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TRIP REPORT - VISIT OF US GAS-COOLED REACTOR TEAM TO UK - JULY 7-18, 1958

PART 1

GENERAL DISCUSSIONS ON GAS-COOLED REACTOR DESIGNS

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TRIP REPORT - VISIT OF US GAS-COOLED REACTOR TEAM TO UK-
July 7 - 18, 1958 - Part I
(Previously prepared and issued in Washington
with no identifying number)

GENERAL DISCUSSIONS ON GAS-COOLED REACTOR DESIGNS

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CONTENTS

	Page
Introduction	1
Summary	1
UKAEA Staff - Risley and Culcheth	4
Fuel Assembly and Performance	4
Physics	5
Pressure Vessel	6
Discussion of Kaiser Engineers - ACF Industries' Design	6
Heat Transfer Group (Windscale)	8
High Temperature Gas-Cooled Reactor (Harwell)	9
Nuclear Power Plant Company	10
Atomic Energy Industry - John Thompson	13
English Electric	14
General Electric Company	15
Dowty Nucleonics, Ltd	16
US Gas-Cooled Reactor Team Members and Their Itinerary	Appendix A

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GENERAL DISCUSSION ON GAS-COOLED REACTOR DESIGN

INTRODUCTION

The US Gas-Cooled Reactor team (members listed in Appendix A) visited the UK during the period July 7 – 18. The purpose of the visit was to obtain details of the Calder Hall fuel element fabrication process and performance characteristics, and to discuss gas-cooled reactor designs and materials development. Discussions were held with both UKAEA personnel and representatives of industrial firms, including the four consortia now involved in civil power station construction.

The trip was highly successful in that complete Calder Hall fuel fabrication and performance data were obtained, and design discussions with both UKAEA and the industrial groups were conducted in an atmosphere of cooperation and freedom of information exchange.

The information obtained during the visit is reported in three parts. Part I, contained herein, reviews the discussions relating to gas-cooled reactor designs; Part II covers materials development, including uranium dioxide, beryllium, and graphite; and Part III reports the data obtained on Calder Hall fuel element fabrication and performance.

SUMMARY

The discussions of reactor design reflect continued optimism in the UK for the future of gas-cooled graphite-moderated reactors. At the same time, an increased concern with problems of reactor safety was in evidence, undoubtedly a result of the Windscale accident in October 1957. A substantial portion of the AEA's research and development effort has shifted from support of the Calder Hall type reactors to the development of the Advanced Gas-Cooled Reactor (AGR), which is based on the use of slightly enriched UO_2 fuel clad with beryllium. In addition to the AGR program, several of the industrial concerns are devoting increasing efforts to this more advanced reactor type.

[redacted]

At present, their work on advanced systems consists primarily of optimization studies based on AEA quotes for slightly enriched UO_2 fuel rods clad with beryllium. Each of the consortia maintain development laboratories and large research and development staffs - on the order of 200 – 300 persons. Their major current effort is to develop the Calder Hall type of reactor to its ultimate economic potential. Bidding on new reactor plants is quite competitive, particularly since the formation of a new consortium to participate in the nuclear energy industry. Techniques of calculation and experiment now permit a guarantee of reactivity within 0.5% and power output within 5%. There appears to be general cooperation among the consortia and the AEA in areas of materials development and testing. Such items as refinements of fuel cladding fin design, worked out by the individual concerns, are considered proprietary information.

The recent UK decision to modify civil power stations for production of military plutonium apparently does not affect the initial three stations (Bradwell, Berkeley, and Hunterston). Additional charge-discharge capacity will probably be installed in the Hinkley Point reactor. Future reactor designs will be optimized for fuel exposures of about 1000 Mwd/T rather than the previously anticipated figure of 3000 Mwd/T.

