Volumetric heat generation, w/cm^3	268	345	455
Heat flux at canning surface, w/cm^2	73.8	92	121
Canning surface temperature, °C	260	265	270
Canning inside temperature, °C	305	321	340
Fuel surface temperature, °C	473	531	596
Fuel center temperature, °C	1800	2180	2760

The hot spot factors referred to in the final column of the table are estimates of 1.20 for flux skewing due to control rods and voids, and 1.10 for local flux peaking due to nonabsorbing materials in the core. These factors are not included in the "Full Power" column. A mean thermal conductivity for UO_2 of 0.02 watt/(cm)(°C) was used in all calculations.

Calculated fluid flow conditions expected in the second core are presented in Table III.

Table III

FLUID FLOW IN HBWR SECOND CORE

	Reactor Power			
	20 Mw		20 Mw + 25% Overshoot	
	No subcooling	30% subcooling	No subcooling	30% subcooling
Average Channel:				
Inlet water velocity, m/sec	1.0	0.9	1.0	0.9
Exit water velocity, m/sec	2.10	1.75	1.80	1.75
Exit steam quality	0.046	0.037	0.058	0.045
Exit void fraction, %	53	48	57	52
Center Channel:				
Inlet water velocity, m/sec	1.0	0.9	1.0	0.9
Exit water velocity, m/sec	2.50	2.0	2.80	2.45
Exit steam quality	0.067	0.053	0.085	0.067
Exit void fraction, %	60	55	64	59
Core Geometry:				
Flow area per channel, cm^2		28.8		
Hydraulic diameter of channel and fuel rods, cm		1.9		
Downcomer area, cm^2		94		

In addition to the 100 basic fuel assemblies, components have been ordered to permit construction of a number of special test assemblies. In-core instrumentation developed by the Project will be mounted in several of these assemblies to permit the measurement of local parameters, such as flow, temperatures, flux, and void fraction For these assemblies, standard fuel segments and stainless steel parts are to be supplied, but a special shroud without coolant holes is provided for further on-site reworking, and the parts are designed for assembly with screws rather than rivets and pins.

The remaining special assemblies are for metallurgical and fuel fabrication test purposes. Here unfilled fuel-cladding components, rather than fuel segments, are provided, and many smaller Zircaloy parts are supplied without surface treatment for corrosion resistance, to allow research in this field. Again the parts are designed for assembly with screws, in this case to permit hot-cell disassembly and inspection after irradiation.

c. Fabrication Details

The author was a member of the delegations which have on several occasions visited the manufacturing facilities of AB Atomenergi in Stockholm to witness various phases of the production of the second charge. Almost all of the uranium oxide pellets have been fabricated, sintered, and ground. A portion of the pellets was made by the Norwegian Institute for Atomenergi Kjeller establishment, by pressing Mallinkrodt enriched UO_2 powder directly. The remainder of the pellets was made by AB Atomenergi in their own installation, utilizing powder obtained by first dissolving Mallinckrodt UO_2 powder and reprecipitating, to obtain improved pressing and sintering properties. The two types of pellets will be kept separate in charging the fuel segments and in assembling fuel elements, so that the effects of the different pellet properties can be observed.

Fuel cladding tubing was obtained from Mallory-Sharon Metals Corporation (now Reactive Metals, Inc.,) Niles, Ohio. (A very active and apparently capable European competitor in this field now is Heraeus of Germany. Heraeus was eliminated from consideration for this procurement because of temporary production difficulties at the time of placement of the order.) The tubing has been tested on a 100% basis by visual inspection inside and outside, eddy current and ultrasonic testing for internal defects, "Vidigage" measurement of wall thickness, and measurements of outer diameter and hardness. In addition, 20% of the tubing has been measured for inner diameter (by air gage), straightness, and length. Lot samples have been given burst, expansion, and flattening tests, metallographic examination, 14-day corrosion tests in 400°C, 100 ata steam, pile oscillation tests, and chemical analyses for impurities.

The large tubing for the shrouds has been ordered, complete with the center bayonet fitting for stringer spacer attachment welded in place, from Imperial Chemical Industries in England.

Completed fuel segments will be radiographed, corrosion tested for three days at 400°C and 100 ata, surface contamination tested, and helium leak tested, prior to further assembly. Acceptance tests of a similar order have been devised for other component parts.

The author was very favorably impressed with the facilities and methods displayed by AB Atomenergi for producing this type of reactor fuel. Their experience in producing similar fuel for the Swedish R3/Adam pressurized D_2O reactor has resulted in the development of excellent pellet-processing and Zircaloy welding techniques, production systems, and quality control methods.

2. Primary Water Purification Facilities

The HBWR primary D_2O-purification system as originally installed consisted of a cooler, prefilters and a pressure-reducing valve discharging to a head tank, as well as a cation resin bed, anion resin bed, and postfilter through which D_2O flowed by gravity from the head tank to a collection tank. The purified D_2O was returned to the system by a makeup pump. The capacity of this system was about 300 liter/hr (1.3 gpm).

A study of the materials in the primary system during operation at 230°C with Zircaloy-clad fuel showed that the situation was essentially similar to that in the Argonne National Laboratory Experimental Boiling Water Reactor (EBWR), wherein a primary water purification system having a capacity of the order of 2000 liter/hr (8.8 gpm) is employed. The decision was made to install a new purification system of this capacity in HBWR in time for second-charge operation. This new system would use mixed beds instead of separate resins, and would be designed for low-pressure operation, receiving its water from the primary system through a pressure-reducing valve, and returning it by means of a high-head pump.

In the case of the aluminum-clad first fuel charge, however, a study* showed that the rate of release of Al_2O_3 to the primary water might be 10-15 gm/hr, even at the lower operating temperature of 150°C. With the existing purification system operating at 300 liter/hr, calculations showed that the equilibrium concentration of solids in the system would be about 50 ppm. Moreover, tests of the existing system showed that because of unexpectedly high system resistances, only 150 liter/hr could be obtained. Since the new 2000-liter/hr system could not be installed in time for first-charge operation, interim modifications were made to the existing system as follows:

a. The head and collecting tanks for gravity flow were eliminated.

b. A positive displacement pump of capacity 0-800 liter/hr, controlled remotely, was added after the postfilter.

c. New heads were installed on the resin tanks to permit operation at pressures up to 10 ata.

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*HIR-4

d. The anion resin bed was changed to a mixed bed. (The cation bed was retained because the present fuel-rupture-detection system depends on the detection of iodine in the water flowing through the purification system after removal of masking cations in the cation bed.)

Room-temperature tests have shown that the revised system is capable of reducing the conductivity of system water (16 tons) to as low as 0.6×10^{-6} ohm^{-1}-cm^{-1} at a flow of 600 liter/hr. In view of the short duration of first-charge operation, it is felt that this system will be satisfactory for removal of ionic and colloidal impurities during this period, even though the capacity may have to be reduced to 400 liter/hr during operation at 150°C because of secondary cooling water limitations.

In order to improve the removal capacity for solid materials (probably about 90% of the released Al_2O_3 will be in this form), the 2000 liter/hr cooler and prefilters for the second-charge system have been fabricated in advance and installed in a temporary bypass loop around the subcooler pump. It is anticipated that this arrangement will maintain the solids concentration below 10 ppm during first-charge operation. When the revised 2000 liter/hr purification system is installed, the temporary bypass will be removed, and the cooler and prefilters will form a part of the new system. The additional cooling required in the new system for operation at 230°C will be supplied by a regenerative cooler installed at that time, along with the new resin tanks, postfilter, and pump.

3. Secondary System Modification

The HBWR as originally constructed is an indirect cycle plant, designed to boil light water on the secondary side of the primary steam transformer (or reboiler) and to deliver this steam at 10 ata pressure to the nearby paper factory. Heat is supplied to the transformer in the form of D_2O steam generated in the reactor. Several factors now in evidence will necessitate changes in this system for operation at the full design capacity of 20 Mw with 230°C in the primary system.

In the present arrangement, the available secondary feedwater is very poor in quality, containing, amond other impurities, 7 ppm of chloride. The raw water is filtered, passed through some auxiliary heat exchangers, and enters the feedwater tank which is maintained at slightly over 100°C for degassing purposes. From the feedwater tank the water is pumped through the secondary side of the primary subcooler, and into the secondary steam drum. A forced-circulation "La Mont" circuit is used to pump water from the steam drum through steam transformer and back to the drum. The forced-circulation flow is such that about 10% of the water is converted to steam as it passes through the transformer.

