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FAST FLUX
TEST FACILITY
PROJECT REPORT

TRIP REPORT
U. S. FFTF FUELS AND MATERIALS
TEAM TRIP TO THE UNITED KINGDOM
AND FRANCE

T. T. Claudson
R. E. Peterson
J. E. Hanson

December 1967

AEC RESEARCH &
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TRIP REPORT
U.S. FFTF FUELS AND MATERIALS TEAM
TRIP TO THE UNITED KINGDOM AND FRANCE

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TRIP REPORT
U.S. FFTF FUELS AND MATERIALS TEAM
TRIP TO THE UNITED KINGDOM AND FRANCE

T. T. Claudson, J. E. Hanson, R. E. Peterson

INTRODUCTION

The purpose of this trip was to review and determine the latest fast reactor fuels, materials, and core design technology related to the United Kingdom Prototype Fast Reactor (PFR) and Commercial Fast Reactor (CFR) programs which could be of value to the Fast Flux Test Facility (FFTF) project. Of particular interest were those items of information which are required to make early decisions on the FFTF reference fuel, cladding and structural materials, and core design parameters.

Those participating in this team effort were:

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T. T. Claudson	PNL-Metallurgy Department
J. E. Hanson	PNL-FFTF Project
R. E. Peterson	PNL-FFTF Project

Upon arrival in London, the team was joined by Mr. Norman P. Klug, USAEC-London, who accompanied us throughout our visits within the United Kingdom (UK). We were also joined at Risley by Dr. Harry C. Dunn, Overseas Relations, Reactor Group, Risley, who also accompanied the team at our visits to Culcheth and Dounreay.

The schedules of sites visited and the principal contacts are listed below. The agendas of each meeting held during the trip are enclosed as Appendix A to this report.

<u>Date</u>	<u>Site Visited</u>	<u>Contact</u>
July 3-4, 1967	Risley	A. Frame
July 4, 1967	Culcheth	A. Frame
July 5-7, 1967	Dounreay	R. Matthews
July 10-11, 1967	Harwell	R. S. Barnes
July 12, 1967	Fontenay-Aux-Roses	P. Sebillieu
July 17-18, 1967	Cadarache	S. Stachura

DISCUSSION

The following relates the information and data obtained at each site visited. Since much of the data presented to the team has not yet been published by the UK scientists and engineers, these data should not be referenced in any way without the expressed approval by the authors of this report or the originator of the data.

RISLEY - JULY 3, 1967

Dr. J. F. W. Bishop and Mr. A. Frame led the discussion for the Risley staff during the first day. Initial discussion was about the organization of the efforts of all the laboratories in the United Kingdom for the design and development of the PFR fuel. A summary by site is given below.

Risley

At Risley there are three main groups:

1. The Fast Reactor Design Office (FRDO)
2. The Project Coordination Office (PCO)
3. The Central Technical Services Group (CTS).

The relationship is largely informal through internal working committees. Culcheth Laboratories perform tests on clean materials and fuel, sodium corrosion work, and structural materials work. There is no plutonium handling within Culcheth. The project office at Risley is headed by Frame. The fuel work is headed by Bishop at Risley. The reactor engineering laboratory is headed by Paul Evans. The Central Technical Section at Risley supplies technical analytical backup for the Fast Reactor Design Office. The civil engineering group at Risley is on an overlay status to the FRDO, and Dounreay is headed by R. R. Matthews.

Culcheth

At Culcheth, Peter Netley is the senior representative for the PFR and is mainly concerned with the coordination of the fuel cladding program. This includes the testing of

unirradiated and irradiated cladding materials which have no fuel within the clad walls. His responsibilities also include the sodium corrosion on cladding materials and structural components. It was emphasized that the cladding materials under study contain no fuel material.

Springfield

The Springfield Laboratory located 35 miles from Risley has the responsibility of fuel element manufacturing development. They are presently manufacturing driver fuel elements such as the magnox element for various reactors throughout the UK which utilize uranium as a fuel. They have no facilities at present for handling plutonium-bearing fuels. Their support to the PFR is related primarily to two tasks. One is to advise the Fast Reactor Design Office on products specifications for the PFR driver fuel. The other task is to devise and produce for the FRDO process specifications for the fuel. The first of these tasks is within the present fast reactor agreement; the latter is outside the exchange agreement and could not be discussed in detail. At the beginning of the PFR program, the selection and evaluations of fuel material and cladding materials at both the Springfield and Dounreay sites were based primarily on unirradiated material data. Judgment was then used in the selection of such materials for actual reactor use.

Dounreay

The responsibilities at Dounreay may be classified into two areas. First, they are responsible for producing the initial fuel specification and flow sheets for processes which are reasonable for upscaling to a pilot plant and manufacturing plant facility. Their work is conducted only at a laboratory scale of effort. In relation to this, the Springfield Laboratory does hardware design and analysis for fuel. Dounreay essentially produces the fuel as a sub-contractor to

Springfield. Springfield does not assemble the fuel, however, since they have no plutonium-handling facilities. The second area of responsibility at Dounreay is the irradiation of both the fuel and cladding materials. These responsibilities are primarily related to the fabrication and irradiation of single fuel pins which are largely conceived and irradiated at Dounreay. The subassembly program comes from Bishop's staff located at Risley. The postirradiation examinations of the fuel and cladding materials are conducted at Dounreay with responsibilities shared by the staff members at both Culcheth and Dounreay. Dounreay has responsibility primarily for the short-term type mechanical property tests on structural and cladding materials, while Culcheth maintains responsibility for long-term, stress-rupture and creep tests.

Atomic Weapons Research Establishment (AWRE)

The personnel at AWRE near Aldermaston will eventually be responsible for production of the first core for the PFR. Their responsibilities include the initial fuel pin process flow-sheets and the engineering flow-sheets required for the production of fuel in a large throughput plant facility. To conduct these responsibilities, they coordinate the efforts with the laboratory facility at Windscale. They will take the initial Dounreay and Springfield process specifications and combine them at Windscale into a pilot and manufacturing plant facility. This work will be done with cooperation of the production group at Risley. Present schedules call for a freeze of design for the subassembly in January, 1968, with the plant being ready sometime in the late summer of 1968. They feel that production will be initiated sometime in March, 1969.

Harwell

The work now in progress at Harwell is not strictly a part of the PFR project, but supports the project in a general nuclear technology sense. They (Reactor Engineering Laboratory)

feel that more and more work at Harwell will be eventually in support of the PFR. The cladding work will be fitted into the basic studies portion of the PFR project.

Risley-Reactor Engineering Laboratory (REL)

In addition to the general overall research development support located with the Central Technical Services portion of Risley, are the facilities at REL. These facilities are presently carrying out flow studies designed to determine the effects of pin spacing, subassembly design, etc., on the pressure drop across the PFR core. In addition, flowing sodium tests are being conducted which will be described more fully later in this report.

Theoretical Studies

Various theoretical studies are now underway in support of the PFR at both Dounreay and the CTS at Risley. These are primarily people involved with the mathematical analysis of core design and heat transfer, and some of the more theoretical engineering work involving studies related to the stress analysis, etc., on various PFR core components.

Following his description of the work underway at the various UK laboratories, Bishop divided his discussion on fuel pin and fuel subassembly performance into categories of fuel swelling, fission gas release, clad deformation, fuel deformation, clad-fuel interaction, fuel movement, and physical and chemical properties of fuel. A summary of his remarks is given in the following paragraphs.

Fuel Swelling

The design of fuel pins was initially based upon the creep properties of the cladding, stressed by means of either fission gas pressure or swelling of the fuel. The present belief is that the dominant characteristic of the fuel is its

swelling characteristics which can cause failure of the clad. Some concern is being given to the possibility of ratcheting and thermal and stress fatigue. There are some reservations whether such clad failure will be caused by creep or a yielding phenomenon.

Two forms of fuel are being studied in support of the PFR program. These include the Vipac and annular pellet materials. The cladding of the fuel pins is now primarily stainless steel in both the solution treated (ST) and cold worked (CW) conditions. One nickel-base alloy, PE-16, is still in the program, but is further back in the technology. They do not consider PE-16 as a candidate for the first core clad material. The stainless steel utilized in the program is of the 316 variety. Both the M316 and the 316 compositions are being tested.

Of the three subassemblies (Mark I, Mark IB, Mark II) which have now been irradiated and evaluated, Mark I subassembly contained both solution treated and cold worked (20% nominally) material. The solution treatment for both conditions was at 1050 °C. The L (low carbon) grade of stainless contained 0.01 to 0.03 carbon and the standard grade of 316 contained 0.02 to 0.06 carbon. In the Mark I fuel subassembly, both large and small gaps between the fuel and fuel clad were studied with 0.067 in. diameter hole in the middle of the annular pellets. An 80% smear density was achieved in both fuel types. The Mark IB subassembly also contained CW and ST cladding material with both Vipac and annular pellet fuel. The Vipac material was clad in CW stainless, and the pellet fuel pin had both CW and ST cladding. Examination results for both the Mark I and IB subassemblies have been previously reported.

The Mark II subassembly is the latest subassembly examined by the UK. Pins in this subassembly have achieved 7.3 at.% burnup. The cladding in this subassembly was nominally 20% CW and ST 316 stainless steel containing both pellet and Vipac fuel with a 25% PuO_2/UO_2 fuel material. The cladding contained carbon between 0.02 and 0.06 wt%. The diametral clearance between the fuel material and cladding was varied between 2 to 7 mils. This was achieved by a selective assembly of fuel material into the pins. Both pre- and postirradiation characterization of the cladding was achieved. Dr. Bagley indicated that an interim report on the examination of this subassembly will be published by the UK and transmitted to the US as a part of the Fast Reactor Agreement. Indications are that this report will be available near the end of August.

The postirradiation examination of selected pins from the Mark II subassembly showed a considerable range of strain in the clad. Data from both solution treated and cold worked pins indicated that strain values from fuel swelling were more closely grouped in cladding which was ST prior to irradiation. A wider range of strain was found in that material which was CW. The UK felt that the CW data showed a spread of values which was due primarily to material variability and not due to irradiation. They, as yet, do not fully understand the differences found, and further study is now underway. Data plots showing the actual measurements of increased diameter in the pins examined and expressed in percent elongation as a function of the length of the pin were shown. Requests for these data were not honored with the explanation that an interim report previously mentioned would be out in the next six to eight weeks. The data do show, however, approximately 1% maximum diametral strain at the hot end of the pin which was estimated to have operated at approximately 649 °C. There was approximately a 250 to 650 °C temperature range represented over the pin length. The pin itself had a 16 in. fuel

length. The maximum strain had shifted toward the bottom end of the pin as compared to subassembly IB. In addition, the inner pins which ran at a higher temperature (approximately 30 °C higher) showed substantially more strain than did the outer pins of the same subassembly. These data are sketched below as Figures 1a and 1b. The cladding fluence of the pins exposed to 7.3 at.% burnup was estimated to be approximately 1×10^{22} n/cm². The UK favored the M316 grade of stainless steel in the CW condition primarily because of its higher strength properties which restrain fuel swelling to a greater degree than does the ST material. In addition, it has several advantages dealing with the handling of the fuel during fabrication. The ultimate decision for the condition of the material has not yet been made, but indications are that the clad will be in the CW condition.