A review of the discussions at each site is presented in a later section of this report. A summary of the major items of interest to the current GCPR design effort is as follows:

1. The lowest CO_2 leakage from any of the three operating Calder Hall reactors remains about one ton per day, which can be expressed as 4 – 5% of the system volume per day. Several of the consortia have established one-fourth ton of leakage per day as the design objective for large power plants and feel reasonably confident of achieving this objective. Blower seals are not considered a major problem; trouble spots exist around the stems of large gate valves and the myriad of small instrument connections.
2. The UKAEA and several of the consortia have investigated the problem of the optimum number of UO_2 fuel rods in the channels of enriched reactors. [illegible] analysis indicates the optimum

number of rods is well above seven with either beryllium or stainless steel cladding. There was not a unanimity of opinion among the consortia on this subject, however, It was stressed that the choice of the number of rods has a strong effect on reactor performance and requires careful evaluation.

3. The UKAEA and consortia members strongly endorse the use of graphite sleeves for supporting GCPR fuel elements.
4. The British believe, in general, that fabrication of a pressure vessel of 4-in.-wall thickness may be too great an extrapolation of existing technology. It appears that their primary concern is with the quality of field inspection. Use of a 4-in. wall thickness, as contemplated in the GCPR design, will require either better inspection techniques than have been developed to date in the UK, or else the standard of inspection will be lower than now considered acceptable in the UK. They stressed the inadequacy, in their experience, of extrapolating conventional pressure vessel fabrication techniques to reactor pressure vessels. However, at least one industrial concern stated that as soon as the present 3-in. technology on the civil stations is established, they would plan on 3-1/2-in. thickness, and that subsequently 4-in. thickness appears quite reasonable.
5. The AEA personnel emphasized the importance of providing reserve blower capacity in an experimental reactor to permit taking advantage of future improvements in fuel performance.
6. The UKAEA feels that the Oak Ridge National Laboratory's estimate of 10% fission gas release from the UO_2 fuel rods is overly optimistic. Kaiser Engineers - ACF Industries' assumption of 50% release is closer to the UK analysis, but even this figure is not considered conservative.
7. An additive factor of 50% is a reasonable figure for converting UK reactor construction costs to US costs. One industrial concern estimated the cost of a Hinkley Point reactor to be 156% of the UK cost, if built on a Lake Ontario site.

Additional discussions with the UKAEA and consortia groups would be of considerable value at an appropriate stage of the GCPR design. Maximum benefit from such discussions would probably be obtained if they were held prior to the start of detailed design.

UKAEA STAFF - RISLEY AND CULCHETH

The Industrial Group staff at Risley and Culcheth had made a detailed review of the helium-cooled GCR-2 design by ORNL, and the discussions centered around their comments. Because of the similarity between the GCR-2 and GCPR, most of their comments are applicable to the helium-cooled GCPR design by KE-ACF. The specific areas of discussion were as follows:

Fuel Assembly and Performance

The UK team did not believe the ORNL fuel assembly design was adequate because of the use of notches in the graphite structure as the method of fuel support and because of the lack of torsional rigidity in the 40-in.-long assembly; it was indicated that this arrangement was likely to cause jamming in the channel as well as fretting of the tubes. The sleeve design is recommended when used with compressive loads.

The British suggested that a magnesia or alumina rod be inserted in the central hole of the UO_2 fuel pellets to prevent UO_2 particles from collecting in the bottom of the element. Graphite is considered unsatisfactory because of its reaction with UO_2. It is not clear at present that any rod is required in the UO_2 inner hole as now designed by KE-ACF and ORNL.

The ORNL methods for calculating heat transfer and fuel element performance are considered satisfactory, with the exception of the allowance for fission gas release. According to UK personnel, the use of Westinghouse data results in 10% release after 2800 Mwd/T and 75% after 7000 Mwd/T. The Oak Ridge National Laboratory had assumed 10% release and a maximum lifetime of 15,000 Mwd/T. The KE-ACF design assumes 50% release. The importance of determining

the actual value as a function of time and temperature is obvious, and the current ORNL experimental program is aimed at developing this information.

The degree of subdivision of the enriched UO_2 fuel was discussed. The Risley group seems to favor 19 – 37-rod clusters for the UO_2-beryllium elements, and perhaps 12-, 13-, or 19-rod clusters for the UO_2-stainless steel.