The system is considered sufficiently safe for first-charge operation at a maximum primary temperature of 150°C with feedwater containing 7 ppm of chloride, even though the subcooler is made of austenitic stainless steel throughout, and the steam transformer has ferritic tubes (very brittle, but probably not subject to chloride stress corrosion), and austenitic double-tube sheets. With a practical blowdown rate from the steam drum, the equilibrium chloride concentration in the steam drum may reach 50 ppm. However, the 150°C temperature is not too serious, and only the tube sheets are affected. Since the design has the double-tube sheets with leak-off space between, the likelihood of losing large quantities of D_2O is small, even if a tube sheet does crack. The subcooler, although all austenitic, is operated at a lower secondary side temperature (100°C), with the 7-ppm chloride concentration. For these reasons, although the system is recognized to be considerably less than ideal, operation with the first core will be carried out with no changes to the system as described.

For second-charge operation at primary conditions of 230°C and 30 ata, other factors prevail. At this temperature, there is much greater likelihood of stress-chloride failure of the steam transformer, and at the higher system pressure the failure would be more hazardous. Moreover, with 230°C water in the primary side of the subcooler, the secondary side can no longer be fed with 100°C feedwater, since excessive thermal stresses would develop in the tube sheet and tubes because of the high temperature difference. Therefore, the secondary water would have to be heated before entering the subcooler, and the only available heating means is the recirculation of some of the hot steam drum water back through the subcooler. This would bring the chloride in the all-austenitic subcooler up to 50 ppm, and this plus the higher temperature and poor initial degassing would almost certainly result in failure. The subcooler has a single tube sheet and is physically located below the D_2O level in the reactor vessel, so that subcooler failure would probably mean serious loss of D_2O and could very rapidly drain the reactor vessel.

A study was made of possible solutions to this problem, in connection with the related problem of specifying the materials of construction and capacity of the second primary steam transformer which must be added in order to bring the heat-removal capacity of the system up to the design value of 20 Mw. The existing transformer is a 10-Mw unit. The following possibilities were set forth:

a. Discard the present steam transformer and subcooler, and procure a new 20-Mw steam transformer and subcooler of Inconel, which is not subject to chloride stress corrosion cracking except possibly under extreme conditions.

b. Retain the existing steam transformer and subcooler, procure a second 10-Mw steam transformer of austenitic stainless steel, and add a new feedwater purification system capable of degassing and demineralizing 27,000 liters/hr (about 120 gpm) of water to the purity where protection against chloride stress corrosion is assured.

c. Retain the existing steam transformer and subcooler, procure a second 10-Mw steam transformer of austenitic stainless steel, and install an intermediate pure water reboiling loop which would include a carbon steel intermediate-to-secondary reboiler compatible for operation with the existing poor feedwater.

The latter plan has been adopted; heat balances and lay-outs have been studied, and many components specified.

A ship-propulsion study group at the Institutt Kjeller establishment has proposed that the new carbon steel reboiler be constructed according to an experimental design felt to have advantage for an indirect cycle boiling reactor for ship application. The design calls for vertical positioning and appears almost identical with the secondary steam generators of the Dresden Nuclear Power Station. As in Dresden, the heating medium is inside U-tubes with the bend of the U at the top. In the Halden case, however, the heating medium is condensing steam rather than forced-circulation water. The steam passes upward on one side of the U-tubes and down on the other, condensing in the process. The principal unknown is whether the steam velocity during the upward passage would be sufficient to sweep all condensate formed in this portion over the top bend in the tubes, or whether the condensate would oppose the steam flow and render the heat transfer surfaces ineffective. No decision has been reached on this proposal, and studies are continuing.

4. Second Steam Transformer Procurement

The background for selection of materials and specification of capacity for the second primary steam transformer has been mentioned above. As part of the investigation of the nature of the secondary system modification to be made, proposals had been invited for a 20-Mw steam transformer both in Inconel and in austenitic stainless steel. After the decision on the secondary system had been made, a detailed bid specification for a 10-Mw austenitic stainless steel unit was prepared, and new proposals were invited from the firms who had replied satisfactorily to the earlier request, plus two other firms recommended by Halden Agreement signatories.

The resulting proposals were evaluated, several of the companies were visited to permit assessment of manufacturing facilities, and a contract was let to AB Rosenblads Patenter of Stockholm. The author was a member of the bid evaluation and visiting team, and assisted in preparing and negotiating the final contract. The unit is now under detailed design, and component materials have been ordered.

5. Proposed Replacement Reactor Vessel

At the time of the original design of the HBWR, the only type of fuel readily available was metallic natural uranium, clad in aluminum. Moreover, the facility was intended to serve primarily as a relatively short-lived research and development tool. These factors had a very great influence on the design.

Shortly after the beginning of critical operation of the plant, however, the Norwegian long-range program was redefined to extend the experimentation at Halden, both in time and in scope, following the present international effort. It became apparent that the existing HBWR vessel left much to be desired in experimental flexibility to carry out such an expanded program. The principal limitation to the use of the vessel is the extreme difficulty in removing the vessel lid to make changes within the vessel, or to test a core of different lattice spacing. Lid removal would involve the cutting away and later replacement of the steam outlet manifold, the subcooled water inlet manifold, and the condensate return line. In addition, the fact that important vessel internals, such as the lower fuel guide and plenum, are suspended from the head would make modifications very difficult even after the lid had been removed.

The problem of alleviating these shortcomings was first studied from the viewpoint of installing a double-flanged section, or "spool-piece," to the top of the existing vessel, such that the lower flange of the spool-piece would mate with the existing vessel closure flange and would be permanently attached, perhaps welded. The upper flange of the spool-piece would be designed for frequent and easy opening, and a new domed lid would be provided to mate with this flange. The piping which passes through the existing lid would be rerouted to enter the vessel through the walls of the new spool-piece, and the vessel internals would be redesigned to rest on the vessel bottom. Discussion with manufacturers showed, however, that the major cost of the spool-piece would lie in the large lower flange. It was indicated that an entirely new vessel would probably cost little more, if any, and would constitute a far more satisfactory solution.

A new replacement reactor vessel was designed, utilizing a maximum of the experience gained at Halden and at other nuclear energy installations during the period since the existing vessel was conceived. Particular attention was paid to alleviating specific problems with the existing vessel, simplifying the replacement process, and making the new vessel a flexible envelope which could contain, with a minimum of further modifications, a wide variety of types of boiling heavy or light water-moderated reactor cores, limited only by physical dimensions and maximum temperature and pressure ratings.

The preliminary design of the replacement vessel is shown in Fig. 10. The surrounding shielding is also shown in order to permit comparison with the arrangement of the existing vessel as shown in Fig. 2.