Fission Gas Release

One-hundred percent gas release is assumed in the design, and so the Fast Reactor Design Office feels that gas release is no problem. The gas release is higher in the vibro fuel than in the pelletized material. In the former, something like 80 to 90% should be expected. Pellet fuel, on the other hand, should be about 50%. The release mechanism for fission gas is not well understood, but they were not especially concerned about the non-release in relation to fuel swelling. At present there is no design theory for including the fuel smear density as a design parameter.

Clad Deformation

The strain criteria of 0.1% allowable strain over the lifetime of the fuel seems also not to be of particular concern to the UK. When asked whether the 1% strain actually measured on the fuel cladding in contrast to the 0.1% design bothered them, they indicated that they were happy

FIGURE 1a. Diametral Strain Mark I-B

FIGURE 1b. Diametral Strain Mark II

to have the 1% strain and not have fuel pin failures. It was evident from our conversations that the UK still have much to learn about the actual swelling characteristics of their fuel material. Some unrestrained fuel swelling studies are being done at Harwell. From a thermal hydraulic and heat transfer standpoint, they consider 1 1/2% maximum strain in the clad as allowable without serious flow restrictions. They have not taken into consideration the fact that the fuel clad has not received the same fluence level associated with a 7.3% burnup that it will receive in actual PFR operation. Therefore, their data are somewhat optimistic. They do feel that they will be within a factor of two of the PFR exposure from their current Dounreay fast reactor irradiation programs on both fuel assemblies and structural and cladding material programs. While they would like better than this, they do not feel too bad about a factor of two extrapolation.

Weldments

The general criteria for welding end caps to the clad is that the clad should not be made weaker by the welding process nor should the weld be weaker than the tube itself. They, therefore, used heavy end caps welded to the thin section clad which eventually give a stress concentration. The new pin design for the PFR, however, contains a swaged and welded closure. No postfabrication heat treatment is given to this swaged end cap design.

Fuel Deformation

The subject of fuel deformation was discussed, and it was apparent that the PFR project has not tried to make swelling studies on unrestrained fuel material since they felt they would not know what to do with the data if they had it. Eric Waite from Harwell has made a theoretical study on swelling and may have some data on unrestrained pin material. They felt that thermal flux was a reasonable start

in obtaining information of fuel material (not cladding) for fast reactor application, although fission product distribution is questionable. In observing fuel material after irradiation, they have observed that the hole in the center of the material has become somewhat smaller as a result of irradiation. Some axial contraction has also been observed. If a gap is left between the fuel and the clad, they have found that the fuel will generally close the gap. They have no data on how long it takes to close the gap. They indicated that the General Electric data showed that 10 to 12 hr were required.

Clad-Fuel Interactions

With respect to clad-fuel interactions, the UK have divided their attention into two areas--that of mechanical interaction and that of chemical interaction. The chemical interaction studies are being conducted at Dounreay, and in general show that 1 to 2 mils intergranular penetration has occurred, but no identification of this interaction has been done. In terms of mechanical interaction, they do not, as yet, have an expression for the strain in the clad as a function of atom-percent burnup of the fuel. They would like an empirical correlation for the strain in the clad as a function of temperature, strain rate, and life of the fuel pin. This is a goal of the PFR program, but it has not yet been attained.

Physical and Chemical Properties

Physical and chemical properties of the fuel are under study at the UK laboratories. The $\int Kd\theta$ values at the start of irradiation were approximately 35 W/cm. They are presently pushing to 60 W/cm where they can expect incipient melting. They feel that at a $\int Kd\theta$ of 35 W/cm they can expect normal operation of the fuel, but above that they are uncertain. One interesting comment about fission product movement was the determination that cesium-137 has been found

to readily redistribute itself. They also have indications that the cesium-137 enters the plenum of the fuel pin. No trouble from fission product plugging has been found. This movement of cesium was later described at Dounreay as a possible contributor to the fuel-clad interaction product that they have observed.

Fabrication of Fuel

The fuel material used in the UK irradiation program is coprecipitated product. They looked at the mechanically blended product, but since their experience has been in coprecipitation, they will stay with this means of providing fuel material. Indications were that Harwell is looking at the sol-gel process and the PFR people will look in the future at Dynapak material. The reference PFR O/M ratio is 1.98 to 2.01.

Final dimensioning of each of the fuel pellets is hoped to be achieved without grinding. However, they are planning to install grinding equipment in the event they need it. With respect to radioactivity of the fuel, they are planning on shielded fabrication techniques based on recycle fuel material. They have run a continuous coprecipitation line at Aldermaston. However, the continuity is limited depending on the transfer of materials in and out of the plant. At present, there are no plans to make reactivity checks of individual subassemblies. They do plan a dimensional "go, no-go" check on the subassemblies. In questioning them about the type of tubing used in the fabrication of the fuel pin, we asked whether welded or seamless tubing material was being considered. The reply was that seamless tubing is presently being used in the irradiation program and that they intend to use seamless tubing in the manufacturing of the actual PFR pin. Their reasoning was that they have more confidence in seamless tubing. The use of seamless

tubing was justified on the bases of availability, quality, and conformance to dimensional requirements for PFR fuel. For these reasons, they feel they will use seamless cladding material. The tubing is made primarily from consumable-arc, vacuum-melted heats. Rejection of tubing has been primarily on mechanical property data such as hardness or yield strength, and not as much on results from their nondestructive testing evaluation. In their present program of screening unirradiated material, they have eliminated the Nimonic 80A and the 20/25 niobium stablized materials. In this program they are now evaluating M316-L, M316, FV-548, and PE-16. The suppliers of their stainless steel tubing are Acclec and Pollock, Old Birmingham; the Technical Director of the Acclec and Pollock Corporation is Dr. John Sawkhill. This tubing manufacturer is a part of the tube investment group.

Review of Fuel Pin and Subassembly Design Criteria

The fuel design criteria for PFR subassemblies is based upon 0.1% strain based upon the secondary creep rates of unirradiated stainless steel. They use the Miller relationship modified to account for creep relaxation. The 0.1% strain related to an approximate hoop strain in the clad during operation of 5,000 to 6,000 psi. The cladding itself is 0.230 in. outside diameter x 0.015 in. wall with 4 in. of breeder fuel in the top of the pin and 18 in. of breeder fuel in the bottom. The fuel column is 36 in. long with three annular stainless steel spacers held in place by crimping the cladding under hydrostatic pressure. A molybdenum filter material is located between the lower breeder fuel section and the gas plenum to allow gas to flow to the plenum (Figure 2) and also to keep particles from falling into the plenum. To date they have found no effect of the crimping of the clad on either the metallurgical structure or gross

Figure 2. PFR Fuel Pin Design

properties of the cladding material. No fast flux irradiation tests have been performed on fuel pins with crimped spacers however. Their present design allows for a 3 mils wastage on the wall thickness of the clad.

The subassembly design is related briefly as follows. The subassembly itself fits into guide tubes resting on stellite alloy spacers. Bars at the bottom of the subassembly support each pin, but are not attached to the pin. In the heat zone, the pins are supported by honeycomb spacers 4 in. apart made of 316 stainless. No hard facing material is between the fuel clad and the spacers. On the top of the subassembly are fewer, but larger, tubes of breeder fuel canned in 316 with wire wrapped fins for coolant mixing. Temperature is monitored on each fuel subassembly at a position approximately 2 in. above the top breeder pin. A 90 to 97 psi pressure drop is expected across the subassembly. A wrapper material around each subassembly is made from 321 stainless. It is 5.6 in. OD across the flat of a hexagonal cross-section by 0.125 in. wall thickness. The selection of 321 stainless was not a critical decision with the UK. They felt only that they had more technology on this material than 304. In many instances, it was learned that they allow an option of either 304 or 321 to the manufacturer of a specific core component. In almost all cases, this option is 321 stainless steel. On the fuel subassembly, the pitch to diameter ratio is approximately 1.26 with the pitch equalling 0.29 in. and the pin 0.230 in. The subassembly is assembled in a horizontal position and shipped as a complete subassembly to the reactor (Figure 3).

The EBR-II type of hydraulic holddown is employed. The lower end of the subassembly will not disengage the coolant channel because it is blocked by the control rod in the center of the cluster. The maximum radial ΔT across the subassembly is 85 °C. Design drawings of the fuel pin design and the subassembly were requested. (These were to be supplied at the conclusion of the trip, but were not received.)

FIGURE 3. PFR Subassembly

Design Evaluation

Dr. Eickhoff next described flow studies conducted at the Risley Engineering Laboratory. Flow pattern studies are conducted in air at five times the full scale subassembly. Mixing studies are performed by using nitrous oxide injection upstream of the subassembly. Air is used in this case to obtain the proper Reynold's number correlations in addition to its convenience for the use of the nitrous oxide detection system. Overall pressure drops on subassemblies are determined in high pressure CO_2 as the viscosity of water is judged to be too high for this work.

Vibrational effects are studied utilizing water and full size subassembly mockups. They get no indication of vibration in the 20 to 25 ft/sec flow expected in the PFR. In addition, they had observed no fretting, galling, or self-welding effects to date. Vibrational mode is detected by use of strain gages placed along the length of tubes to measure both the x and y movement of each tube. Empty tubes are used in testing as well as sand-filled tubes, and they find only a change in vibration frequency in either case but no change in amplitude. The strains are so low that no effect is anticipated due to vibrational effects. Their experiences in sodium have been primarily in a 4 in. sodium loop operating at 650 °C with 20 ft/sec flow past the test specimen. The oxygen content in this sodium loop has been 10 to 15 ppm with low carbon. The test section itself is isothermal, but the loop, ΔT , is approximately 85 °C which duplicates the radial ΔT expected in the PFR subassembly. The loop material is made from either 347 or 321 stainless steel; they were not sure. As a part of their flow studies, experiments were run to determine whether or not sufficient strength in the grid spacers has been designed into the element. Data are

required based upon three criteria: the weight of the assembly, the flow forces, and ratcheting due to temperature differences in the subassembly. They checked this criteria by mocking up grids and pins with various materials which will duplicate the differences in the core thermal expansion expected to give the equivalent stress and strains in the PFR subassemblies. As a result of these studies, they again saw no evidence of fretting and made some minor changes to the grid design.

Shielding Requirements in the PFR Fuel Fabrication Plant

Mr. Bill Kent described some of the shielding requirements expected in the PFR fuel fabrication plant. They felt that an operator should be 4 to 5 ft away and will be working behind a shield face consisting of 4 to 6 in. of concrete with windows and manipulators. Bulk handling of fuel will be limited to 20 to 25 kg. In this way, the shielding requirements will be met by distance and local or portable shielding. This was in view of the fabrication of a fuel manufacturing plant. In pin fabrication, the single pin line will require operators to maintain hands away from the material at a distance of at least 6 in. No shielding will be required in these instances. Twenty pins may be run through in a single batch and in these cases a shielding equivalent to approximately 1/2 in. of lead or a few inches of polythene will be provided. In the subassembly section of the plant, it will be shielded to about 12 in. of polythene and 1 1/4 in. of lead. Again, portable shields will be provided which will move with the subassembly. As a general summary of Kent's discussions, it was obvious that they plan to provide a complete remote handling fuel fabrication line.

Physics

Cross sections being employed in the design of the PFR are based on the British FD-II data set. A new set, FD-III,

is being constructed. Nuclear design calculations are verified in ZEBRA mockups. The mockups are as follows (Figure 4):

- 7A Plutonium in Zone 1 - Graphite Reflector
- 7B Graphite Reflector
- 7C Different Arrangement of Shield
- 7D Different Arrangement of Blanket
- 7E Reaction Rates (Figure 5).