Physics:

The Harwell representative discussed discrepancies which were found in the ORNL physics work:

1. While over-all agreement on initial enrichment was obtained, low ORNL values for resonance capture appear to be compensated for by a low value for neutron yield/absorption.
2. Ten to twenty per cent shorter lifetime is predicted by ORNL than calculated by the UK for a given enrichment.
3. A large discrepancy in the moderator temperature coefficient exists between the UK and ORNL calculations. The comparison is shown below:

	Initial	Irradiated
ORNL	minus 4×10^{-5}	minus 14×10^{-5}
UK	slightly positive	plus 13×10^{-5}

When asked how concerned the UK people were about positive coefficients, it was stated that it has led them to a very serious consideration of a second independent shutdown system; this the only major design influence so far. A large positive graphite coefficient, greater than about $+15 \times 10^{-5}$, could cause a positive power coefficient, in which case an adiabatic power increase causes divergence. A graphite coefficient greater than $+20 \times 10^{-5}$ would probably result in flux shifting instability. (The Calder Hall temperature coefficient for the graphite becomes positive at about 200 Mwd/T due to the build-up of plutonium).

4. A further major discrepancy occurred in the control rod worth calculations. A calculation by Harwell using the super-cell method, which assumes that the rods are black to thermal neutrons and neglects epithermal captures, and which gives close checks on Calder Hall, gave 3 or 4% for the rod worth instead of 14% as in the ORNL report. After evaluating the probable accuracy of both UK and ORNL calculations, it was concluded that the ORNL method gives results too optimistic by a factor of about two.
5. Rod-withdrawal rates in the ORNL design are a factor of 50 higher than in Calder Hall. The UK conclusion was that ORNL must rely far more heavily on the reliability of instrumentation.

Pressure Vessel

The Risley and Culcheth groups summarized their review of the pressure vessel design. The use of 212-B carbon steel is not considered advisable because the relatively high (0.35%) carbon content could lead to carbon segregation and would require a preheat temperature of 200°C.

The AEA representatives pointed out that the ORNL design did not properly account for neutron streaming, and that irradiation-induced embrittlement could proceed rapidly in certain areas affected by this streaming.

The primary objection to increasing the pressure vessel thickness to 4 in. (as is proposed in the KE-ACF design) is the difficulty of achieving adequate sensitivity in x-ray inspection to assure high-quality welds. The usual quality standards for radiographs call for a 1% sensitivity; achieving this becomes more difficult with increasing wall thickness. With present x-ray equipment, the length of exposure becomes prohibitively long.

Before constructing a final 4-in. walled vessel, it would be of great usefulness to see an analogous structure constructed, preferably

by methods which would be used for the fabrication of the reactor vessel (i.e., shop and/or field erection).

In selecting forged materials which would be welded to the vessel, it is desirous to use material of a chemical analysis matching that of the vessel. This is primarily to avoid the problem of mixing and diluting weld-rod materials in order to achieve uniformity of structure.

It was suggested by the UKAEA that test specimens of the pressure vessel material be inserted into the reactor for intermittent sampling. This would be a valuable test program, particularly in the case of a prototype or developmental reactor.

Discussion of Kaiser Engineers - ACF Industries' Design:

In addition to the detailed comments on the ORNL report, some general discussion of the KE-ACF design took place. The current CO_2-cooled prototype was described, and the implications of the recent change to helium coolant were discussed.

The AEA personnel warned against optimism in obtaining leak tightness and said that to obtain 1% leakage per day would require extremely careful design and testing. The current-usage figures for Calder are 4 – 5%/day, of which actual leakage is 1 – 2%. Calder leakage is actually about one ton/day of CO_2 per reactor. Although Calder is not the best design to minimize leakage, considerable effort has been expended to cut the leakage, without much success. Their recommendations to minimize leakage are:

1. Design of the entire system with the requirement of low leakage constantly impressed on the designers.
2. All-welded construction.
3. Avoidance of castings because of porosity.
4. Careful design of all instrument connections.
5. The charge-discharge system will present difficult problems both from "leakage" and "usage" during operation.
6. Axial blower design should be better from leakage viewpoint.
7. Gate valves are a source of large leaks.

8. Vacuum-leak-test techniques should be used. Each major component should be so tested before installation. Superheated steam tests are not sufficient.
9. Double "O" ring flange seals seem satisfactory. Small valves have been a serious source of leakage. Use of bellows-sealed valves is probably justified.