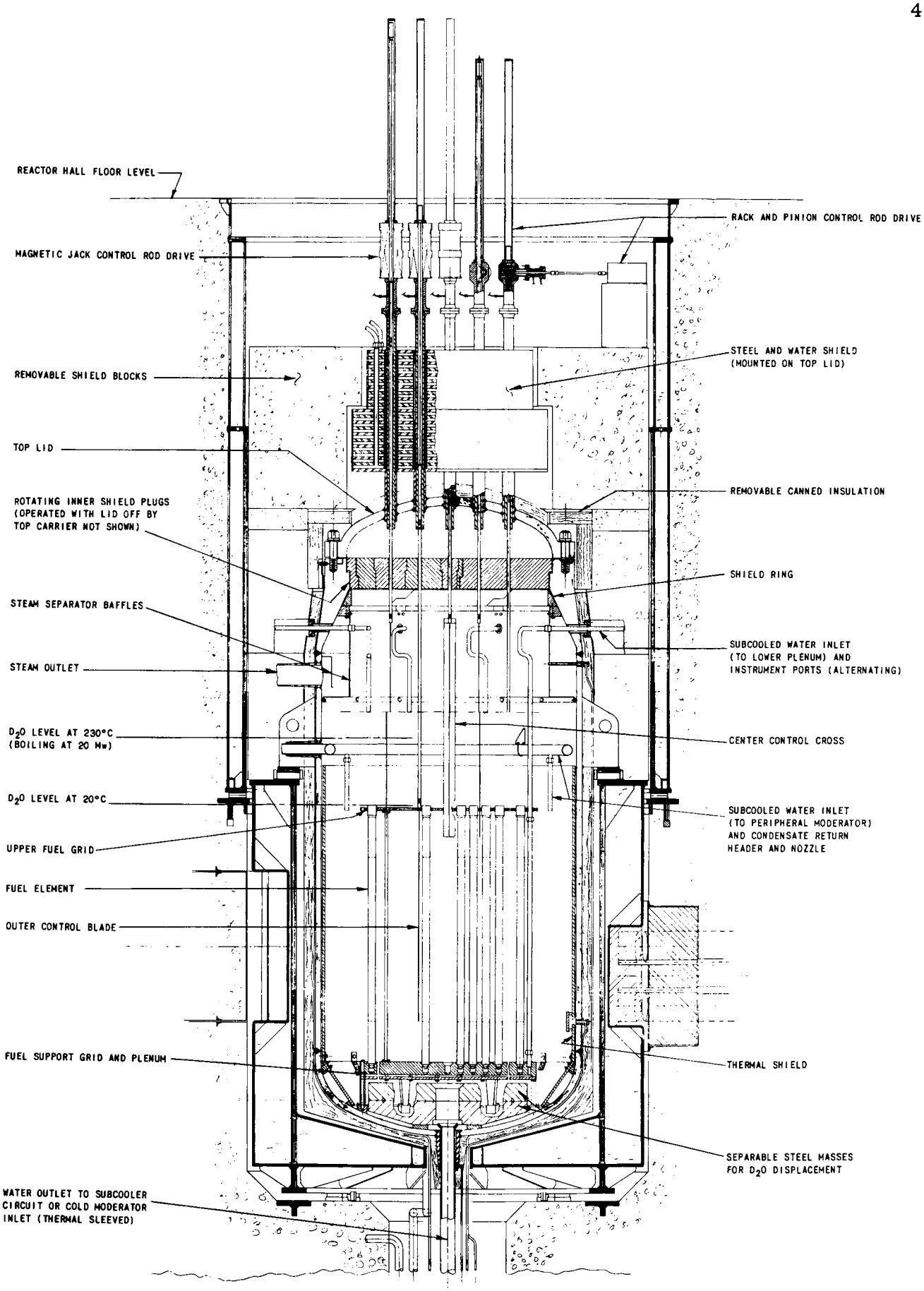

FIG. 10
PRELIMINARY DESIGN OF REPLACEMENT REACTOR VESSEL AND SURROUNDING SHIELDING

The following significant features are included in the concept:

a. The maximum core diameter is reduced to include a permanent radial D_2O (or H_2O if a light water core is tested) reflector of 300-mm thickness. Therefore, the clear opening for installing core components is reduced to 2000 mm (79 in.) from the 2600 mm (102 in.) required in the existing vessel. This makes it possible to use commercially available gasket types which have proved reliable in reactors elsewhere.

b. In the existing vessel, 30 control rods and drives are required for the second fuel charge because of the limitation that the rods must pass through the 77-mm diameter holes in the top lid. In the new design, the number is reduced to 13 through the use of blade-type rods, which are more effective. Moreover, the rods can be placed between, rather than in, fuel lattice positions to permit testing of both an open hexagonal and a closed triangular lattice. For illustration, Fig.10 shows both magnetic jack and rack-and-pinion types of drive mechanisms.

c. The core structure is mounted in the bottom of the new vessel in such a way as to make later replacement with other experimental structures a relatively simple matter. The water outlet to the subcooler circuit, located at the very bottom of the vessel, has been provided with a thermal sleeve where it penetrates the vessel lower head, and with a conical socket fitting at its upper end. This would facilitate connection of the nozzle to a different core support plate, by means of a spring-loaded ball joint. The nozzle could then be used as a cool water inlet directly to the fuel elements, or moderator space above the plate, in experimenting with other types of cores requiring different flow patterns.

d. Additional nozzles are provided in the vessel wall for installation of lead wires and tubes for semipermanent, in-core instrumentation. Extra nozzles are provided in the head for additional control rods, if needed, or for temporary instrumentation.

e. The steam space in the vessel is enlarged for better steam separation, and separator baffles are included. A condensate distribution ring and separate liquid level instrumentation nozzles are provided.

f. A 1% boron-stainless steel thermal shield is included to protect vessel walls from radiation damage, and a 300-mm thick internal top shield plug, similar to that used in EBWR,[8] is provided to shield the top lid as well as personnel working above the reactor during lid-removal operations.

g. A steel-and-water shield is attached to the top lid of the vessel to be moved with the lid as a unit. A fitted concrete block shield, easily demountable, fills the space between the lid shield and the reactor pit walls.

The method of vessel replacement was studied, and a cost estimate and time schedule were prepared. It was shown that considerable financial advantage could be gained by carrying out the replacement between the first and second fuel charges in the summer of 1961. Moreover, the schedule indicated that the change could be accomplished with no delay in the international effort, principally because the engineering program already included a large amount of labor to be expended in modifying the existing vessel for the second charge.

A bid specification for the new vessel was prepared, and proposals invited from 17 companies. Bids were received from ten companies, including one American firm. These bids were analyzed, some of the companies were visited, and an evaluation and recommendation were prepared. It was shown that a satisfactory vessel could be obtained within the cost and delivery estimates originally made. However, because of a number of nontechnical factors, including financing, the decision was made to delay ordering the new vessel at least until such time as would be appropriate for installation after second-charge operation and after conclusion of the international program. In the interim period, the design of the new vessel will be examined further in the light of later decisions regarding the future Norwegian program.

6. First-charge Fuel Element Corrosion

After completion of the low-power physics program with the reactor, a complete fuel element inspection was carried out. It was found that some rather severe pitting of the aluminum cladding of the fuel element had occurred in the lowermost 20 cm of length. Similar pitting was found on the inside of the aluminum shrouds, in the same area. The effect was much worse in the outermost elements, in positions where the communicating holes into the subcooled water inlet plenum were plugged, even though these elements had been in the reactor for a much shorter time than the center ones. Some of the pinholes were as deep as 0.6 mm, or one-third of the total cladding thickness of 1.8 mm.

An extensive program was carried out by the Kjeller Research Establishment to determine the causes of the observed corrosion and to recommend corrective measures. Consultations were also held with British and French corrosion specialists. It was generally agreed that the observed pitting corrosion was primarily a galvanic effect caused by the combination of unsatisfactory water purity and the electrical coupling of the aluminum fuel cans and shrouds to the adjacently located stainless steel shroud guides at the lower ends of the elements. The situation was probably aggravated to some extent by the presence of oxygen in the water, but this effect was felt to be relatively minor. No water purification equipment had been operated during the low-power program.

The Kjeller establishment had observed this type of corrosion previously, but only at low temperatures, as in the Halden case; the effect had never been observed at high temperatures. Because of this, it was thought probable that the elements might still be satisfactory for carrying out the short high-temperature power runs contemplated for the first charge, provided the water quality was improved. However, in view of the very serious effect a fuel rupture could have on the remaining Halden international program, the decision was made to repair the fuel elements at the reactor site.

A total of 316 fuel elements were repaired by the following procedure:

a. The aluminum shroud and stainless steel shroud guide, seal ring, and nose piece were removed. The shroud guide was discarded.

b. The threaded portion of the lower end plug of the element was removed by sawing.

c. The fins and a uniform layer, about 0.3 mm thick (less than the deepest pit depth), were machined from the canning surface in the pitted area, from the lower end (as cut) to a point about 28 cm upward on the rod. The purpose of the machining was to provide a satisfactory surface and uniform geometry for the subsequent fitting of the additional cladding sleeves to be installed. During this process, most of the original end plug weld was machined away, but the elements remained sealed.

d. The additional cladding sleeve, with a new end plug already welded in place, was heated to 250°C and shrunk in position on the element. A vent hole was provided in the new end plug to facilitate this operation and to permit subsequent helium leak testing.

e. The upper end of the additional cladding sleeve was welded to the original canning material.

f. The space between the original canning and the new sleeve was evacuated through the vent hole and filled with helium, and the assembly weld was tested with a mass spectrometer. (The weld of the end plug to the sleeve had been tested before sleeve installation.)

g. The vent hole was closed with a short length of aluminum wire and seal welded. This weld was also helium leak tested.

h. A new shroud guide, made of aluminum to prevent recurrence of the galvanic effect believed to have been a primary cause of the corrosion observed, was installed with the original seal ring and nose piece, and the shroud was reinstalled.

The material for the new cladding sleeves and end plugs was selected to duplicate the original canning material as closely as possible. Filler material of the same composition as the sleeves was used in the welding.

Figure 11 shows the lower end of a fuel element as repaired. A comparison may be made between this detail and the original element, shown in Fig. 3.