They believe that they are obtaining about 15% accuracy on their control mockup critical tests (Tables I and II). A two-zone enrichment is planned for PFR. The two enrichments employed are 0.21 and 0.28 plutonium-239 fractions for center and outer zones, respectively. The Doppler coefficient is stated as $-9 \times 10^{-3} \pm 30\% T \frac{dk}{dt}$. This value is down by a factor of about two without sodium. The sodium coefficient has been checked against ZED, PR-6, and ZEBRA-6 experiments. The FD-II data were used with Cram and Pert Codes. The results are as follows:

Leakage	$-8.1 \times 10^{-6} \pm 0.08 \Delta K/^{\circ}C$
Central	$+7.1 \times 10^{-6} \pm 1.5$
Total	$-1.0 \times 10^{-6} \pm 2.0.$

A 1% uncertainty in the multiplication constant due to the uncertainty in plutonium-240 composition is estimated. This has been based on analysis. The uncertainty is of little concern at this time, because of much larger uncertainties associated with certain experimental loadings to be placed in PFR.

Neutron flux levels are limited in the peripheral equipment to a thermal flux of 5×10^4 n/cm²-sec. This will result in activation levels which are tolerable. Graphite is used in the radial shielding at an optimum concentration of 66% graphite and 34% stainless steel. The graphite is clad in Cr-Mo steel. The inner three rows are clad with carbon steel. All of these metals are exposed to the primary circuit coolant. The flux level in the inner surface of the shield is 2×10^{13} n/cm²-sec; in

FIGURE 4. ZEBRA Mockup

FIGURE 5. Reaction Rates - ZEBRA Mockup

TABLE I. *Comparison of Calculation and Zebra Experiments, Off-Central B10 Rods*

Wt of B10 (kg)	Radius (cm)	No. of Pins In Rod	Experimental Worth (% dk)	Experiment/Calculation	
				Rod Displacing Al	Rod Displacing Na
1.07 ^a	65.1	19	0.35 ± 0.02	1.13 ± 0.08	1.00 ± 0.08
1.40 ^b	65.1	25	0.46 ± 0.02	1.18 ± 0.06	1.07 ± 0.07
2.83	65.1	25	0.60 ± 0.03	1.03 ± 0.06	0.95 ± 0.06
1.07	69.1	19	0.27 ± 0.01	1.12 ± 0.07	1.04 ± 0.07
1.40	69.1	25	0.35 ± 0.02	1.17 ± 0.09	1.09 ± 0.09
2.15	69.1	19	0.38 ± 0.02	1.06 ± 0.08	1.00 ± 0.07
2.20	69.1	12	0.42 ± 0.01	0.98 ± 0.05	0.93 ± 0.05
2.83	69.1	25	0.47 ± 0.02	1.04 ± 0.06	1.00 ± 0.06
3.49	69.1	19	0.47 ± 0.02	1.02 ± 0.06	0.98 ± 0.06

Notes a The measurement was repeated with the graphite breeder reflector replaced by a steel breeder reflector and gave an identical result.

b The measurement was repeated with four sodium filled elements adjacent to the rod to simulate a vacant position in the P.F.R. lattice. Both calculation and experiment agreed that the sodium had no influence on the worth of the rod.

TABLE II. *Comparison of Calculation and Zebra Experiments, Off-Central Tantalum Rods*

Wt of Ta (kg)	Radius (cm)	Experimental Worth (% dk)	Experiment/Calculation	
			Rod Displacing Al	Rod Displacing Na
96	65.1	0.31 ± 0.02	1.07 ± 0.09	0.94 ± 0.08
125	42.9	0.98 ± 0.05	1.05 ± 0.05	1.01 ± 0.05
96 ^a	42.9	0.84 ± 0.03	1.04 ± 0.04	0.99 ± 0.04
96 ^{a,b,c}	40.0	0.91 ± 0.03	1.04 ± 0.04	-
96 + 96	40.0	1.63 ± 0.08	0.87 ± 0.05	-
	opposite 42.9			
96 + 96	40.0	1.60 ± 0.08	0.98 ± 0.05	-
	adjacent 40.0			

Notes a The values quoted are the average of two experimental results.

b Measuring the rod worth with the reactor critical gave 0.96 ± 0.02% dk.

c The measurement was repeated with four sodium filled elements adjacent to the rod to simulate a vacant position in the P.F.R. lattice. Both calculation and experiment agreed that the sodium had no influence on the worth of the rod.

the outer shield it is 2×10^6 ; at the center of the heat exchanger, the flux is 10^4 . There is no shielding beneath the core, and a total flux of 10^{12} is incident to the diagrid. The flux above 0.1 MeV is 7×10^{10} at the diagrid which is located 9 ft 6 in. below the centerline of the core in a temperature environment of 400 °C. The core peak flux is given as 9×10^{15} n/cm²-sec. The radial neutron distribution factor is 1.23; the axial distribution is 1.21. The neutron flux is down by a factor of two at the core edge.

System Control

The primary and intermediate sodium pumps are ganged. The reactor outlet temperature varies 30 to 40 °C with turbine load. An analog model of the reactor and associated heat removal and power generating systems was constructed. Point kinetics and three axial heat transfer regions were employed in the core. The model incorporated Doppler, sodium moderating and leakage effects, fuel expansion, control rod expansion, and bowing. Coefficients were varied over the present range of uncertainties. No instability problems were encountered in the plant.

In-Core Instrumentation

The principal in-core instrumentation is a failed fuel detection device that operates on the delayed neutron principle. Subassembly outlet coolant samples are conducted through tubing down through the fuel center post carrier and out through the diagrid to the detection equipment. The device uses a multiple valving arrangement with 186 tubing junctions. The valve is now in conceptual design and is subject to further development work.

The signal from the failed fuel detection device is tied to the safety circuit. Boiling detection is a second type of instrumentation being considered; however, this is felt to be developmental at this stage. Subassembly outlet thermocouples were stated to be useful only as monitors and are not tied to the safety circuit but only to an alarm system. Bulk detection

of fission gas in the primary system is also employed by monitoring the gas blanket system.

Instrumentation

The objectives of the core monitoring system are to assure fuel integrity, to prevent propagation within a subassembly, to prevent propagation between subassemblies, and to locate trouble spots. Fuel integrity is assured by determining the hot-spot limit, locating defects and by detection of activity. The items of concern in propagation with a subassembly are fuel blockages, local boiling, and gas release. Propagation between subassemblies is characterized by vaporization (Spert), melt through, and gas propagation.

In attempting to monitor the temperature, the sensitivity is poor and, in general, is not what is desired. Detection of fission products can be accomplished by either delayed neutron detection or fission product detection. The latter is felt to be less desirable since it involves disengagement from the sodium. Sodium boiling can be detected by acoustic methods. The problem of concern is the signal strength relative to the background. This technique has been successfully used on Dounreay Fast Reactor (DFR), but pump noise arising from mechanical pumps in the PFR is definitely a problem. Reactivity measurement is another method that can be used. The background is difficult to predict. At the present time, bubble collapse in sodium and water systems is being investigated; and a boiling noise source experiment is planned for the DFR.

In answer to a question of how their failed fuel detection system might work with vented fuel, they stated that the present system has been designed for a sealed pin. They further stated that they could not say how this system would work with vented fuel, and that vented fuel presented a different set of problems.

Hydraulic Mockups

Hydraulic mockups of the reactor core have been employed to determine flow patterns, pressure drop, and development of the Burst Cartridge Detection (BCD) system. Initial work was carried out on a one-seventeenth scale water model using plexiglass. The Reynold's number was ignored in this experiment. Results from this test indicated a relatively large fountain effect and surface turbulence. This was corrected by employing baffles near the surface connected to the fuel carrier assembly. Gas entrainment was observed in addition, and this dictated the use of a still larger scale model to get closer to the true Fraude and Weber numbers. A one-fifth scale model was then constructed to study the gas entrainment. Vibration problems were also explored in this model. Again, primarily for gas entrainment studies, a still larger five-eighths scale model was employed. The model used only a partial (1/6) segment of the core. The principal difficulty here appeared to be small bubbles of gas that were being swept into the heat exchanger. Illumination of the water in the model through plexiglass windows permitted visual observation of these bubbles. It was concluded that they remained near the surface and were not being swept into the heat exchanger in large quantities. No sodium models are anticipated at this point.

CULCHETH LABORATORY - JULY 4, 1967

A meeting was held with P. Netley, A. Thorley, and I. P. Bell of Culcheth Laboratory to discuss the irradiation effects and sodium technology programs in progress at Culcheth. I. P. Bell first discussed the irradiation program supporting the PFR cladding efforts. Since the Libby-Cockcroft meeting in 1966, the Culcheth irradiation program has emphasized obtaining creep and creep-rupture data on austenitic stainless steel and nickel-base alloys. Their program is in direct support of the PFR with investigations on M316 stainless steel in the solution treated and cold-worked conditions, FV-548, and PE-16. No work is being conducted on AISI 304. Their program will compare the relative effect of irradiation on M316 and FV-548. These latter data are being obtained after an irradiation in the Dounreay fast reactor at irradiation temperatures to 650 °C. This is in contrast with the data of last year which was primarily short-term tensile data after an irradiation DFR at 300 to 350 °C. Table III summarizes the materials and irradiation and test parameters for the program underway at the Culcheth Laboratory.

Fuel Clad and Structural Materials Irradiation Experiments

Uniaxial and biaxial creep-rupture and tensile specimens of M316 and FV-548 alloys have been irradiated and tested at Culcheth and Harwell. Several reports covering work on these alloys, irradiated mostly at temperature to 350 °C, have been previously issued. The data presented below describe the latest results on FV-548 and M316 irradiated at temperatures to 650 °C.

In-reactor creep tests on biaxially-stressed tube specimens were conducted in the DMTR (thermal reactor) at 650 °C and at a stress level of 18,000 psi. The results of these tests are shown in Figure 6. Tests were initiated after

TABLE III. PFR Irradiation Test Program for Structural and Cladding Materials

Material	Condition	Temperatures		Goal Exposure, nvt	Tests	Stress Levels, psi
		Irradiation	Test			
FV-548	1075 °C Solution Treated	DFR { 300 °C 525- 700 °C	650 °C	3 x 10 ²² to 3 x 10 ²³	Tensile Uniaxial and Biaxial Stress Rupture	40-70,000
	6-20% Cold-Worked	DMTR - 650 °C				
M316	1075 °C Solution Treated	DFR { 300 °C 525- 700 °C	650 °C	3 x 10 ²² to 3 x 10 ²³	Tensile Uniaxial and Biaxial Stress Rupture	40-70,000
	6-20% Cold-Worked	DMTR - 650 °C				
PE-16	Solution Treated and Aged	DFR { 300 °C 550- 650 °C	550- 650 °C	1.5 x 10 ²² to 5 x 10 ²²	Tensile Stress-Rupture	Variable

pre-exposing the specimens to three fluence levels, 1.3×10^{19} , 1.7×10^{20} , and 3.1×10^{20} n/cm². The specimens either failed, or the tests were terminated at the various fluence levels shown in Figure 6. Little or no difference was found in the control data between specimens in the as-received or thermally-aged condition. Specimens irradiated at 650 °C showed substantial decreases in strain for any particular test time indicating a decrease in creep rate. Two tests failed at diametral strains of less than 2%, but the data indicate that all tests terminated would have failed at strains lower than the unirradiated control specimens.