The KE-ACF representatives were asked if the possibility had been considered of having two separate graphite temperature coefficients of reactivity in the design using the graphite sleeve. The answer was given that because of the small percentage of the total graphite in the sleeve, this effect had been assumed unimportant. In this and later discussions at GEC, it developed that this problem had caused concern, and that only after considerable analysis and experimental effort has it been shown that the effect is in fact negligible in the GEC design. Although the effect is almost certainly also negligible in the KE-ACF design, analysis of the problem should be a part of Title II design.

HEAT TRANSFER GROUP (WINDSCALE)

The Windscale heat-transfer group is working on improvement of the Calder Hall fuel element heat transfer performance and development of heat transfer data for cluster elements. The program emphasis is shifting strongly toward cluster work in support of the AGR. The beryllium-clad UO_2 fuel element design uses clusters of un-finned tubes to obtain a high surface-to-volume ratio and to avoid excessive temperatures that could occur in a large oxide fuel rod. According to Mr. Hall, the experimental techniques for investigation of cluster heat transfer performance have been developed, but no experimental results are available at present. The use of smooth tube calculations as currently used by KE-ACF and ORNL for un-finned elements is considered adequate for preliminary design studies; experimental investigation of heat transfer performance in a test rig using air at atmospheric pressure is needed for final design; and a pressurized test apparatus may be desirable for refinement of the data to within a few per cent. The added

precision attainable in a pressurized rig is of questionable value because of uncertainties in ascertaining local perturbations of reactor flux,and hence, heat generation rates.

The technique used at Windscale for investigation of cluster geometries is an extrapolation of the method used for studying heat transfer characteristics of the Calder Hall fuel can surface. The test rig consists of an open-end tube through which air is blown. The fuel cluster is simulated by resistance-heated stainless steel tubes. Temperatures are measured with an internal probe. In the case of the seven-rod cluster currently under test, heat input to the central tube can be varied independently of the remaining six. This refinement is desirable because of the difficulty of predicting central element power in the reactor due to partial shielding by the outer elements.

HIGH TEMPERATURE GAS-COOLED REACTOR (HARWELL)

The High Temperature Gas-Cooled Reactor (HTGC) is under development at Harwell. It is anticipated that the project will be a cooperative program within the Organization for European Economic Cooperation (OEEC), in which the design team will consist of representatives of the participating nations, and research and development work will be distributed among the several countries involved. The HTGC is visualized as the next development step after the AGR (500 – 550°C gas temperature) which is planned for completion in 1961.

The HTGC contemplates an exit gas temperature of about 800°C, a ceramic core, and a slightly contaminated primary circuit. The core consists of fuel-impregnated graphite rods enclosed in loose-fitting graphite sleeves. The coolant (helium or nitrogen) removes heat from the external surface of the sleeves. Fission products are collected from the annulus between the fuel rod and sleeve, and trapped in a charcoal bed. Of major concern is the leakage of fission products into the primary coolant and the disposal of fission products collected in the charcoal bed. The increased British concern with reactor safety was in evidence, and determination of the required number of independent containment circuits for the HTGC is receiving serious attention.

Experimental work in support of the HTGC is being performed in an in-pile loop in the Pluto reactor, and in a high-temperature, zero-energy critical experiment. Some work is in progress to determine methods of removing fission products from the coolant. The Zero Energy Reactor (ZER) is an example of a very high-temperature physics test facility. A sketch of the facility is shown in Fig. 1.