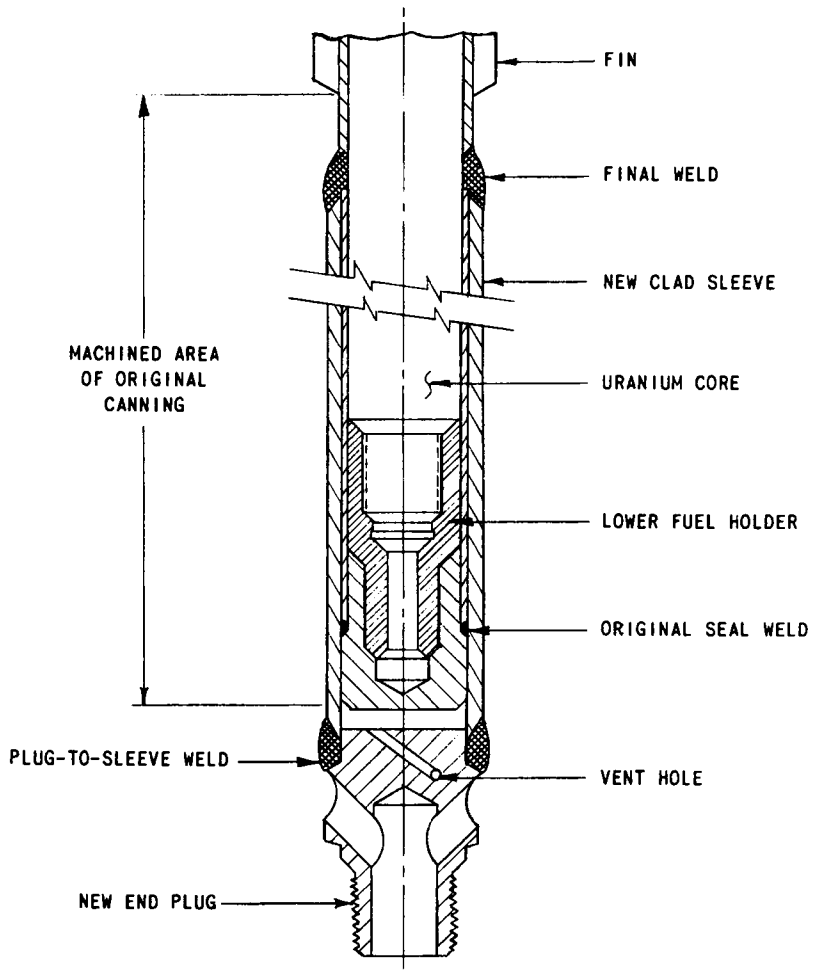

FIG. 11
DETAILS OF FIRST CHARGE FUEL ELEMENT REPAIR

Heat transfer calculations have been carried out to determine the effect of the additional tube on the fuel center temperature. The temperature increase was found to be acceptable, primarily because of the distance between the affected area and the core center.

In addition to the fuel element repair, an increased emphasis was placed upon improving water quality before the elements were

recharged into the reactor. The D_2O-purification system was revised to increase its capacity and effectiveness, as described above. The system was operated for five days, and conductivity was reduced from 9.6×10^{-6} to 0.61×10^{-6} ohm^{-1}-cm^{-1} before the fuel was reloaded. Furthermore, operating regulations were revised to require that the purification system be operated continuously during future low-power experiments, except for short periods when such operation would compromise physics measurements in progress. The purification system is required to be in operation at all times when the reactor is at power levels above 10 kw.

In an effort to gain further knowledge of the pitting corrosion phenomenon, comparative corrosion tests on the following samples will be run in high-purity, room-temperature water:

a. one typical pitted fuel element as exposed to the reactor water during first-charge low-power runs, unrepaired in any way;

b. one typical pitted fuel element as above, unrepaired except for replacement of the stainless steel shroud guide with an aluminum guide;

c. one element as in (a), but chemically cleaned to remove all corrosion product from the surface and as far as possible from the pits as well;

d. one element as in (b), chemically cleaned as in (c); and

e. one fully repaired element.

These tests will be performed with aluminum shrouds in place on the elements, and with the elements electrically insulated from the containers. High-temperature (150°C) tests and cycling tests in progress at Kjeller will also continue.

7. Containment Improvement

The original plan for the containment cavern for the HBWR was to excavate the cavern and leave the walls and ceiling in the "as-blasted" condition. It was later found that the selected location in the mountain contained many rock faults, and required the addition of a concrete lining.[9] Because of the irregular shape of the cavern surfaces after blasting, this lining was not sealed to the rock at all points. However, direct leakage through rock fissures was not considered to be a major problem because the minimum thickness of the rock surrounding the cavern is 30 m (98 ft), and intensive geological studies have shown that the rock fissures are filled with a type of clay which possesses good ion exchange properties.

On the other hand, it appeared possible that fission products resulting from a major accident might find a direct leakage path through cracks in the disturbed rock surrounding the cavern entrance tunnel or through unsealed areas between the rock tunnel and its concrete lining, and thus bypass the tunnel air locks and escape rapidly. In order to correct this situation, about 140 radial holes were drilled through the concrete lining and continued for varying distances up to 6 m (20 ft) into the adjacent rock, within the air lock. The holes were drilled in three groups, with two groups adjacent to the air lock walls, and the third group in the center of the air lock between the walls. Each hole was individually leak tested by applying air pressure at 3 ata, and progressively filled with fine concrete grout by high-pressure injection until further leakage tests showed the hole to be sealed. Latex injections were also used in the center group of holes. Initial testing revealed that some of the holes were leaktight before injection, while others could not even be brought up to the test pressure.

Although it is not considered possible to effect an absolute solution to a leakage problem of this nature, it is felt that the technique described has provided a reasonable assurance that the rock surrounding the tunnel air lock walls in the radial direction has been sealed at least to the extent that leakage through this path will be no greater nor more rapid than through the main rock cover of the cavern.

A pressure test was run on the cavern prior to power operation of the reactor in order to establish the leakage rate and thus provide a basis for setting safe limits of fission product buildup in the first core. A pressure limit of 1000 mm of water gauge was imposed to avoid damage to reactor equipment and instrumentation. Both air lock doors were closed during most of the test, and the pressure decay was monitored over a 25-hour period in both the cavern and the air lock. Air temperatures at five points in the cavern, humidity in the cavern, and outside barometric pressure were also measured throughout the test. The outer air lock door was opened for the final six hours of the test period. The cavern volume is 4200 m^3 (148,000 ft^3).

The results of the test showed that leakage followed an exponential decay. As pressure built up in the air lock and in cavities in the surrounding rock, the decay constant for the cavern (the time required for reduction of pressure by a factor of e) was determined to be 30.7 hours over the period from 3 to 14 hours after the start of the test. The ratio of pressure in the cavern to that in the air lock showed that equilibrium between these two volumes was reached after about 9 hours. After the initial transient, the leakage decay constant decreased to 19.4 hours. Upon decompression of the air lock after 19 hours of the test, two specific leaks were found in the inner door and in the ventilation duct. These were calibrated and found to constitute about 2.5% of the total cavern leakage rate. The longer decay constant of 30.7 hours compares favorably with similar tests performed on a small test cavern in Italy;[10] in the latter case, a decay constant of 30.3 hours was determined.

Extrapolation of the observed rate to higher cavern pressures has been made. At cavern pressure of 2.0 bar * absolute, the extrapolated rate is 30% of the cavern volume per day measured at atmospheric pressure and 20°C.

In comparison with the extrapolated leakage rate of 30% per day, the leakage rate specification for steel shell containment in the United States ranges from 0.1 to 0.25% per day. It must be recognized that, in the case of steel shell containment, gases which leak out are released immediately to the surroundings, whereas leakage from cavern containment involves both a time delay during which radioactive decay can be significant, and the possibility of absorption of the active products by the rock or clay through which the leakage is proceeding.

Correlation of the leakage rates observed in the pressure test with anticipated fission product releases under possible accident conditions is in progress, and an attempt is being made to consider the factors of delay and absorption in the correlation. It is expected that further details of the test and the results of the correlation efforts will be pbulished as a Halden report.

8. Metallurgy and Chemistry Programs

In view of the limited financial and staff resources of the Halden Project, it has been necessary to restrict the HBWR program to certain selected fields. In general, the program is oriented strongly toward research in the physics and dynamic behavior of the reactor, with emphasis on the development of in-core instrumentation. Effort in metallurgy and chemistry has been expended and is planned only to the extent necessary to ensure safe reactor operation.