Figure 7 shows the results of 650 °C creep-rupture tests on cold-worked FV-548 irradiated at 300 °C to a fluence level of 2.2×10^{22} n/cm² in the DFR. Stress levels of 40,000 and 45,000 psi were evaluated before and after irradiation. The data show that at a stress level of 45,000 psi little difference exists in the fracture strain due to irradiation. There is, however, a large reduction in rupture life in the irradiated material which exhibited almost no evidence of steady-state creep. The irradiated specimen tested at 40,000 psi fractured after 1.5% strain. Specimens irradiated and tested at both stress levels showed evidence of higher strain rates than their respective controls, which was in contrast to the in-reactor biaxial stress tests described in the above paragraph.

FIGURE 6. Irradiation Effect on Creep of M316 Tubing (Irradiated in DMTR at 650 °C Deformation at 1050 °C, 1/2 hr Hoop Stress 18,000 psi)

FIGURE 7. Creep-Rupture Properties of 10% Cold-worked FV-548 at 650 °C [Irradiation 300 °C 2.2×10^{22} n/cm⁻². 1050 °C ST +10% CW (ASTM 9)]

No direct comparison or correlation can be made between these two tests, however, because of the many differences in test conditions.

The pre- and postirradiated 550 °C creep-rupture properties of 20% cold-worked FV-548 and M316 materials irradiated at 300 °C in the DFR to a total fluence level of 1.5×10^{22} n/cm² are shown in Figure 8. Stress levels of 60,000 and 70,000 psi were used in these tests and show that for M316, reductions in rupture strains were caused by the irradiation. The reduction in rupture life for M316 was negligible at the 60,000 psi stress level, but the irradiated specimen failed before entering third-stage creep.

The data for FV-548, shown in Figure 8, indicate a large increase in total strain for the irradiated specimen tested at 70,000 psi as compared to the unirradiated control specimen. An increase in creep rate was observed with most of the strain occurring in the third stage of creep indicating mechanical instability. A large reduction in rupture life was also found. At a stress level of 60,000 psi, the higher strain level observed in the irradiated material occurred mostly in the primary stage of creep with the secondary creep rate about equal to that of the control. Neither the irradiated nor the control specimens were run to rupture.

Of particular interest are the data for 20% cold-worked M316 irradiated at 525 °C in the DFR to a total fluence of about 3×10^{22} n/cm² shown in Figure 9. The irradiation temperature is calculated from the DFR operating conditions. Postirradiation examination of the specimens indicated that the irradiation temperature may have been as high as 700 °C. These tests on PFR cladding material indicated that failure occurred in specimens at diametral strains from 1 to 2% depending upon the stress level of the test. Generally, lower strain rates associated with the lower stress levels produced lower failure strain levels.

FIGURE 8. Creep-Rupture Properties of Irradiated M316 and FV-548 (20% CW) at 550 °C (Dose 1.5×10^{22} n/cm² Total 300 °C. M316 1050 °C ST +20% CW FV-548 950 °C ST +20% CW (Fine-grained) ASTM 11-1/2)

FIGURE 9. 550 °C, Postirradiation Tests on 20% CW M316 Tubing
 [Irradiated $\sim 3 \times 10^{22}$ n/cm⁻² at 525 °C Uncertain,
 Could be as High as 700 °C (ASTM 7)]

Similar effects are found in solution-treated M316 cladding as shown in figure 10 which also gives an indication of the effects of irradiation temperature. As compared to the control data, specimens irradiated at about 450 °C showed no effect of the irradiation. A test conducted after irradiations at higher temperatures (possibly 700 °C) shows a marked reduction in diametral strain as a function of test time. Failure occurred at a strain of less than 1% after 800 hr.

A series of tests were also conducted exploring the effects of cold-work on the FV-548 material in both the irradiated and the unirradiated conditions. The temperature of irradiation was 650 °C carried out in the DMTR at a fluence level of 1×10^{20} n/cm² total. Boron content of the material was approximately 20 ppm, and calculations indicated that there was 5.6×10^{-2} cm³/cm³ of helium produced. Cold-worked material to 6, 14, and 26% was studied. Effects of the cold-work were found only in the 26% level where weakening occurred in the tensile properties. No weakening in the material occurred in tension properties of the 6 and 14% cold-worked material.

Long-term thermal aging studies are now in progress at the Culcheth Laboratories on FV-548 and M316 alloys. These tests are being conducted at 650 °C in both the cold-worked and solution treated conditions. Mechanical properties in terms of tensile data and room temperature hardness data are being obtained after various times at temperature. Tensile data have now been gathered after aging temperatures of 1000 hr, and hardness data are now available for materials aged for a period of 10,000 hr. No significant changes in tensile data have been found. However, hardness data have shown a slight decrease after 5000 hr of the 10,000 hr exposure.

FIGURE 10. Irradiation Effect on Postirradiation Creep of M316 Tubing (Deformation at 1050 °C, 1/2 hr Hoop Stress 18,000 psi (ASTM 6 Grain Size))

The FV-548 alloy solution treated at 1150 °C for 60 hr and followed by a 850 °C aging treatment for 3 hr was irradiated in the DMTR. The irradiation temperature was 650 °C, and a total exposure was 1.3×10^{20} . In this case, a total elongation of 14% was obtained as compared to 28% for the control specimens. Electron microscopy of the irradiated specimen showed that a precipitation of niobium carbide had occurred. Based on the present UK technology, they felt that a reasonable maximum operating temperature for the PFR fuel for M316 cold-worked material was 650 to 700 °C. For the FV-548 alloy, they felt a temperature of 750 °C would be reasonable. The maximum fluence expected in the PFR clad is approximately 3×10^{23} n/cm².

There is a small effort now being placed over a long period of time on refractory material at Culcheth. They are looking only at molybdenum, where they could expect no damage from helium generation. The main disadvantage to this material was the neutron cross-section. Vanadium alloys may be looked into in approximately two years. Thorley has tested the Argonne National Laboratory material in sodium at 650 °C where 10 ppm oxygen was in the coolant. Under these conditions the corrosion rates were extremely high. In static sodium they got a layer of reaction product which showed primarily oxygen and carbon contamination. This layer was swept away by flowing sodium, thus giving the high corrosion rates.

The problem of welding was discussed with Netley, Thorley, and Bell. No weld or weld material has been placed in their irradiation or corrosion program. The reason for this was that they felt that no weld was in a severe stress condition in the PFR. From this standpoint, they did not feel that they had sufficient manpower or funds to conduct studies on welded material.

Thorley next described the sodium technology program now underway at Culcheth Laboratories. The tests being conducted

at Culcheth are primarily on M316 at 650 °C in a sodium loop which passes sodium containing 10 ppm oxygen, past the specimens at a velocity of 25 ft/sec. The effect of stress on carburization and corrosion is being determined as well as the effects of both the cold-worked and solution treated stainless steel. One major effect observed was that the cold-worked material had carburized in an intragranular manner, while the solution treated material had carburized intergranularly.

The UK feel confident that both the austenitic and ferritic alloys can be used in the same primary sodium coolant system. The main factor is that the ferritic materials be of a stabilized grade. In the PFR, they are presently planning to use the M316 material for cladding, 321 austenitic stainless steel for most major in-core components such as sub-assembly wrappers, and the vessel itself. Various parts of the shielding material will be clad in both the 2 1/4 Cr-1 Mo-0.5 Ti stabilized ferritic alloy and a mild steel alloy containing about 0.16 to 0.20 carbon which has been stabilized with approximately 1% niobium. These materials are felt to be more stress limited than any limitations being placed upon them by corrosion in sodium environments. The basic philosophy expressed by Thorley is that after a relatively short time in the reactor at temperature of 650 °C all materials on the surface are primarily of one type, i.e., ferrite. Once the ferrite layer has been formed on austenitic alloys, the problem is no longer one of corrosion but one of diffusion. They do have concern, however, in placing both low-nickel austenitic stainless steels and high-nickel base alloys in the same system.

The corrosion mass transfer program at Culcheth also includes the study of what must be done with radioactive corrosion products which collect in other portions of the

system outside the core. They have found that nickel plates out on the cold side of the primary systems, and their data show that it collects primarily in the heat exchanger in regions of low flow. These data are described in TIG-1356 and show that active products should plate out in the heat exchanger where the temperature is approximately 400 °C. To date, the UK data indicate that the position of corrosion product plate-out may not necessarily be in the coldest portion of the primary system, but is a factor of both temperature and velocity. In answer to a question whether or not they felt that irradiation itself affected the corrosion rates, they could see no reason why it should. One other interesting aspect to the UK data was that involving the velocity on corrosion rate. Their data show that, once an initial corrosion rate has been obtained, the velocity term is relatively independent which they feel is due to the saturation of elements such as nickel on the cold portion of the primary system. In this respect, they feel that they can explain the velocity dependence indicated by the General Electric data. In the GE data, an initial effect greater than the initial effect in the Culcheth data was observed which they felt was due to the relative impurity content in the GE loop. The UK loop is cleaned prior to testing; and the GE loop is not, in their opinion. Therefore, the GE loop has a higher impurity content at the initial stages of their tests which gives them an overall higher corrosion term associated with velocity.

DOUNREAY EXPERIMENTAL REACTOR ESTABLISHMENT - JULY 5, 1967

Meetings were held with Mr. R. R. Matthews, Director; Mr. Williams, Head of the Metallurgy Department; and Mr. Phillips, Deputy Head of Reactor Operations; and other members of the Dounreay Fast Reactor (DFR) staff on July 5, 1967. The agenda was set and a tour of the DFR arranged. This tour was relatively short. The reactor was shutdown which allowed the tour to include examination of the reactor area itself inside the containment vessel. Work was progressing on a NaK coolant leak in the vessel. It was later learned that they were reinstrumenting the leak.

The tour was followed by a description of the DFR operating experience and fuel pins, and a historical presentation of the development of these pins through the operating periods of Dounreay starting in 1961 to the present. The DFR achieved 1 MW in 1959. The original fuel material for the DFR was a uranium-molybdenum, 1/2% chromium fuel material, 45% enriched and clad in niobium. The original fuel pins were operated in 1961, but problems in sodium-potassium coolant purity in gas entrainment necessitated a redesign of the fuel. After clean-up of the sodium-potassium coolant by hot trapping the material, they achieved an oxygen level of approximately 30 ppm. The oxide content limited the operating temperature of the reactor due to the niobium can picking up oxygen. By 1962, they had a fairly clean system and achieved an 0.8% burnup on the fuel pins which at that time had a composition of 10 wt% molybdenum, 45% enriched uranium. A maximum operating power level of 30 MW was achieved which was the power limit established due to the oxide level of the coolant. In 1963 they achieved 60 MW, but a redesign of the fuel was required which was primarily an increase in the enrichment of the ^{235}U to 75%. The reactor cycle at this time was a 90-day cycle which was later reduced to approximately 55 days. To get a better temperature

distribution in the core, they changed the fuel design by adding more fissionable material. They are presently operating on a 55-day cycle with the center of the fuel operating at a maximum temperature of 670 °C, and an average temperature of about 650 °C. The limitation on burnup at the present time is based upon clearances in the fuel allowed for swelling. Design criteria are that the fuel must not swell to the point where it contacts the can at any place. They were conservative at a design of 1% burnup and have found that they can now go to about a 3 at.% burnup of heavy atoms at 650 °C. A two-zone core loading is now used in which the center zone achieves 2.2% burnup, and the outer zone 2.9%.