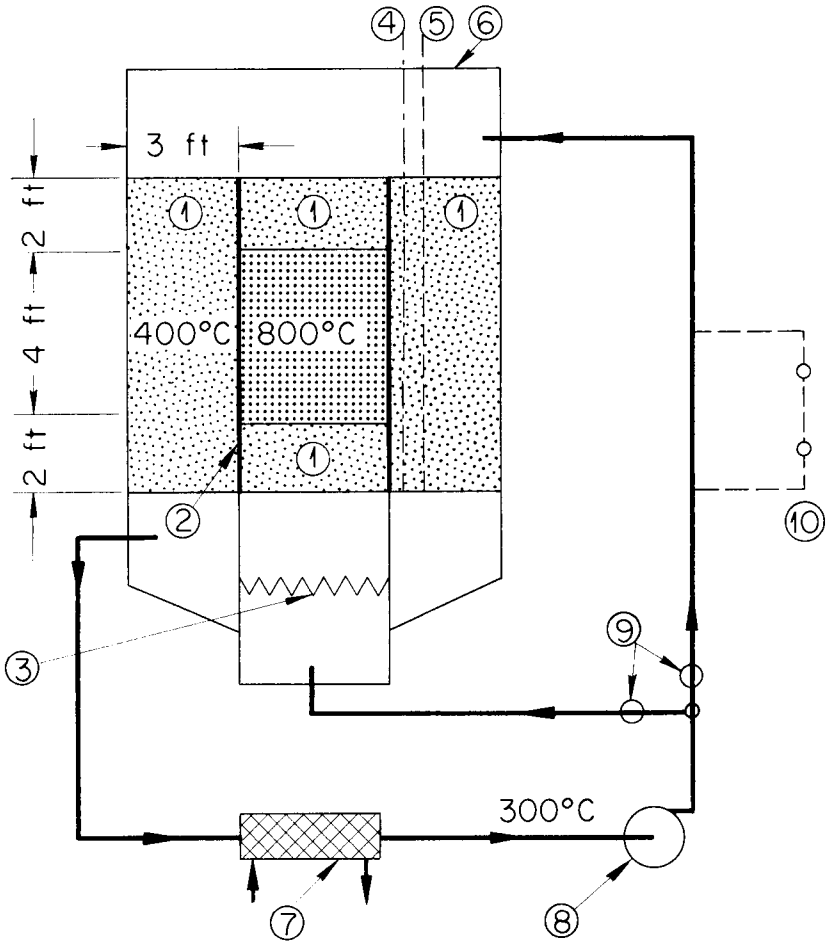
The ZER is contained in a pit which is maintained at a pressure below atmospheric by suction fans. The access doors over the core are separately "hooded" and flooded with argon. The reactor is maintained with a trolly-type rig from this point. Considerable concern is expressed over the possibility of a graphite fire if air leaks into the containment vessel. Interlocks are provided to shut down the reactor if symptoms are abnormal; a stack gas monitor would be typical. It is expected that the ZER will be commissioned in May 1959, and the experimental program should give an insight into the value of the Th-U^{233} cycle. It would be possible to convert the core to test a pebble-bed concept at some later date.

NUCLEAR POWER PLANT COMPANY (NPPC)

The NPPC is responsible for the 300 Mw Bradwell power station. Like the other consortia groups, they have a high interest in more advanced reactors and have initiated design studies on UO_2-fueled reactors using beryllium or stainless steel cladding. Several topics of general interest to gas-cooled reactor designs were discussed, including charge-discharge requirements, fuel rating and lifetime, pressure vessels, and graphite. The development laboratory was visited briefly.

The Bradwell fuel elements are designed for 15% volumetric expansion (due to swelling) at 3000 Mwd/T. The nominal design surface temperature is 435°C; a hot-spot allowance of 50°C is used for estimating maximum surface temperature. Nuclear Power Plant Company studies have considered "shuffling" the fuel elements, but have determined that single-channel location is better unless fuel element life is significantly lower than expected. The flux is flattened by stainless steel rods to a 1.2 ratio of maximum to average. In the smaller Calder Hall reactors, this ratio is about 1.6 to 1.7.

[redacted]
ORNL-LR-DWG 32521



Notes
a) A separate system is provided to detect impurities in purge gas stream
b) Larger quantity of cooler gas mixes with core gas and flows down through side reflectors
c) Coolant gas is N_2
d) Fuel starts as oxide and changed to carbide *in situ*

(1) Graphite reflector
(2) 2 in. of carbon black granules for insulation
(3) Electric heaters
(4) Instrument positions
- 4 telescopic detection chambers
- 2 "DC" Chambers to effect shutdown, separately cooled

(5) Control rod positions
(6) Containment shell is insulated. Gas at low pressure, about 10 in. water
(7) Water cooled heat exchanger
(8) Circulating pump
(9) Metering valves
(10) Clean up system

Fig. 1. Zero Energy Reactor (ZER).