A metallurgical program has been carried out to determine the impact properties of the reactor vessel construction materials.** Identical sets of full-size Charpy V-notch specimens were made up from the rolled plate used for the vessel sidewalls and lower head, the forged material used in the top lid and upper flange, and the heavy welds joining these two materials. One set of samples was tested unirradiated, partly at the Engineering Research Foundation of the Technical University of Norway, and partly at Oak Ridge National Laboratory. Two other sets of samples were irradiated at 200°C to integrated fast fluxes (above 1 Mev) of 5×10^{18} and 1×10^{19} nvt, respectively, and tested at ORNL.

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* The bar is a unit often used to express barometric pressures. One bar is defined as 1.000×10^6 dynes/cm^2, or 0.987 atm.

** HIR-21

The ductile-brittle fracture transition temperatures of all materials were observed to increase with irradiation, as has been observed elsewhere. A tentative decision has been made, as a result, that after an appropriate point in the high-power operating period of the reactor, the vessel will not be permitted to cool below 40°C, in order to avoid the brittle range. An irradiation thimble facility has been installed in HBWR near the vessel wall, and the periodic testing of samples exposed in the facility will provide a record of changes in vessel material condition. Safe operation will be assured by frequent reappraisal of operating rules and procedures, based on these test results.

Another metallurgical program planned involves the use of special second charge fuel element component parts to fabricate special assemblies for exposure in HBWR as part of the fuel development effort of the Kjeller Research Establishment.

Early plans for chemistry work at Halden included only such items as maintenance of the water-purification system, deuterization of ion exchange resins, analyses for impurities and for isotopic ratio, and certain decontamination problems. The recent incident of the first fuel element corrosion, however, has refocussed attention on this subject, and a somewhat more detailed chemistry program is now under consideration. The proposed program includes:

a. factors controlling the dissociation and recombination of heavy water in an indirect cycle boiling reactor, including N^{16} evolution and its effect on system shielding requirements;

b. distribution, chemical composition, and specific induced radioactivities of the different components dissolved and suspended in the reactor water under varying reactor conditions;

c. effects of corrosion and radiation on materials of construction;

d. factors controlling the corrosion of the aluminum and Zircaloy-2 cladding of fuel elements;

e. probable chemical composition and distribution of fission products throughout the primary system following a fuel burst;

f. decontamination factors of the ion exchange resins for the various radioactive isotopes; ion exchanger studies and filter investigations; and

g. decontamination and cleaning of primary system components after contamination or corrosion.

The program as visualized provides for intensive cooperation with the Kjeller Research Establishment, including parallel experimentation in out-of-reactor loops and in an in-reactor loop under consideration for the JEEP-2 reactor at Kjeller.

D. Safeguard Committee

Late in 1959 the author was requested to serve on the Halden Reactor Safeguard Committee, which at that time consisted of three members from the Institute for Atomenergi at Kjeller and two members from the regular project line organization at Halden. It was felt that the author's position as General Consultant to the Project Management gave him the necessary interest in Halden matters, but at the same time allowed him to retain a detached attitude not always possible with those in the line organization. After a short time, the author was asked to serve as Chairman of the Committee.

Initially, the responsibilities of the Committee and its relationship with the Project Management and other safety bodies at Kjeller were not clearly defined. The Committee developed a proposed statement of position on these subjects, as well as a mode of operation patterned after reactor safeguard procedures in use in the United Kingdom and the United States. This was adopted and later served as the basis for a complete reconstitution of the IFA safety organization.

Under the new arrangement, the highest safeguarding body is the Institutt for Atomenergi (IFA) Safeguard Committee, which acts in a direct advisory capacity to the Director of IFA. The IFA Safeguard Committee consists of the Director of Physics at Kjeller, who serves as Chairman, the Director of Chemistry and Metallurgy at Kjeller, the Chief of Operations at Halden, one member elected by the employees of IFA (according to Norwegian Law), and the Chairmen of four working subcommittees concerned with (1) Kjeller reactors; (2) the Halden reactor; (3) the handling of radioactive materials; and (4) industrial safety.

The Halden Safeguard Subcommittee, in turn, consists of a Chairman, three members from Kjeller, and two members from Halden, all specialists in diverse fields. A permanent secretary serves both the IFA Safeguard Committee and all of the working subcommittees.

In practice, the Halden Reactor Safeguard Report is considered the basis for all Halden safeguard considerations. Within this framework, regulations and procedures compiled by the Project Management are submitted to the Safeguard Committee for approval. Physical changes in the plant, wherein the actual construction is made to differ from the descriptions in the Safeguard Report, are also submitted to the Committee for approval, together with justifications as to Management's claims that the revised arrangements are safe. The Safeguard Committee also has the right to bring to the attention of Management any conditions or operations which it observes and considers potentially hazardous, but it remains the prime responsibility of Management to bring proper items to the Safeguard Committee for consideration, and to act appropriately upon the Committee's recommendations.

Each item is reviewed initially by the Halden Subcommittee. Decisions of the Subcommittee are referred to the Chairman of the main IFA Safeguard Committee, who then either issues them directly to the Halden Chief of Operations over his own signature as well as that of the Halden Subcommittee Chairman, or calls a meeting of the full IFA Committee to consider the matter further. Copies of all submissions and decisions are transmitted to the Director of IFA for his information and further action as desired.

During the author's tenure, the Halden Safeguard Subcommittee assisted the Project Management in outlining the necessary documentation for approvals prior to power operation, and reviewed such items as the operating certificate which contains all core loading, physics and nuclear instrument information for operation; status reports on the fuel elements, the reactor plant, and containment; a statement of the responsibilities of the operating staff; rules for handling radioactive materials; general rules of experimental operation; and emergency regulations.

E. Halden Committee and Halden Technical Group Meetings

During the period of participation, three meetings of the Halden Committee and six meetings of the Halden Technical Group were held. As previously mentioned, the USAEC is invited to attend these meetings as an Associated Party to the Halden Agreement. Because of other commitments, the USAEC Representative in Paris was rarely able to attend these meetings, and the author's predecessor at Halden, J. A. DeShong, Jr., had arranged to attend these meetings in order to assure U. S. participation. The author continued this practice.

Table IV lists all meetings of the Halden Committee and Halden Technical Group which have been held to date, together with an indication of U. S. representation at the meetings. The names of J. A. DeShong, Jr. and the author are indicated in parentheses, as their attendance at meetings did not constitute official USAEC representation.

The most important subject discussed at the meetings of both committees during late 1959 and 1960 was the prolongation of the Halden Agreement from three years to four and one-half years. The original three-year program was compiled on a very approximate basis, and by mid-1959 it was apparent that completion by June 30, 1961, would be impossible without a very significant curtailment of experimentation. For example, power operation of the second fuel charge would have had to be virtually eliminated, since the fuel elements could not be obtained before March, 1961.

A detailed report was prepared, describing the project history to date, scheduling the engineering and research work to be carried out, and estimating budget and staff requirements for the remainder of the project,

including the prolongation period. After several revisions, this report was accepted by both the Halden Technical Group and the Halden Committee, and a draft was submitted to the signatory countries for action. Final ratification was completed on July 13, 1960, formally extending the Halden Agreement to December 31, 1962.

Table IV

HALDEN COMMITTEE AND HALDEN TECHNICAL GROUP MEETINGS

	Committee and Meeting No.			
Date	HC	HTG	Location	U.S. Representative
7/10/58	HC-1		Halden	J. R. Quinn
9/6/58		HTG-Informal	Geneva	-
9/9/58	HC-Informal		Geneva	A. Bishop
10/17/58		HTG-1	Halden	-
1/30/59		HTG-2	Halden	-
1/31/59	HC-2		Oslo	J. R. Quinn
5/21/59		HTG-3	Halden	(J. A. DeShong, Jr.)
5/30/59	HC-3		Paris	-
8/11/59		HTG-4	Halden	(L. W. Fromm, Jr.)
11/4/59		HTG-5	Paris	(L. W. Fromm, Jr.)
1/19/60		HTG-6	Oslo	(L. W. Fromm, Jr.)
2/9/60	HC-4		Paris	J. K. Rouleau, (L. W. Fromm, Jr.)
4/4-6/60		HTG-7 enlarged	Halden	(L. W. Fromm, Jr.)
5/20/60	HC-5		Halden	(L. W. Fromm, Jr.)
8/25/60		HTG-8	Halden	(L. W. Fromm, Jr.)
	Meetings Held since Sept. 22, 1960			
10/17/60	HC-6		Paris	-
11/22/60		HTG-9	Halden	-
	Future Meetings Scheduled			
2/21/61		HTG-10	Halden	
5/3/61		HTG-11	Halden	
5/25/61	HC-7		Halden	

The disagreement as to patent procedures, between the USAEC on the one hand and European organizations such as the Halden and Dragon Projects and EUROCHEMIC on the other, has been discussed at several Halden Committee meetings. An impasse had arisen from the fact that

under U. S. policy the public has free use of all patents resulting from government-supported research, whereas under normal European practice the governments hold such patents and profit by them for the benefit of the people as a whole. The principal concern of the Committee was that the continuing disagreement constituted a block to the further participation of American personnel in these European projects. (It is the understanding of the author that this problem has now been resolved, at least in principle.)