Examination of the DFR fuel has continued throughout the operations of the reactor. Dr. Evans described some of the results of these examinations during the course of the meeting. In order to monitor the performance of the fuel, they have been running accelerated tests on 74% enrichment material leaving variable clearances between the fuel material and the clad. Volume available for swelling in the present fuel is approximately 30 vol%. The initial elements which were examined were found to be in good condition with some surface and interior defects. There was some cracking intergranular and transgranular, as well as evidence of surface melting, but these defects did not seem to affect the performance of the fuel. The melting was thought to occur due to fission gas in the coolant gap between the fuel material and the clad, thereby causing an increase in temperature due to the decrease in thermal conductivity at the gas pocket. This problem was solved by design modifications, which gave a higher velocity of coolant through the gap between the niobium can and the fuel, to wash away any fission gas bubbles which might form in this region. The flow prior to the design modification was approximately 2 ft/sec. This was increased to a flow of 10 ft/sec through the annulus. With

this new flow velocity, no defects have been found on the surface of pins irradiated at 1.8 at.% burnup. Pins irradiated at 2.5 at.% have swelled to the point where the old gap size exists, and under these conditions they find a reoccurrence of the surface defects. The specific power of this fuel pin is approximately 200 W/g. The degree of enrichment has had no apparent effect on the swelling rate, as little difference has been found in either enrichment level which has been examined.

A further modification of the original DFR fuel was required when the reactor was put into service as a test reactor for the PFR. This was caused by the loss of reactivity through the addition of experimental fuel and structural material subassemblies. They, therefore, decreased the molybdenum content to about 7 wt% and found that its performance was very similar to that of 9 wt% alloy. In the higher temperature ranges, the 7 wt% alloy showed a swelling rate slightly greater than the 9 wt% alloy. Studies have indicated that there are approximately three ranges where swelling occurs: a low temperature range, which is less than 500 °C; a high temperature range, which lies between 565 to 675 °C; and a 510 to 565 °C intermediate range. They have related the swelling rates in these various ranges to the metallurgical condition at which the fuel is operating. The gamma phase transforms to the alpha prime phase in the 425 to 525 °C range. They feel that a stronger gamma phase case supports the gamma plus alpha phase; and results seem to indicate that, as more alpha phase is formed, less restraint is afforded the material and more swelling occurs. The swelling rates which have been observed to date are as follows: 3%/at.% burnup in the low temperature range, 6%/at.% burnup in the intermediate range, and 11%/at.% burnup in the high temperature range. Most of these data have been related

in past issues of the Fast Reactor Newsletter. On the fuel material itself, an oxide skin has been found 2 to 3 mils in depth which has been shown to be the effect of the fuel material gettering sodium-potassium coolant. This oxide skin does not flake off at burnups up to 7 to 9% upon cracking. On the 2 1/2 wt% material, some cracking did occur; and a loss of the oxide layer was found as evidence of corrosion and erosion. However, this particular fuel pin was examined when the oxide content in the coolant was high. A later test showed no effects when the coolant was in its present, clean position.

The niobium cladding was examined after operating at a temperature of 520 °C in high oxide coolant. Evidence of oxidation was dramatic in that 20 mils were lost after 165-day operating period. In low oxide coolant at the same temperature, less than 1 mil was observed lost in the same time period. In coolant with 30 to 40 ppm oxygen at 520 °C, an 8 mil loss in clad thickness was found in a week of operation. The only mechanical property tests run on the niobium clad were ring tests. These ring tests found that the ductile/brittle transition temperature for this material was in the range of 120 to 170 °C. The material was, therefore, operating in a ductile condition during reactor operation where the clad temperature was approximately 600 to 650 °C.

Gamma scans have been carried out on fuel at burnups to 2.5% that yield the flux distribution obtained on the fuel. Zirconium and niobium scans appeared normal. However, peaks were observed near the cool ends of the fuel consisting primarily of barium and lanthanum indicating a migration of the volatile fission products. No problems have been observed with niobium at 520 °C clad temperature for 160 days at power.

Operation proceeds until the fuel fills the available void space. Employing the center region of the DFR as an experimental zone resulted in a perturbed temperature distribution which created a limiting temperature condition in the second fueled zone. No other driver fuel types are anticipated for DFR. At the time of this visit, the DFR had been shutdown because of a sodium leak in the reactor vessel as previously mentioned. Since the leak was small and did not penetrate the outer vessel, it was planned to continue operation with the leak; and they had, in fact, operated for a period of about 2 1/2 days just prior to our arrival. Shutdown and installation of a new sodium leak detector was underway during the visit. They expected to resume operation shortly. (Subsequent events have not borne out this optimism.)

An oscillator rod has been used to demonstrate the stability of the DFR. A reactivity meter is also employed and has been found extremely useful for control-rod calibration and worth measurements. Reactor flux noise has been used also in conjunction with analog equipment to obtain cross or auto correlation. Earlier, a frequency component of 6 to 7 cycles/sec with an amplitude of 0.1% of the power was observed.

Variation of the reactor flow from 25% to full flow indicated a variation in the amplitude of this frequency component. It is believed that this is possible due to slight control rod motion created by the down flow in conjunction with an eddy shedding mechanism. In any event, the amplitude is sufficiently small that it does not constitute an operating problem. The principal importance in understanding the cause of the oscillation is in assuring that it will not get worse, or constitute a mechanism for a sudden uncontrolled instability.

Thermocouple noise has also been investigated as a potential method for detecting blocked channels. This technique

employs the detection of the noise created by the fluctuation of the flow over the thermocouple. Although the device is slow with respect to its response to a blocked channel condition, the sensitivity may be substantially greater than simple detection of the bulk temperature. The noise has been calculated and is in good agreement with the experiments in gas performed at REL. Activation analyses have been performed to obtain reaction rates for various fissionable materials in the DFR, particularly in the region of the nickel reflector. The DFR was modeled at Aldermaston in the Vera facility and experimental results agreed with calculation on the ^{235}U fission rate to within $\pm 5\%$.

The first three rows of the reflector are nickel with some experiments included. There are about 100 irradiation experiments in the core. It has been found difficult to introduce instrumentation. They operate using 3 of 12 control rods. Four of the unused rods can be used for instrumented experiments.

Dr. I. R. Birss completed the afternoon discussion with a review of the irradiation program at Dounreay. The center region of the DFR now carries about 50 subassemblies with experiments. An equal number of tests are located in the breeder section of the core. The DFR is deficient in instrumentation capability much in the same way as EBR-II. There are, however, four control rod positions which can be used for thermocouple instrumentation for irradiation experiments. Entry for these facilities is through a port in the shield plug. In addition, there are three boron-10 neutron absorber facilities which can accept instrumented experiments. These are of limited use due to their proximity to control rods giving rise to a variable flux condition.

The best dosimetry to date indicates that the central core flux in DFR is 2.5×10^{15} n/cm²-sec total, most of which is above 1 keV. Heat generation rates resulting from reaction rates of ^{235}U , ^{239}Pu , ^{238}U were 280, 360, and 17 W/g, respectively.

Almost all the fuel experiments in the DFR are fully enriched so that fuel burnups can be achieved at a reasonable heat generation rate. The clad fluence, therefore, is substantially lower than the 3×10^{23} n/cm² expected in the PFR clad. Most experiments cannot be instrumented, thus the flux and temperature must be determined by calculation. Reaction rates are felt to be known to within $\pm 5\%$.

Temperature monitoring is by three methods:

1. Fusible alloys where problems of material compatibility were experienced at temperatures greater than 300 °C,
2. Ferritic steel plugs (templugs) using hardness profiles which are temperature sensitive,
3. SiC plugs which have been found to be the best technique to date.

The use of the SiC plugs (1/8 in. diameter x 1/2 in. long) is based on the principle that the lattice expansion due to radiation damage saturates, and that the saturation length increase is a function of temperature. The method involves a series of either length or lattice parameter measurements after a series of postirradiation anneals (Figure 11). If the last highest temperature was 400 °C, the recovery temperature, as shown in Figure 11, will be the irradiation temperature.

FIGURE 11. SiC Postirradiation Anneal

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R. R. Matthews described the PFR subassemblies now in the DFR under irradiation or which have already been discharged. The basis of the program at Dounreay is the PFR reference fuel pin design. The standard test vehicle contains 77 pins arranged in a rhomboid (diamond) geometry. Each pin contains PuO_2/UO_2 mixed oxide at $\sim 80\%$ smeared density. The $15\% \text{PuO}_2$, $85\% {}^{235}\text{UO}_2$ operates at about 200 to 240 W/g, and is clad in M316 cladding; 0.230 in. outside diameter x 0.015 in. wall. Operating temperatures are at 650 to 700 °C. Some irradiations are planned for $30\% \text{PuO}_2$ material. Both single pin and subassemblies have been irradiated with the total number of pins exceeding 5 at.% burnup in excess of 500. The maximum burnup achieved to date is about 9.8 at.%. Something in the order of 2% failures have been experienced. There have been eight such subassemblies; the descriptions of each are given below:

1. Mark I (Reported in TRG-3032)
Vipac and annular fuel pellets
M316 clad in 20% cold-worked and solution treated condition
Maximum clad operating temperature of 570 °C
5.3 at.% burnup
2. Mark IB (Reported in TRG-3728)
Vipac and annular fuel
20% cold-worked and solution treated M316
Maximum clad temperature 585 °C
7 at.% burnup
3. Mark II (Currently under examination)
Vipac and annular fuel
20% cold-worked M316 clad
Maximum clad temperature 630 °C
7.3 at.% burnup

4. Mark IIA (Currently being nondestructively examined)
Similar to above with
Maximum clad temperature 720 °C
5 at.% burnup to date
5. Mark IIB (Currently in storage)
Both carbide and oxide pins
6 at.% burnup to date
Carbide pins will probably be removed and replaced with oxide.
6. Mark IIC
Oxide pins only with fuel spacers
3.4 at.% burnup to date
Improved grid spacers with variable thickness, etc.
M316 cladding
7. Mark IID
Reference fuel design with 0.020 in. clad (0.240 in. OD)
Fuel spacers used
Recently inserted
8. Mark III (Currently in storage)
Contained larger (size unspecified) pins.

Dr. Ken Swanson gave a detailed description of the Mark II subassembly which had reached a peak burnup of 7.3 at.%. A preliminary report on this examination was promised by the end of August by Matthews. These pins, clad in 20% cold-worked M316 ran at a specific power of 216 W/g with a clad temperature of 630 °C. The fuel material was 25% PuO₂/75% UO₂ with the enrichment 75% ²³⁵U.

The subassembly was shipped to the Dounreay hot cells after a 60- to 80-day cooling period. The subassembly was stored in NaK, shipped in NaK to the hot cells, and stored under argon. The NaK is removed by connecting the subassembly

to hot argon until it reaches a temperature of $>100^{\circ}\text{C}$. Dry steam is then reacted with the NaK, followed by wet steam and several water rinses.

The subassembly is examined according to the following procedures:

1. Radiograph upon receipt of subassembly
2. Clean by steam/argon reaction
3. Radiograph again to check for any fuel washout
4. Detailed measurement of pin spacing, wrappers, pin diameters, etc.

The results of the examination showed that the subassembly wrapper had shrunk in the minor axis direction about 0.010 in. A corresponding increase in the major axis was noted. The pin support grids, made from 0.0075 in. material had distorted due to differential pin movement from center to edge of the subassembly. The grid located third from the bottom had the maximum distortion—dishing about 0.040 in. The grid second from the bottom tore during disassembly. Had the subassembly not distorted, this tearing probably would not have happened. The forces to remove the pins were also measured to give indications of any fretting, galling, or self-welding. Although no such effects were observed, 10 to 20 lb were required in most instances to remove the pins as compared to 5 lb for initial insertion. The worst pins required up to 40 lb.