The NPPC representatives commented that the excavation required for the present GCPR building design, with bottom loading, would not be practical for most UK sites because of the high-water table. All of the consortia except GEC are using top loading based on economic studies indicating larger building sizes for bottom loading. For exit-gas temperatures less than 750°F the vertical stacking of metallic uranium fuel elements is satisfactory.

It was indicated by NPPC that prevention of distortion of the channels, due to graphite exposure, requires careful attention. For minimum graphite temperatures of 270 – 280°C the problem is believed to be negligible. It is understood from discussion with others that future UK reactors will be designed for inlet temperatures compatible with a minimum graphite temperature of 260°C (508°F).

Nuclear Power Plant Company commented favorably on the GCPR graphite block design, stating that wastage during machining is an important cost factor, and that the hexagonal block should result in low wastage compared to alternate designs.

The Bradwell pressure vessel is fabricated primarily of 3-in. plate; the structure is strengthened at nozzle penetrations with 4-in. thickness. The material of construction is a silicon-killed steel with less than 0.2% carbon. Design temperature on the vessel is 400°C. As was noted at all the UK discussions, the British question the soundness of using an over-all 4-in. vessel thickness because of the lower standards of inspection that can be obtained with 3-in. thickness. As a general observation, it is believed that thicknesses beyond 3 in. will be adopted by British industry after experience with this thickness is accumulated. The NPPC foresaw serious problems with the GCPR design in cooling the upper pressure vessel section. Their approach is to insulate the vessel with a material such as diatomaceous earth. Flow tests have been conducted in a 1/12-scale model vessel with air; the Reynolds number in the model was approximately 1/80 of that in the reactor.

The development laboratory contained a pressurized-CO_2 heat transfer test rig. It was indicated that the apparatus was designed and put into operation within a six-month period. An atmospheric pressure, "off-the-shelf"

blower was adapted by canning it in a pressure shield. It was stated that the maintenance and replacement of bearings for the off-design operation was economically superior to obtaining a special blower designed for the particular test conditions.

ATOMIC ENERGY INDUSTRY - JOHN THOMPSON (AEI-JT)

This group designed the 300 Mw Berkeley power station. In addition to the design discussions, the construction site was also visited. The AEI-JT representatives stressed the need for accurate determination of reactivity and output because of the highly competitive status of the UK power reactor industry. As an example, it was noted that a 6-in. reduction in diameter of a 40-ft vessel is quite important. It was stated that they now guarantee the reactivity within ± 0.5% and the output within ± 5%.

The use of helium in the UK has been considered, but was discarded because the cost of shipping large amounts was found to increase the final cost of the gas by a factor of ten. The problem of minimizing leakage was again emphasized. It was stated that the loss of CO_2 may be as high as 3 or 4% of the system volume per day. It was pointed out that impurities such as CO_2, O_2, and H_2 in the helium might impose a temperature limitation because of reaction with graphite or cladding materials. Purification requirements will thus be quite stringent.

The Berkeley fuel assembly uses graphite struts to support the fuel elements. Future designs are expected to use graphite sleeves to permit increasing the minimum graphite temperature to 260°C or above. The Berkeley plant uses a dual-pressure steam cycle similar to that of Calder Hall. For reactor outlet temperatures in the range 700 – 800°F, an approach temperature in the steam generator of 20 – 25°F is optimum. Direct comparison between US and UK designs from this standpoint is hazardous because of difference in capital charge rates. Atomic Energy Industry - J.T. believes a single pressure cycle will be used in their Be-UO_2 design; the single pressure cycle is used in the GCPR design.