The seventh Halden Technical Group meeting was organized as a two-day "enlarged" meeting on April 4 through 6, 1960. The regular Halden Technical Group members were privileged to bring with them delegations of specialists to participate in first and second-day symposia on the accomplishments of the low-power physics program and the future power program, with the objective of reaching a decision as to proceeding with power operation. A total of 38 persons attended the symposium sessions in addition to the Halden staff. The regular Halden Technical Group meeting was held on the third day.

A similar enlarged Halden Technical Group meeting is tentatively scheduled for the autumn of 1961.

A recent change in the format of Halden Technical Group meetings has greatly assisted the Group's performance of one of its important functions, that of broadening the base of the project by bringing the detailed experience of the participating countries to bear on Halden problems. With the new format, principal technical subjects for discussion are announced well in advance of the meeting, and the members either brief themselves on the subjects, or arrange for appropriate specialists to attend the meeting as advisers. Three such advisers attended the eighth HTG meeting on August 25, 1960. The principal subject was the first-charge fuel element pitting corrosion, and much information valuable to the Project was presented. At the forthcoming meeting on February 21, 1961, the principal subject will be the plans for mechanical modifications to the plant between first and second-charge operation.

V. FUTURE PROGRAM

A. General

The Project is currently engaged in a program of physics measurements at low-power, high-temperature conditions in the reactor. This will be followed by power operation of the reactor up to 5 Mw with the first fuel charge. During this period the plant dynamics will be determined and compared with the results of analog computer studies. Present plans call for completion of this phase by the summer of 1961. After shutdown for installation of the second core and associated equipment, the final year (calendar 1962) of the international program will include power operation up to the full 20-Mw design capacity, with experimentation similar to that carried out with the first core.

B. Research

A low-power, high-temperature program will be carried out before the reactor is again brought up to power with the first fuel charge. The main objective of this program is the measurement of the temperature dependence of the void- and temperature-reactivity coefficients. During these measurements, the reactor will be heated electrically. The void reactivity coefficient will be measured at system temperatures up to about 80°C, and the temperature coefficient up to operating conditions (150°C). It has been difficult to develop a method for measurement of void coefficient which can be used above atmospheric pressure, and this has resulted in the strong temperature limitation for these measurements.

The void distribution will distort the neutron flux distribution in the core. Determination of the neutron flux distribution therefore will be made axially and radially at different operating conditions during power operation with the first fuel charge. Preliminary calculations have indicated that the effect is not very strong. As has been pointed out by others, the effect seems to be intimately coupled to the size of the reactor core. The large neutron leakage from the first fuel charge seems to "stiffen" the neutron flux distribution for moderator perturbations of this type.

A series of nonboiling experiments is contemplated before the first systematic boiling measurements start. During these experiments, boiling will be suppressed and the power removed from the reactor through the subcooler loop. Measurement of xenon and samarium poisoning will be carried out.

The main objective of power operation with the two fuel charges is to study the dynamic characteristics of the plant under different operating conditions. Different perturbation experiments are planned in which the response in one or several reactor parameters due to the perturbation

in another is measured and analyzed. Three different perturbation experiments are proposed for HBWR: (1) the reactivity can be oscillated and the power transfer function obtained; (2) the steam load can be perturbed; and (3) variation in the subcooler power can be introduced.

Determination of the reactor power transfer function for different operating conditions will constitute one of the major experiments during both first and second core power operation. A sinusoidal reactivity oscillator, consisting of a rotor and two stators upon which are mounted suitably shaped cadmium disks, has been designed and constructed for insertion in one of the fuel element positions. The oscillator is capable of frequencies from 0.002 to 10 cycles/sec, and for peak-to-peak amplitude variations from 0.02 to 0.1%. Transfer function measurements will be made in these ranges and correlated with analog computer predictions.

Introduction of step and ramp inputs of reactivity will also be made as part of the study of plant dynamics. A subsize control rod in a thimble will be placed in the reactor for this purpose. Reactivity perturbation experiments will be carried out parametrically in power level, reactor steam pressure, and degree of subcooling.

The effect of perturbations in steam load will also be measured. Unfortunately, it is rather difficult to devise a flexible and safe control over the primary steam flow. The experimental facilities are therefore more limited in this case. For the first core operation a fast-regulating valve is incorporated in the secondary steam line with a flexible control system. The valve can be operated with a critical pressure drop so that the steam load is a direct function of the valve opening. Thus, well-defined step variations in steam load can be introduced in the secondary system. This again will reflect back into the primary system as a somewhat distorted step change.

Step changes in subcooling, as introduced by rapid changes in the position of the subcooler bypass control valve, will also be investigated as to their effect on reactivity and power.

Throughout the power runs, analyses will be made to attempt correlation of boiling flux "noise" with other parameters, with the goal of developing new techniques for predicting limits of stable operation.

Thermodynamic studies are under way at Halden to improve the understanding of boiling dynamics phenomena in HBWR, in order to evaluate the potential of heavy water boiling reactors with respect to the power removal of the reactor core, and to develop instrumentation to measure the thermodynamic parameters in situ.

Four special elements have been ordered under the second fuel charge contract to be used for in-core experiments. These elements are special in the way that they allow a certain degree of mechanical modification, dictated by the development program to be carried out.

According to the present proposal, the following instruments will be incorporated in the design of the instrumented fuel assemblies:

(1) two turbine-type flow meters, located at the entry and the exit of the coolant channel;

(2) three pairs of thermocouples, one located at the lower end of the coolant channel, the second at the lower end of the downcomer space, and the third at the subcooled water inlet;

(3) one "hot wire" flow meter at the subcooled water inlet;

(4) a void meter (unspecified type) at the channel exit; and

(5) two capacitive displacement meters connected to the bottom and top end spacers for the measurement of changes in the canning length, yielding the average fuel surface temperature through correlation with thermal expansion properties.

Operation with the first charge core will also yield information as to the effect of radiation on some of the instruments devised. Prototype turbine flow meters will be tested under reactor conditions. This should give information as to the bearing problem. It is intended to place the instruments in a special channel containing no fuel element, with the flow meters driven by the subcooled water flow. A "hot wire" flow meter will be installed in the same channel. The feasibility of this instrument depends on the method employed to support the wire. In addition, a number of different thermocouple systems will be tested.

A capacitive method for measuring the void fraction in air-water and steam-water flow is being studied jointly with the Christian Michelsens Institute in Bergen, Norway. A void meter based on a nuclear principle is under consideration. Briefly, the meter would reflect the change in the thermal neutron flux incident on the surface of a thermal neutron "black" body, due to changes in the density of the surrounding moderator. The principle, however, seems at this point to be more applicable to light water boiling reactors.

The instrumented fuel assemblies will be installed in the reactor during second fuel charge power operation, and measurements will be carried out concurrently with the reactivity perturbation experiments. Such instrumentation used in connection with the reactivity oscillator will make possible the establishment of transfer functions between important

parameters such as reactivity, power, exit and average voids, and fuel canning temperature, and the experimental results can be correlated with the detailed transfer function model developed.*

An extensive reactor physics program will be carried out on the second fuel charge elements in the Swedish RO reactor and in the Norwegian Zero Power Facility, NORA. The measurements, which will be carried out parametrically in fuel-to-moderator ratio, will involve buckling, temperature effects, void effects, and intracell flux distributions. Twenty-five fuel elements will be available for this program. This is sufficient for running some of the experiments with a critical reactor containing only second-charge fuel elements. Some of the less reactive lattices to be studied will require the experiments to be carried out in the subcritical state, or by using the substitution method. These experiments will to some extent shorten the low-power program with the second charge, as they will be carried out concurrently with the necessary second-charge engineering modification program.