Examination of the pins included radiography, diametral, and axial measurements. Some bowing was observed which was associated with the 100 to 150°C temperature difference across the pin. Of particular significance was the fuel pin swelling data as measured by diametral changes in the clad. Figures 12, 13, and 14 illustrate the data presented. Copies of these data were requested but to no avail. They will be a part of the report to be issued in August, 1967. High strains were observed in the pins running at the higher 630°C

temperature indicating the weaker creep strength of the cold-worked M316 clad. The location of the maximum strain was nearer the bottom of the pin indicating that maximum swelling occurred close to the maximum temperature region and where the burnup was 75 to 80% of the maximum burnup. These data indicate that the strain in the clad was due to fuel swelling and not pressure caused by fission gas release. Furthermore, in comparing the maximum strain observed of 1% to the 0.1% design allowable, indications are that the fuel swelling characteristics are not well known and that it is substantially greater than expected. This, coupled with the lower fluence level in the clad than can be expected in the DFR, also indicates that the clad performance, while successful to date, is very marginal. One additional piece of information was that pin length increase measurements ranged from 0.027 to 0.051 in. and averaged 0.041 in. Future examinations will include metallography, gas release measurements by piercing the clad, burst tests on fuel cladding, and reinsertion of irradiated pins to achieve additional burnup.

FIGURE 12. Diametral Strain in M316 20% CW PFR Fuel Pins Irradiated in DFR to 7.3 at.% Burnup

FIGURE 13. Number of PFR Fuel Center Pins Achieving Diametral Increases after Irradiation to 7.3 at.% Burnup in the DFR

FIGURE 14. Number of PFR Fuel Outer Pins Achieving Diametral Increases after Irradiation to 7.3 at.% Burnup in the DFR

Dr. D. M. Donaldson discussed the results of single pin irradiations being conducted in support of the PFR program. He limited his discussions to those pins receiving burnups of 7.5 at.% and greater. To date, 21 pins have been irradiated to burnups greater than 7.5 at.%, the highest being just over 10%. These pins have been clad in cold-worked and solution treated M316 and contained both annular pellet and vibro compacted fuel. Fifteen pins have been irradiated in 20% cold-worked M316 material; 6 of these contained annular fuel, and 9 contained vibro compacted fuel. Three pins of each fuel type failed. Six pins were irradiated with M316 solution treated cladding. Two of these contained annular fuel and four contained vibro compacted fuel. Of these, one vibro compacted fuel pin failed. Therefore, a total of 7 pins failed of the 21 irradiated. A general conclusion reached from this series of irradiations was that they should be able to attain an 8 at.% burnup for the PFR fuel pins. Failures of pins containing cold-worked clad will be less predictable than pins clad in solution treated material. Beyond 8 at.% burnup, they are skeptical of success. They have determined that the strain in the clad is primarily due to fuel swelling. Beyond 8 at.% burnup, the solution treated cladding strains appreciably. In addition, they have found intergranular attack probably caused from fission production deposition. Intergranular cracks penetrating approximately one-third of the can wall were observed. Actual analysis of the reaction layer has not yet been conducted. Micro-drilling of the fuel material showed migration of cesium from the center of the fuel to the cladding. They, therefore, inferred that the reaction layer on the clad may be cesium rich.

The experimental single fuel pins were irradiated within the confines of a DFR fuel pin. Several photographs of pins which failed during irradiation were shown. Multiple failures on the same pin were observed and thought to be caused

possibly by fission gas release from the initial failure being trapped between the experimental and DFR fuel pins. Entrainment of this gas would inhibit the transfer of heat, giving rise to a melting condition on the clad of experimental fuel pins. This problem was corrected by orificing the DFR fuel to provide a higher rate of flow past the experimental fuel pin.

The strain patterns determined from postirradiation diametral measurements of the experimental fuel pins were discussed. From these profiles, it was determined that the strain was caused by fuel swelling primarily and not fission gas pressure. Figure 15 shows the strain pattern on a fuel pin irradiated to 7.8 at.% burnup, clad in M316-L solution treated material. This fuel pin operated at a specific power of 152 W/g and exhibited an axial strain of 0.43%. The maximum strain occurred 5 in. from the bottom (hottest) portion of the pin and shows a compromise between temperature and burnup. In general, the maximum strains were observed 2 to 4 in. from the bottom of the pin.

A calculation of the data from several experimental pins is shown in Figure 16. This figure shows a plot of diametral strain as a function of fuel burnup. In general, very little strain is observed in the 3 to 5% burnup region at burnups greater than 5%. Strain in the cladding may approach 2% at 8% burnup. Of two tests, A and B in Figure 16, which failed utilizing cold-worked M316 clad, the pin containing sintered annular pellets failed at a higher strain level than a pin containing vibro compacted fuel material.

From the data in hand, they feel they have a first core PFR fuel. From a fuel performance standpoint, there is very little difference between the vibro compacted and annular pellet fuel. No decision as to the fuel type has yet been made.

FIGURE 15. Strain Profile for M316-L Fuel Cladding Irradiated in DFR to a Burnup of 7.8 at.%

FIGURE 16. Diametral Strain Range as a Function of Fuel Burnup for PFR Single Pins Irradiated in the DFR

Figure 17 shows a curve of percent fission gas release as a function of the mean burnup. The data from both pellet and vibro compacted fuel fell generally on the same curve. However, pellet fuel fell generally below the curve and vibro compacted fuel fell above the curve. The maximum gas release observed on vibro compacted and pellet fuel was 80% and 65%, respectively.

FIGURE 17. *Fission Gas Release for Vipac and Annular Fuel Pellets Irradiated in the DFR*

Mr. Hicks next described some of his work on the theory of fuel swelling based upon the analysis of all the pin data now available in the UK. A numerical analysis of the strain in M316 cladding solution treated averages 1.3% and occurs 5 in. from the bottom end of the pin. Least squares analysis of the data on solution treated pins from Mark I and IB subassemblies gave the following equations

$$\epsilon (\%) = 0.28 \text{ (B.U.)} + 0.45 \frac{T}{100} - 3.04,$$

where $T = ^\circ\text{C}$,

B.U. = %.

A simplified form of this equation can be written

$$\epsilon = 0.083 \text{ (B.U.)} \frac{T}{100} - 1.51 \quad .$$

From these curves, strain rate in strain/at.% burnup can be obtained. Utilizing data from all solution treated pins, insignificant strains are predicted for burnups less than 4%. This agrees well with Donaldson's data. Hicks is now extending his work to the cold-worked cladding material, but has not met with the same complete success. They are presently working on this aspect of the program.

Work is continuing on the development of a physical model of the swelling process of the fuel. This work is reported in TRG-3394. Figure 18 shows the curve developed for strain rate versus fuel temperature and shows that at fuel temperatures less than 1000 $^\circ\text{C}$ the clad does not restrain the fuel. At temperatures greater than 1000 $^\circ\text{C}$, the cladding begins to restrain fuel swelling. The next step in their analysis was to vary clad thickness and to attempt to predict clad strain. To date they have been unable to make these predictions, but are continuing their efforts in this direction. No work is being conducted in an attempt to predict fission gas release.

Their fuel process development work has concentrated solely on coprecipitated mixed oxide in the range of 14 to 50% PuO_2 content. Up to 30% PuO_2 content it is very easy to get single phase material, but it is very difficult to get a single phase above 40% PuO_2 . A typical analysis of their

FIGURE 18. Strain Rate Versus Fuel Temperature for PFR Fuel Pins

plutonium is ^{239}Pu 94.5%, ^{240}Pu 5.1%, and ^{241}Pu 0.4%. Some segregation below 40% PuO_2 has been observed with variations in stoichiometry. Isotopic analysis for low exposure Pu is ^{239}Pu , 94.06 ± 0.1 (at the two sigma level), ^{240}Pu $5.53\% \pm 0.1$, ^{241}Pu $3.9\% \pm 0.025$, ^{242}Pu $0.02\% \pm 0.005$. Physicists want a three sigma level. This is still being discussed between the fuel fabricators, chemists, and the physicists. It was stated that the three sigma level is attainable but it would cost more.

Dimensions on pellets is not a major problem. They have a $\pm 1\%$ tolerance on the diameter as sintered. They want to get away from grinding. Uniformity from batch to batch is not felt to be a problem. They will have 150 to 200 kg batches of liquid and the powder will be characterized before preparing pellets. During the early stages of the fuel process development they had rejections of fuel materials and pellets due to

improper processes. However, now they feel that they have this pretty well controlled and do not anticipate it to be a problem. They do not reject pellets for small chips. A single phase material is specified for the PFR fuel.

Specific questions were sent to the DFR site from Risley concerning fuel fabrication for PFR. Mr. Norman Parkinson's (DFR) answers to these questions are given below:

1. What is predominant reason for rejection of pellets? Dimensions? Homogeneity? Composition? Density?

Answer: No predominant reason for rejection. Our experience was the following:

Dimensions - Diameter was held to $\pm 1\%$ as sintered. There was no criterion on small chips. (No answer was offered by Parkinson whether or not there was an acceptance criterion on pellet cracks.)

(Pu/U + Pu) ratio - Criterion was $\pm 1\%$ about a 15% nominal. There was little cause for rejection with their technique of controlling composition of blended uranium nitrate-plutonium nitrate. There were no incidences of gross segregation during pellet fabrication. (He did not state the minimum size pellet sample examined.)

Density - Criterion was 93.5% T.D. minimum, no upper limit was established. It was possible to get up to 98% T.D. although this requires greater care in process control. Rejection due to density criterion has been minimal after initial development stage.

2. What techniques are used to assure and measure pellet homogeneity and uniformity, void distribution?

Answer: Plutonium distribution is insured at powder stage. Sintered pellets are examined by ceramography.

Some work is being done on electron microprobe and autoradiography techniques at Harwell and Windscale.

3. What is the isotopic composition of the Pu and is this composition expected to change over the years? If so, what are the implications of such a change on (a) ability to fabricate the fuel; (b) physics and safety aspects of the reactor; and (c) the reprocessing of the fuel?

Answer: Typical isotopic contents now being used in tests:

	<u>%</u>	<u>%</u>
^{239}Pu	94.5	93.37
^{240}Pu	5.19	5.88
^{241}Pu	0.421	0.54
^{242}Pu	-	0.22

The composition is likely to change over the years from Magnox Pu, which will be used for PFR, to that at CFR equilibrium.

	<u>From Magnox Reactors, %</u>	<u>CFR Equilibrium Cores, %</u>
^{239}Pu	79.0	42.0
^{240}Pu	16.0	41.4
^{241}Pu	4.2	9.6
^{242}Pu	0.5	7.0

The implications of this change are (a) Fuel fabrication - difficulties expected with Pu content above 40% since it is difficult to meet single phase now specified. (b) Physics and safety aspects - the present physics specification calls for $\pm 1\%$ tolerance on Pu/Pu + U content. (Parkinson was not too familiar with this area.) (c) Reprocessing - this may be an area of concern. High Pu contents (>40%) cannot be easily dissolved by standard techniques.

4. Is the Windscale plant designed to handle high ^{240}Pu (up to say 30%) content fuel?

Answer: Yes, there is no manual handling of plutonium in this plant. The design philosophy for such commercial plants is to work behind sealed, shielded faces. A little extra shielding may be required in going from Magnox fuel to CFR fuel.