The AEA pricing schedule for enriched uranium places a larger premium on low enrichment than is the case in this country. This fact provides a

strong incentive for the use of beryllium in the advanced reactor designs using UO_2 fuel. In these designs an outlet-gas temperature of 500°C (932°F) would be used, together with a design can surface temperature of 550°C with a maximum allowable of 600°C. The beryllium-canned fuel would be in the form of 7 – 19-rod clusters. It was stated that the fuel cycle cost with such fuel at an exposure of 5500 Mwd/T would break even with costs of the present metallic fuel at 3000 Mwd/T. From this information, and using $56/kg for the combined cost of uranium and fabrication for the natural uranium elements, it is calculated that the combined fabrication and enriched (0.8% U^{235}) uranium cost for the UO_2 fuel cluster might be in the range $105 – 110/kg U. The average specific power in this design is about 7 Mw/T (average specific power in KE-ACF CO_2 design was 10 Mw/T.

Complete xenon override has been specified for this design because it must be a load-following plant. This implies that during normal operation the reactor must operate with the rods partially inserted, or some fully inserted, or else provision made for inserting special fuel-containing assemblies to override the xenon.

ENGLISH ELECTRIC

This group is responsible for the Hinkley Point reactor, the largest and most advanced of the four commercial stations now underway. This reactor utilizes a spherical pressure vessel with an inside diameter of 67 ft. English Electric believes that for power levels of 100 Mw(E) or greater, spherical vessels are more economical. This view is not shared unanimously by the other consortia. Effective utilization of pressure vessel volume is an important factor in the highly competitive UK reactor industry. Hinkley Point achieves a core volume of 40% of the pressure vessel volume as compared to 28 – 30% in most designs.

Construction cost of the 500 Mw, two-reactor Hinkley Point station is £110 ($310)/kw, not including the initial fuel loading interest during construction or minor site improvements; these latter two items would increase the cost by about £10/kw. Since the beginning of the project in March 1957, a 5 – 6% escalation has occurred. For a single-reactor 200 Mw

station the UK cost would be about £130 – 135 ($370 – 380). A cost differential of about 50% apparently exists between US and UK costs. As a specific case, the cost of the Hinkley Point reactor, built on a Lake Ontario site, was estimated as 156% of the UK cost. In this plant the reactor and steam generators account for 45% of the total cost.

Like the other consortia, English Electric is actively interested in more advanced gas-cooled concepts, particularly the UO_2-beryllium system. They believe beryllium may be used as fuel cladding at surface temperatures up to 570°C (1058°F). The high cost of beryllium indicates recovery of beryllium fuel jackets is essential; costs on the order of $65 – 70/lb for tubes are estimated for quantity production.

GENERAL ELECTRIC COMPANY (GEC)

A brief tour was made of the GEC experimental facilities at Erith, Kent. The primary facilities of interest were the heat transfer laboratory containing a pressurized CO_2 apparatus, materials development laboratory where creep tests on uranium and magnesium alloys were underway, beryllium laboratory, primarily for physical property measurements, analog computor used currently for hazards analysis, and a 1/20-scale model of the Hunterston reactor used for study of flow asymmetries in the core and vessel.

The GEC representatives had made a preliminary evaluation of the safety aspects of the ORNL GCR-2 design. Their comments in general paralleled those made by the Risley staff. The major objection to the GCR-2 design concerned the relatively rapid rod withdrawal with the attendant hazards during reactor start-up.

The Hunterston reactor's fuel-loading system is similar in principle to that of the GCPR. That is, the fuel elements are contained in graphite sleeves, and the charge-discharge (C-D) operation is performed from beneath the reactor. Several design considerations relating to the Hunterston C-D system were discussed. The provision of a spare C-D machine is considered essential for a power plant application: for a two-reactor station, three machines would be provided; for a single reactor station, two machines would be used. The spare machine would undergo an intensive program of planned

maintenance when not in actual use. To reduce operating problems associated with activation and contamination of the machine, those portions of the machine inserted into the reactor are removable for disposal or for decontamination by dry or wet methods. Use of low-Co and low-Mn materials to minimize activation, and use of stainless steel parts where contamination is expected, will contribute to better operating efficiency.

Cooling of fuel elements during discharge is recognized as a rather difficult design problem. The additional pressure drop imposed by the C-D machine can reduce the flow to the point where inadequate cooling of the affected channel is obtained. Conversely, when only one element remains in the tube, the flow may increase to the point where the remaining element is blown out the top of the channel. In the Hunterston scheme, a separate gas supply feeds CO_2 at 75°C into the C-D machine so that more effective cooling is obtained at reduced flow.