The heat transfer and hydrodynamic characteristics of the second fuel charge elements will be studied in a high-pressure boiling heat transfer loop at the Technical University of Eindhoven, Holland. A maximum power of 1.1 Mw is available in the loop, which will enable extrapolation far beyond the limit in HBWR (the HBWR central channel will deliver 370 kw when the reactor is operating at 20 Mw).

C. Engineering

During the period of first-charge operation of the reactor, engineering activities will be concentrated upon design and procurement of components to be installed for second-charge operation. These components will be installed during an extended shutdown in the latter part of 1961. Major items in this category include:

(1) second charge fuel elements (currently in fabrication);

(2) magnetic jack control drive mechanisms and associated electrical equipment (components currently in procurement);

(3) second steam transformer (currently under detailed design by the vendor);

(4) intermediate closed secondary loop;

(5) primary system safety valves;

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*HPR-5.

(6) enlarged D_2O-purification system;

(7) revised top shield and plug handling coffin; and

(8) modified fuel coffin for second-charge elements.

During second-charge power operation the engineering effort will be directed toward functional evaluation of the system from the standpoint of future improvement.

Appendix A

PROJECT STAFF

Personnel are sponsored for the Halden Project staff by the signatory organizations to the Halden Agreement, and are appointed by the Norwegian Institutt for Atomenergi, usually for 18-month periods. Salaries are paid from international project funds.

A few long-term guest scientists, working with the project staff, are also in attendance in Halden. Such persons have been assigned to Halden with the approval of the Institutt, but continue to be paid by their regular employers.

The staff reports to Dr. G. Randers, Director of the Norwegian Institutt for Atomenergi.

Toward the end of 1960, the Project reached the point at which the contracts of the earliest foreign staff members began to expire, and there has consequently been a considerable turnover of personnel in recent months. Table V lists the Halden Graduate Staff as of August 15, 1960, and is representative of the organization during most of the author's period of assignment.

Table V

HALDEN GRADUATE STAFF STRUCTURE

(as of August 15, 1960)

Kåsa, O. R., Project Manager (N)[a]

Fromm, L. W., General Consultant (USA)[b]

Ager-Hanssen, H., Chief of Research (N)			Lien, K. P., Deputy Project Manager & Chief of Operations (N)			
Reactor Physics	Reactor Dynamics	Thermo-dynamics	Mechanical Engineering	Experimental Program	Reactor Operation	Electrical Engineering
Blomberg, P. (S) Grp. Leader	Rydell, N. (S) Grp. Leader	Asyee, J. (NL) Grp. Leader	Jamne, E. (N) Grp. Leader	Brown, C. (GB) Grp. Leader	Strand, R. (N) Grp. Leader	Braun, H. (D) Grp. Leader
Bryhn-Ingebrigtsen, K. (N)	Christensen, P.I.C. (DK)	Buis, H. (NL)	Lüscher, O. (CH)	Enger, F. (N)	Wullum, T. (N)	Buclin, J. P. (CH)
Højerup, C. F. (DK)	Freycenon, J. (F)	Schenk, K. (A)	Lyng, M. (N)	Knudsen, K. (N)		Lucas, H. (F)
Smidt-Olsen, H. (N)[b]	Husebye, J. (N)[b]		Ottersen, U. (N)	Swanenburg, R. J. (NL)		
Sokolowski, K. E. (S)	Moshuus, T. (N)		Rose, R. (N)	Vapaavaori, O. (SF)		
Standal, N. (N)	Schmidl, H. (A)		Skaardal, R. (N)	Øvreeide, M. (N)		
Sutter, A. (CH)[b]						

Vik, T., Office Manager (N)

Firing, J. A., Tech. Secretary (N)

Jahren, K., Health Physicist (N)

[a] A Austria; CH Switzerland; D Germany[c]; DK Denmark; F France[c]; GB United Kingdon; N Norway; NL Netherlands[c]; S Sweden; SF Finland; USA United States

[b] Guest Scientist

[c] Sponsored by EURATOM

Table VI lists the staff as of December 15, 1960, and is representative of the organization which should continue for at least the next year. The national affiliations of staff members are indicated in parentheses.

Table VII lists the members of the Halden Committee and Halden Technical Group as of the latest meetings of those groups prior to December 15, 1960. The representatives of the associated parties other than the USAEC are also listed.

Table VI

HALDEN GRADUATE STAFF STRUCTURE

(as of December 15, 1960)

Kåsa, O. R., Project Manager (N)[a]

Ager-Hanssen, H., Chief of Research (N)

Reactor Physics	Reactor Dynamics	Thermo-dynamics
Standal, N. (N) Group Leader	Schmidt, H. (A) Group Leader	Asyee, J. (NL) Group Leader
Bryhn-Ingebrigtsen, K. (N)	Christensen, P.I.C. (DK)	Bowring, R. (GB)
Habermann, F. (NL)	Freycenon, J. (F)	Buis, H. (NL)
Smidt-Olsen, H. (N)[b]	Hellsten, A. (SF)	Schenk, K. (A)
Sokolowski, K. E. (S)	Husebye, J. (N)[b]	
Tosi, V. (I)	Moshuus, T. (N)	
	Åkerhielm, F. (S)	

Lien, K. P., Deputy Project Manager and Chief of Operations

Mechanical Engineering	Electrical Engineering	Chemistry	Experimental Operation	Plant Operation
Jamne, E. (N) Group Leader	Braun, H. (D) Group Leader	Lyng, M. (N) Group Leader	Vapaavuori, O. (SF) Group Leader	Strand, R. (N) Group Leader
Dalen, E. (N)	Buclin, J. P. (CH)	Küng, A. (CH)	Firing, J. A. (N)	Knudsen, K. (N)
Delves, P. (GB)	Lucas, H. (F)		Siverts, J. (N)	
Ottersen, V. (N)			Sutter, A. (CH)[b]	
Skaardal, R. (N)			Swanenburg, R. J. (NL)	
			Wullum, T. (N)	
			Øvreeide, M. (N)	

Vik, T., Office Manager (N)

Enger, F., Technical Secretary (N)

Jahren, K., Sr. Health Physicist (N)

Brattebø, A., Health Physicist (N)

[a] A Austria; CH Switzerland; D Germany[c]; DK Denmark; F France[c]; GB United Kingdom; I Italy[c]; N Norway; NL Netherlands[c]; S Sweden; SF Finland

[b] Guest Scientist

[c] Sponsored by EURATOM

Table VII

HALDEN COMMITTEE AND HALDEN TECHNICAL GROUP MEMBERS

(as of December 15, 1960)

Country	Halden Committee	Halden Technical Group
Austria	Dr. R. Renner	Dr. M. J. Higatsberger, Vice-Chairman
Denmark	Mr. P. Loft	Mr. P. L. Ølgaard
EURATOM	Dr. J. Gueron	Dr. M. Bogaardt, Chairman
Norway	Dr. G. Randers	Mr. E. Jansen
Sweden	Dr. S. Eklund	Dr. G. Holte
Switzerland	Dr. W. Zunti, Vice-Chairman	Dr. G. Psarofaghis
United Kingdom	Sir Basil Schonland, Chairman	Mr. J. E. R. Holmes
	Representatives of Associated Parties	
Finland	Mr. M. Mutru Mr. J. Mäkinen	Mr. S.-O. Hultin
ENEA	Mr. P. Huet Mr. E. Saeland Mr. J. L. Weinstein	Mr. L. W. Boxer

APPENDIX B

LIST OF HALDEN REPORTS

The Halden Project originates four series of reports and memoranda, having purposes and distributions as indicated below. With a very few exceptions in the HP-Memo category only, all documents are published in English.

A. Halden Project Reports ("HRP-")

Halden Project Reports, bearing "HRP-" numbers and bound in light green covers, constitute the official publications of the project. These are available for worldwide distribution and are intended for full library reference as open literature. All technical data obtained by the Project eventually will be released in this form.

To date, seven Halden Project Reports have been published or are in the process of publication. The HBWR Safeguard Evaluation Report, HPR-1, however, has not been issued publicly as a result of a management decision to follow British custom in this instance.

HPR-1 H. D. Brown, V. O. Eriksen, G. Jenssen, K. P. Lien, L. H. Prytz and R. Strand, Safeguard Evaluation of the Halden Boiling Water Reactor (December, 1958) (not publicly released).

HPR-2 N. Hidle and O. Dahl, The Halden Boiling Water Reactor (January, 1959). Also published in the Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva (1958), Vol. 9, p. 255.