5. Would you expect any significant differences in irradiation performance of a high ^{240}Pu fuel? (as compared to your current irradiation results)

Answer: There would be an effect since thermal conductivity decreases causing a temperature increase (^{240}Pu in oxide form replaces ^{238}U in oxide form which has a higher conductivity). There may also be some slight difference in terms of gas release. Also, there may be some plutonium segregation in a thin film at the edge of the pellet.

6. Has any experimental work been done to measure dose rates from high ^{240}Pu fuel? If so, are your results available for discussion?

Answer: No experimental work has been done. Dose rates have been obtained by extrapolation from low ^{240}Pu dose rates although some work is still in process. An uncertainty in shielding calculations of about 1/2 in. to 1 in. of lead exists.

7. Can you give us some information on operation of the coprecipitated fuel line which served as a basis for the Windscale plant - size throughput, output, etc.?

Answer: Glove box space $\sim 4000 \text{ ft}^3$. Throughput $\sim 1 \text{ kg/hr}$ of U-Pu, about 500 kg/mo of U + 15 to 20% Pu. This reduces to about 100-120 kg/mo for irradiation test material.

8. How confident do you feel about the attainable accuracy of Pu isotopic analyses? Do you feel it is a problem in determining the total ^{239}Pu enrichment in the initial core?

Answer: Accuracies for a typical plutonium isotopic distribution:

	Content, %	Accuracy - with two standard deviation
^{239}Pu	94.06	± 0.1
^{240}Pu	5.52	± 0.1
^{241}Pu	0.4	± 0.025
^{242}Pu	0.02	± 0.005
Total	100.00	

In general, the attitude expressed was that any reasonable desired statistical confidence level could be obtained if one is willing to pay for the analysis. Physicists may want 3σ level - can get it if cost can be justified.

9. What techniques are used for Pu assay in sintered pellet materials?

Answer: For plutonium, potentiometric analysis is used - the technique described in Talanta vol. 13, pp. 477-488, 1966. For uranium, specific titrimetric analysis is used - the technique described in Talanta vol. 11, pp. 1203-1211, 1964.

10. How do you plan on obtaining uniform ^{239}Pu , ^{240}Pu content in fuel from batch to batch and subassembly to subassembly?

Answer: By characterization of liquor and powder batch compositions. A typical batch size is about 150 to 200 kg.

11. What are the tolerable impurity levels in the fuel? Which impurities are worst actors from standpoint of fabrication and irradiation performance?

Answer: Iron: Limited to 1000 ppm (test fuel specs.). It is noted that 3000 ppm iron can prevent reaching

high densities during sintering. Irradiation effects due to iron content are not known.

Carbon: Limited to 500 ppm. Irradiation effects due to carbon are not known.

Minor impurities:

<u>Element</u>	<u>Limit</u>	<u>Reason for Limit</u>	<u>Experience in Meeting Limit</u>
Aluminum	400 ppm	For consistency with past material handled	Fairly easy to meet
Calcium	400 ppm	For consistency with past material handled	Fairly easy to meet
Halogens	25 ppm	Possible reaction with clad	No trouble
Gas (primarily nitrogen)	0.04 ml/g fuel	Based on thermal reactor experience of nitrogen with clad	Easily attainable

Other items of interest learned are summarized as follows:

1. One of the considerations in the design of the Windscale plant was the fact that they would have to use industrial labor rather than laboratory technicians for the manufacture of the fuel. This apparently had strong considerations in going to a totally remote plant.
2. They have studied the effect of iron impurities on the manufacturing of the mixed oxide. A paper was published about three years ago which discussed the effects of impurities on mixed oxide fuel properties. He states that at iron contents of about 3%, iron reduces the sintered density when the fuel is sintered in hydrogen.

If the fuel is sintered in CO_2 , an iron oxide-mixed oxide eutectic is formed. They occasionally find high iron contents in the fuel which is attributed to the ball-milling process.

3. Parkinson stated that their coprecipitated powder is very reactive and they heat treat it in hydrogen or CO_2 up to 1000°C to get $2\text{ m}^2/\text{g}$ surface area, which is perfectly stable and better for use with binders. They ball mill, add binder, (Cranko) granulate, and then they either sinter for feed material for Vipac or debond and sinter pellets.
4. Irradiation data are not available to perfectly establish the effect of impurities on the irradiation performance of mixed oxide fuel.
5. Carbon impurity is given as 500 ppm maximum in the PFR fuel. They have seen "lots" of carbon in sintered pellets whose density has been low. The effects of carbon on irradiation performance are unknown.
6. Homogeneity is checked at the powder stage and in sintered pellets by ceramography which includes porosity distribution. They use electron microprobe and autoradiography for plutonium distribution evaluation. Autoradiography was not thought amenable to mass production techniques.
7. Apparently their physicists have specified a $100\text{ }\mu$ maximum particle size for plutonium in the fuel, and a single-phase material is specified for the PFR fuel.

Allister Washington and P. Prock discussed irradiation rigs and temperature detection used in the single pin irradiations. It is learned that the flow rates through the test rigs are about 400 lb/hr, that the temperature uncertainty in these rigs was about 70 °C in the single pin tests and 90 °C in the tri-pin rigs, and that the construction material of the rigs was unknown (possibly 347 or 321 stainless steel).

Experiments were conducted in which the fuel center temperature and fission gas generation were measured. Two rigs 36 in. long, containing 0.23 in. diameter fuel pins of mixed PuO_2 - UO_2 fuel, were used. The first rig, No. 96-1, used a tungsten-rhenium thermocouple surrounded by static sodium in a gas gap. The second rig, No. 96-2, used a center tungsten-rhenium thermocouple and pressure transducer diaphragm to measure the gas pressure change resulting from a change in the resonant frequency of the diaphragm. The thermocouple insulation resistance fell rather drastically in the 96-1 rig over an 80-day period. Failure of the insulation was believed to be due to a metal ceramic seal. The pressure transducer gave erratic results. The thermocouple in the second rig went open on startup, and the pressure transducer of the strain gage type gave no readings at all. The diaphragm type did give some data, however.

Derek Mosedale described recent results from electron microscopy examination of irradiated fuel pin material. In these examinations, large populations of voids were observed of a size and density that could not be explained by helium formation from (n,α) or ^{10}B burnup in the M316L stainless steel examined. An analysis for helium was successfully obtained on the irradiated specimens which roughly agreed with the calculated value. Since the void volume observed was some 50 times that which could be explained by helium formation, it was concluded that the voids observed were gas free.

Specimens for thin film transmission examination were taken at various positions along PFR fuel pins. In one pin, clad in solution-treated M316L the temperature varied from 270 to 570 °C from the top to the bottom of the pin while the fluence went from 5.6×10^{22} to 2.6×10^{22} n/cm². There were no voids at temperatures of 270 to 380 °C. The lowest temperature at which voids were observed was 450 °C. At this temperature, the void density was 47×10^{14} cm⁻³. At 570 °C this void density decreased to 0.26×10^{14} cm⁻³. As the number of voids decreased, the size of each void increased from 135 Å at 450 °C to 490 Å at 570 °C. The void volume (density x void size) peaked at a temperature of 510 °C. At 450 °C the void volume was 0.8, and at 570 °C it was 0.2.

Similar effects were found on 20% cold-worked material, with voids being observed generally at a lower temperature. Voids in the material irradiated in the DFR to exposures of 5 to 7.1×10^{22} n/cm² total fluence were observed in the 370 to 580 °C range. At 370 °C and 580 °C the void density was 4.8×10^{14} cm⁻³ and 3.5×10^{14} cm⁻³, respectively. peak in void density was observed at 525 °C to be 9×10^{14} cm⁻³. The void sizes increase from 68 Å at 370 °C to 212 Å at 580 °C. The void volume ratio (volume percent voids/volume of SS) was 0.015% at 370 °C, peaked at 450 °C at a value of 1.1% and was 0.35% at 580 °C. The lowest temperature at which any voids were observed was at 360 °C on a cold-worked M316L having an exposure of 6.5×10^{22} n/cm² total fluence.

A large population of voids was observed on a M316L pin irradiated to 7.8×10^{22} n/cm² total fast fluence at 580 °C. Here the density of voids was 0.78×10^{14} cm⁻³ having a size of 1100 Å. The void volume in this case was large, 6.8%. Of particular interest was a solution treated M316L specimen irradiated at 640 °C to a total fluence of

6×10^{22} n/cm² which showed no voids. This indicated that voids can either increase in size with increase in temperature or they can be annihilated.

One specimen from a 321 stainless steel wrapper from the Mark IB subassembly was examined, and voids found. This wrapper operated at 480 °C to a total fluence of 4.5×10^{22} n/cm². The void density was 13×10^{14} cm⁻³ and had an average diameter of 112 Å. A void volume of 0.16% was calculated. Annealing studies were conducted on selected specimens of M316L. One specimen, cold-worked 20% before irradiation, which showed voids at 360 °C at an exposure of 6.5×10^{22} n/cm² was annealed at 700 °C for one hour with no effect on the void density or size. After an anneal at 900 °C for one hour, all voids disappeared except what could be accounted for by helium. On a second specimen which showed the 6.8% density decrease, an anneal at 900 °C for one hour changed the voids only slightly, mostly denuding the grain boundary region of voids. At 1200 °C for one hour, the voids increased in size until they could be seen optically.

It was estimated that the contribution of voids to the strain found in the diametral expansion of the cladding was about 66 to 75% in the solution treated material and about 50 to 66% in the cold-worked material.

Following lunch and a presentation by the team to the UK on the FFTF and EBR-II operating status, a short tour of the laboratory facilities was taken. Below are some items of interest learned during the course of the tour.

1. M316L forms massive sigma phase at 700 °C after thermal aging studies. This starts forming after 500 hr. They have not checked the effects of this formation on the material's mechanical properties.

2. Donaldson's group is looking at compatibility problems associated with high temperature spacer program (looking at molybdenum, stainless steel, etc.).
3. The operating efficiency of DFR is now about 66%.
4. Electron microprobe analysis is done on irradiated cladding alloys by use of an AEI machine. They can now get down to aluminum. The machine can analyze carbon and nitrogen by means of a special attachment which they do not now have. They can now look at irradiated material at a maximum radioactivity of ~ 10 R/hr without undue distortion. A cell is under construction to allow them to look at irradiated fuel material.
5. Silicon carbide temperature monitors 1/8 in. OD x 1/2 in. long were made from material supplied by the Morgan Crucible Co. and is Beta SiC 99+% pure, listed as "Crucilite." This SiC method was described earlier at Risley. Dounreay has been able to relate their findings by an equation

$$\text{Log}_{10} G = \frac{430}{T} + 3.06 \quad ,$$

G = growth in ppm,

T = ° absolute .

One must be careful to remember that the SiC length change relates that temperature the specimen saw just previous to reactor shutdown. At Dounreay flux levels this is the last 10^6 sec or so. Alpha SiC should not be used.

6. Mosedale is not doing thermal aging studies on FV-548. They are doing long-term aging studies on M316 in solution treated and cold-worked conditions at temperatures to 700 to 750 °C (results in item 1 above).

A quick tour through the RML cells was taken. Normally, the cells are on air, but they keep one very tight so that they have the capability for going to an argon atmosphere. This particular cell can go to less than 50 ppm oxygen and 10 ppm moisture in one day. The cells do have alpha containment. The manipulators which are not sealed are bagged both on the outside and the inside. The outside bagging is a secondary seal.