General Electric Company will use TV cameras for visual alignment of the C-D machine with the pressure-vessel connection. Mechanical indexing devices are not recommended. An arrangement of hydraulic jacks and heavy springs is used to effect the seal. The Hunterston machines will service three channels per day, either to invert the charge from top to bottom or to reload the channels.

DOWTY NUCLEONICS, LTD.

Dowty Corporation is a large component manufacturer, primarily of specially engineered hydraulic and electronic equipment. The items of interest to the gas-cooled reactor team were gas-bearing compressors and burst-slug-detection equipment.

The technology of gas-bearing compressors obtained by the UKAEA in the manufacture of compressors for the diffusion plant has been made available to UK manufacturers. Dowty is making compressors for use in the Berkeley burst-slug monitoring system. One compressor is now under test at the Dowty plant. The parts of a second which had operated for about 1200 hr were displayed. It had a rating of 25 hp and had undergone 2000 stop-starts

without repairs. It was estimated that a similar unit would sell for about $5,000 and could be delivered in approximately six months.

Dowty is also engineering and manufacturing the electronic and valving system for the Berkeley station. A new system was demonstrated which is comparable to the Calder Hall system, but occupies only one-third of the space of the Calder equipment. At the request of ACF, Dowty had done some preliminary design for the GCPR detection system. The proposed design will be forwarded by Dowty in the near future.

APPENDIX A

US GAS-COOLED REACTOR TEAM MEMBERS AND THEIR ITINERARY

APPENDIX A

US GAS-COOLED REACTOR TEAM MEMBERS AND THEIR ITINERARY

The US Gas-Cooled Reactor Team consisted of the following members:

Name	Organization
U. M. Staebler, Team Leader	AEC, Washington
J. M. Atwood*	AEC, Washington
W. J. Lindsey	AEC, Washington
J. B. Philipson	AEC, Idaho Operations Office
W. J. Larkin*	AEC, Oak Ridge Operations Office
W. D. Manly	Oak Ridge National Laboratory
A. M. Perry*	Oak Ridge National Laboratory
J. J. Cadwell	GE, Hanford Atomic Products Operation
R. E. Nightingale*	GE, Hanford Atomic Products Operation
E. E. Hayes	Savannah River Laboratory, Dupont Company
T. E. Stephens*	Kaiser Engineers
J. K. Davidson*	ACF Industries, Washington

E. L. Brady, AEC Scientific Representative in London, accompanied the team on a part-time basis. The team was split into two groups for the majority of the discussions. Those individuals marked * were concerned primarily with reactor design discussions, while the remainder of the group concerned itself with Calder Hall fuel technology. The discussions relating to materials development involved personnel from both groups.

The itinerary of the team was as follows:

July 7 - London

Orientation session and meeting of AEC representatives with UKAEA officials.

July 8 - Risley, Culcheth

Fuel element and design discussions with UKAEA staff.

July 9 - Risley, Knutsford

Fuel element discussions with Risley staff and design discussions with Nuclear Power Plant Company and AEI-John Thompson; these two groups are respectively responsible for the Bradwell and Berkeley nuclear power stations.

July 10

Tour of UKAEA Springfields plant from ore processing to finished Calder Hall fuel element.

July 11 - Calder Hall - Windscale

Tour of Calder Hall, discussions of fuel performance and examination, discussion of heat transfer aspects of design, and discussion of graphite technology.

July 14 - Harwell

Discussions of UO_2, beryllium, and graphite technology, and general discussion of High Temperature Gas-Cooled Reactor (HTGC).

July 15 - Cheltenham, Whetstone

Discussions with Dowty Nucleonics, Ltd., manufacturers of burst-slug-detection equipment, and English Electric, designers of the Hinkley Point power station.

July 16 - Berkeley

Tour of Berkeley power station construction site.

July 17 - Erith

Discussions with General Electric Company, designers of the Hunterston power station.

July 18 - London

Organization of notes and preliminary drafting of reports. Review of tour with UKAEA officials.