HPR-3 H. Ager-Hanssen, G. Apelqvist, P. Blomberg, K. Bryhn-Ingebrigtsen, C. F. Højerup, G. Ingram, and N. Standal, HBWR Physics Experiments (January, 1960).

HPR-4 Heinz Braun, Calibration of Reactivity Effects at Low Power for the Natural Uranium Charge of HBWR (March, 1960).

HPR-5 H. Schmidl, G. Ambrosini, N. Rydell and O. Vapaavuori, A Transfer Function Model of the HBWR Plant (March, 1960).

HPR-6 G. Bernander, P. E. Blomberg, A. Drageseth, C. F. Højerup, and G. Ingram, HBWR Void Experiments (June, 1960).

HPR-7 H. Ager-Hanssen and V. O. Eriksen, The Halden Project Experimental Programme (in process). Originally a paper presented at the European Atomic Energy Society Enlarged Symposium on Heterogeneous Heavy Water Power Reactors, Mallorca, Spain, October 10-14, 1960.

B. Halden Internal Reports ("HIR-")

Halden Internal Reports, bearing "HIR-" numbers and bound in grey covers with red binding, were originally intended for circulation only among the project staff and the Institute staff at the Kjeller Research Establishment. The reports are preliminary in nature, usually reviewing a phase of the work which requires later verification or which will become part of a major subject for a Halden Project Report. It is the intention that all appropriate material initially issued in Halden Internal Reports eventually will be reissued or incorporated into Halden Project Reports.

The Halden Internal Reports are not to be placed in libraries nor to be used as references in open literature. They are sometimes issued to individuals outside the project who have a specific interest, but only where the recipient understands the preliminary nature of the documents. A recent decision has enlarged the distribution to include Halden Technical Group members and representatives, but only with the understanding that the reports are for the members use and considered circulation only and are not to be placed in libraries or used for general reference.

The author has a complete file of these reports.

Halden Internal Reports issued to date are as follows:

HIR-1 This report has been superseded by HIR-21 below.

HIR-2 J. A. Fleck, Jr., A Model for Boiling Reactor Kinetics (January, 1959).

HIR-3 E. Engebretsen and Å. Kristiansen, A Method for Sealing Plastic Cables for Air Leakage underneath the Outer Coating (February, 1959).

HIR-4 T. J. Barendregt and R. Rose, Heavy Water Purification for the HBWR (February, 1959).

HIR-5 H. Ager-Hanssen and P. Blomberg, Travel Report on Visit to Sweden in Connection with Spike Experiments (April, 1959).

HIR-6 E. Engebretsen, Induced Activity in the D_2O of HBWR (April, 1959).

HIR-7 P. Olstad, Testing of Nuclear Instrumentation (April, 1959).

HIR-8 James A. DeShong, Jr., Flux, Reactivity, Steam Void, and Steam-Water Interlayer Noise Spectrum in the EBWR (April, 1959).

HIR-9 Nils Rydell, Programme for Analogue Computer Studies of HBWR Dynamics (April, 1959).

HIR-10 Nils Rydell, Characteristics of the Magnetic Jack Type Control Rod Drive and Control Station in HBWR (April, 1959).

HIR-11 J. Asyee, Summary of an Experimental Investigation on the Heat Removal of the First Fuel Charge of the HBWR (May, 1959).

HIR-12 Finn H. Enger, Treatment of Sealing Seats for Fuel Elements in the Reactor Lid (May, 1959).

HIR-13 H. Ager-Hanssen, HBWR Critical Experiments (June, 1959).

HIR-14 C. L. Brown, H. Buis and O. Sjaastad, Modifications to the Process Instrumentation of HBWR (June, 1959).

HIR-15 G. Ambrosini and H. Schmidl, Analogue Model of the Reactor Core and Primary Loop Dynamics of the HBWR (August, 1959).

HIR-16 G. Ambrosini and O. Vapaavuori, Analogue Model of the Light Water Circuit Dynamics of the HBWR (October, 1959).

HIR-17 J. Asyee, Subcooler Thermal Tests (October, 1959).

HIR-18 G. Ingram, Some Theoretical Aspects of the Zero Power Transfer Function Measurements (December, 1959).

HIR-19 N. Rydell, Programme for Studies of the HBWR Power Dynamics (January, 1960).

HIR-20 H. Ager-Hanssen and R. J. Swanenburg de Veye, A Compilation of Information on the Second Fuel Charge of the HBWR (January, 1960).

HIR-21 R. Rose, Impact Testing of the Ferritic Steel Components of the HBWR Pressure Vessel Before and After Irradiation (January, 1960). This report supersedes report HIR-1.

HIR-22 G. Bernander, P. E. Blomberg, A. Drageseth, C. F. Højerup and G. Ingram, Summary of Measurements and Calculations of the Reactivity Controlled by Voids in the Halden Boiling Water Reactor (March, 1960).

HIR-23 C. L. Brown and F. H. Enger, The Commissioning Tests on the HBWR, Spring 1959 (March, 1960).

HIR-24 P. la Cour Christensen and K. Solberg, A Nonlinear Model of the Hydrodynamics of HBWR (August, 1960).

HIR-25 O. Vapaavuori, Power Limitations of the HBWR with the First Fuel Charge (August, 1960).

HIR-26 N. Rydell, Some Aspects of Heavy Water Boiling Reactor Dynamics (October, 1960). Originally a paper presented at the European Atomic Energy Society Enlarged Symposium on Heterogeneous Heavy Water Power Reactors, Mallorca, Spain, October 10-14, 1960.

HIR-27 O. Vapaavuori, Fission Product Poisoning of HBWR (October, 1960).

HIR-28 E. Engebretsen, Environmental and Associated Surveys at HBWR (November, 1960).

HIR-29 P. la Cour Christensen, A Nonlinear Analogue Model of HBWR (November, 1960).

HIR-30 N. Rydell and W. Serigstad, Low Level Discriminator (December, 1960).

C. HRP-Documents

Documents bearing "HRP-" numbers are those prepared by the Project Management specifically for the attention of signatories, Halden Committee members, and/or Halden Technical Group members. Included in this category are the quarterly technical progress reports,* budget reports and proposals, meeting announcements and records, and summaries of particular subjects prepared as introductory material for discussions at meetings.

D. HP-Memos

The HP-Memo series is the open forum of strictly internal memoranda in which the Halden staff is encouraged to disseminate preliminary data for the cognizance and comment of other staff members, and to express ideas and proposals for experiments, plant changes, etc. Free use of this medium is promoted by severely restricting distribution. HP-Memos are not circulated outside the project and Kjeller staffs except in answer to a specific question raised by a Halden Technical Group member. Even in this case, a HRP- document is usually prepared and circulated to answer such questions.

*Future plans include issuing quarterly technical progress reports as HPR- reports.

REFERENCES

References to HPR- and HIR- reports will be found in the lists of such reports given in Appendix B. Other references are as follows:

1. V. O. Eriksen, Void Effects in Boiling Water Reactors, Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva (1958) Vol. 11, p. 466.
2. E. S. Beckjord, Dynamic Analysis of Natural Circulation Boiling Water Power Reactors, ANL-5799 (March 1958).
3. J. A. DeShong, Jr., Power Transfer Functions Obtained Using a Sinusoidal Reactivity Driving Function, ANL-5798 (Jan. 1958).
4. J. A. DeShong, Jr., and W. C. Lipinski, Analysis of Experimental Power-Reactivity Feedback Transfer Functions for a Natural Circulation Boiling Water Reactor, ANL-5850 (July 1958).
5. J. A. Fleck, Jr., The Dynamic Behaviour of Boiling Water Reactors, J. Nucl. Eng., 11 Part A, 114-131 (Feb. 1960).
6. E. Andersen and O.Aspelund, Exponential Experiments with Uranium Oxide Clusters in Heavy Water, op. cit. Ref. 1, Vol. 12, p. 374.
7. R. Persson et al., Exponential Experiments in Heavy Water Natural Uranium Metal and Oxide Lattices, ibid., Vol. 12 p. 364.
8. J. M. West et al., The Experimental Boiling Water Reactor, ANL-5607 (May 1957), p. 60.
9. N. G. Aamodt, Underground Location of a Nuclear Reactor, op. cit. Ref. 1, Vol. 11, p. 92.
10. T. Leardini, Research on Containment Features of Cavern for Underground Location of Reactors, SADE Conference, Rome, June 1959.

Price $1.75 . *Available from the Office of Technical Services, Department of Commerce, Washington 25, D.C.*