The X-ray machine they use for X-raying subassemblies is a 300 kV machine that uses a slotted lead wheel which rotates to keep down the exposure of the film from the subassembly activity.

Subassembly DFR 166-2A was in the cell. All we could see was the wrapper with some of the top end connecting screws removed which looked very clean and bright and shiny.

Profilometer type measurements for cladding outside diameter are not continuous measurements along the pin (they measure OD in 1/2 in. increments). The electric motors which drive the profilometer cause a great deal of interference in the electronics associated with the linear motion transducers. Air motors will be used to eliminate this problem.

From the hot cells, an inspection of the DFR fuel fabrication facility was made. The principal point of interest was the sodium bonding apparatus for bonding each driver fuel pin at 500 °C (there is a sodium-filled annulus between the fuel and the cladding). Eddy current testing is used to check the integrity of the bond. They can detect a 1/16 diameter bubble. Any detectable bubble is cause for rejection.

In the experimental assembly room where a brief dissertation by a Mr. M. G. Veel was given, experimental pin hardware in various stages of fabrication was inspected.

Nicrobraz 50 is used in the reactor core, if it is absolutely necessary. An automatic programmed welding setup, which was very elaborate, is available as well as an electron beam welding machine.

Sodium facilities at Dounreay are being used for investigation of static pool boiling, superheating and boiling noise measurements. Both up and down flow are available. Heat transfer and acoustic noise measurements are carried out with 14 in. long electrically heated pins in flowing sodium of velocities of 3 to 5 ft/sec. The heated pins consist of 1/4 in. OD material with an internal concentric heater and magnesium-oxide insulation (Figure 19).

The heater is capable of operating at a heat flux of 120 W/cm^2 at a temperature of 900°C for 24 hr. The power supply delivers 400 kW at 110 V. The tests are concentrating only on boiling effects. There is no attempt to perform thermal hydraulics tests with this apparatus. In the static pool boiling tests, 100°C superheat maximum has been observed. This is uncorrelated so far. In tests on the acoustic detection device, the signal-to-noise ratio has been examined. A signal-to-noise ratio of 30 dB on the test rig and 30 to 40 dB in the DFR was measured. A 10 dB attenuation along a coolant channel was observed. The device has been capable of picking up bursts in experiments installed in DFR. The delay associated with the signal is a few milliseconds at most. The behavior of boiling water and boiling sodium is believed to be different because of entrained gas in the former.

Test rigs available at Dounreay are as follows:

1. Four-Inch Sodium Loop. This rig started as a prototype of one of the 24 circuits for DFR. Thus, downflow is obtained through the specimen chamber. The rig

FIGURE 19. Heated Pin Configuration

has been used for development of an automatic plugging meter and is currently being used for the boiling noise measurements described above.

2. Flow Calibration Loop. This loop is used for direct calibration of flow in sodium.
3. Twin High Temperature Loop. This loop is capable of 700 °C operation and has a specimen chamber of 9 in. long by 1 in. diameter. It is being used for fission product studies in sodium.
4. One-Quarter Inch Loop. This loop was developed in 1957 and was originally used for rho-meter development and for development of the Blake meter used on DFR as well as other detection devices. The high temperature regions of this loop have been replaced since it was first put into operation.
5. One-Half Inch Loop. This loop is used for oxygen solubility studies and measurements and the development of the PFR primary sodium sampler system.
6. Components Test Rig. This rig was built in 1960 and has a capacity for tests 6 in. in diameter by 7 or 8 ft in length. It has a temperature capability of 650 °C and was built for the development of DFR core components and testing. The loop is also used to investigate small water leaks and hydrogen behavior in sodium/water steam generators. There were also numerous static setups which were not described.

HARWELL - JULY 10-11, 1967

In addition to the US Fast Reactor team, the following UK members were in attendance:

Messrs: R. Bellamy - Section Leader, Irradiation of
Fast Reactor Fuel
I. Everson - Research Reactor Division
Drs: R. S. Barnes - Head, Metallurgy Division
B. R. T. Frost - Group Leader, Advanced Fuels
Irradiation
V. S. Crocker - Research Reactor Division
E. Wait - Chemistry Division
L. E. J. Roberts - Chemistry Division
J. K. Dawson - Reactor Chemistry Division
D. R. Harries - Group Leader, Cladding Studies
Metallurgy Division

Bob Barnes emphasized that the Harwell Laboratories were primarily a basic research establishment with emphasis on programs dealing with fuel swelling; mechanical properties of fuel; mechanisms of embrittlement on cladding materials; new fuel development, such as carbides; and other basic research programs not associated with a particular project. They support the PFR program in only a general manner. Communications with Risley, Culcheth, Dounreay, and other UK establishments are both on a formal and informal basis. They have members on the Atomic Energy Research Establishment committees and subcommittees dealing with fuels and cladding program as chaired by John Bishop at Risley. Harwell uses the DFR by working through the Dounreay safety committees much in the same way as we use the EBR-II. They have, in addition, an internal committee which coordinates the various program activities within the divisions of Harwell. Informal exchanges with other AERE laboratories are also held.

The major interest of the Chemistry Division is in reactor fuels primarily in the preparation of oxides, carbides, and nitrides. They are interested in the chemistry of high burnup

fuel. They are doing some sol-gel studies on a small effort basis. There is presently no coolant chemistry work being done in the Chemistry Division. In the Analytical Chemistry Division, they are interested in gamma scanning methods and burnup determinations of reactor fuels.

The Research Reactor Division is essentially a service division responsible for the maintenance and operation of the DIDO, PLUTO, and BEPO; also, they do rig design for these thermal reactors, plus rig design for the DFR work. The thermal reactors have had their powers increased in order to provide higher fluxes. Plans for a high flux beam reactor design for solid-state physics work are in progress as well as a pulse type reactor. There was a paper presented in Los Alamos last September on this work. Of major interest was the conceptual design of a fast flux materials test reactor now in the initial planning stages.

The rest of the morning was devoted to a presentation by the US Fast Reactor team concerning the goals, objectives, and status of the FFTF project.

In the afternoon, a meeting was held with Don Harries, Peter Pfeil, Jeff Broomfield, and Phil Higgins on the Harwell fuel cladding program. This program started approximately four years ago and was concerned primarily with the austenitic stainless steels. These included the 20/25/1 niobium 316 cold-worked, 316 solution treated, 347, and 321. One nickel-base alloy, PE-16, was also included for the PFR program. To date, the 20/25/1 niobium stainless has been selected for the advanced gas reactor project. No work is being conducted on Type 304 stainless steel. The goal of their program is to form a basis for the understanding of the loss of high temperature ductility in cladding and structural materials. Tensile, creep, fatigue, and tube burst tests are being conducted on materials irradiated in both thermal reactors and

the DFR. Uses of boron at high and low levels of concentrations have been utilized in these studies and have been previously reported. These studies showed that helium in thermal reactor irradiations was the principal cause of the high temperature embrittlement. A model has been developed using helium levels and stress as a criteria for the failure model. Irradiation variables used have been dose, temperature, and metallurgical conditions of the material. Most of the work has been conducted using thermal reactors, but fast reactor data are now being obtained where they are looking at voids which occur in certain temperature ranges as a result of the fast reactor irradiation. In thermal reactor tests, they have achieved fluence levels to 1×10^{21} n/cm² at temperatures up to 700 °C. Their fast reactor irradiation conditions are primarily fluence levels to 5×10^{22} n/cm² at irradiation temperatures in the 500 to 650 °C range. These are being conducted on instrumented rigs utilizing gas gaps to attain irradiation temperatures. Some tests are being conducted using heaters as a means of regulating the irradiation temperature. Postirradiation mechanical property tests are being conducted at the irradiation temperature. While in the past most tests have been short-term tensile tests, more emphasis will be placed on postirradiation creep tests. Their future program activities will include in-pile creep testing, initially in PLUTO and eventually in the DFR. The role of boron in M316 and PE-16 will be emphasized. All postirradiation examinations will include extensive electron microscopy analysis.

Mechanical property tests at Harwell are being conducted with no extensometry on the specimen itself. Uniform tensile elongations are determined directly from the load elongation charts measured by the crosshead motion of a shop-made tensile machine. A discussion was held, concerning the void formations found by thin-film microscopy on M316 clad. This material was

solution treated for one-half hour at 1050 °C and was irradiated at 500 °C $\pm$ 20 °C in the DFR to a fluence level of 4×10^{22} n/cm² total. The gas concentration was calculated to be 4×10^{-6} cm³/cm³ of material and a decrease of 0.8% in material density was measured. The void volume was estimated at 0.05%. This specimen came from a small stress tensile specimen with no stress applied. Postirradiation annealing of this specimen at 750 °C for one hour showed a complete recovery of yield strength but not elongation. Partial recovery of the total elongation was observed and some voids were still present after annealing. When questioned as to what they felt the mechanism of the void formation was, they indicated that no real model has been developed. They felt, however, that it could be caused by: 1) non-equilibrium helium bubbles, 2) internal stress, or 3) another gas such as hydrogen. They have observed that the formation of voids is somewhat dose dependent in the 500 to 550 °C range for materials irradiated between 1 and 5×10^{22} n/cm². They have not observed voids in specimens irradiated at 400 °C even after annealing at 750 °C. In summary, they felt that the actual damage to mechanical properties is due to helium in thermal reactors and the voids in fast reactors. They feel that a void should act identically to a helium bubble. It was their opinion that the voids observed in the material should not be stable and may not form under certain temperature, flux, and fluence conditions.

No data are being gathered at Harwell on the FV-548 and the vanadium-base alloys. Efforts are being conducted at Harwell on PE-16 using essentially the same parameters as described for the M316 work. Both 347 and 321 stainless steels are being studied at a very low level of effort, primarily examining the effect of boron concentration on mechanical properties.

Figures 20, 21, and 22 show the data obtained by Pfeil at Harwell on Type 316 stainless steel irradiated in the DFR as compared to similar material irradiated in thermal reactors at 550, 650, and 750 °C. These data show the total elongation to failure as a function of helium concentration calculated to be in the material as a result of the irradiation. The same general effect is found at each test condition. That is, the fast reactor irradiations cause a lowering of ductility with smaller concentrations of helium. Some effect of irradiation temperature in thermal reactor irradiations may be seen in Figure 22 where, for a constant concentration of helium, higher elongations to failure are found for higher irradiation temperatures. Comparing data for specimens irradiated at similar temperatures but in fast and thermal reactors, it was observed that irradiation in fast reactors caused lower strains.

In fast fluxes, substantial density changes ($\sim 0.8\%$ density decrease in the maximum observed to date for dose of 1.4×10^{22} n/cm²) were found. In a discussion of ductility versus dose, it was pointed out that ductility in a fast flux is less than that anticipated from the thermal flux data and that there has been found very little carbide precipitation in the grain boundaries of fast flux irradiated material.

Mr. B. W. Mott and Dr. G. Long discussed sodium compatibility and corrosion. There is, at present, no corrosion work as such being done at Harwell although some work is being done on nonmetallics in sodium. Carbon, oxygen, nitrogen, and hydrogen will be studied later. Carbon solubility in pure sodium is of interest, but they feel the solubility is extremely small.

The Chemical Engineering Division is conducting studies on the sodium-oxygen system. They are also looking at the safety aspects of sodium boiling. A sodium purification loop that

FIGURE 20. Tensile Ductility of Type 316 Steel As a Function of Helium Concentration (Test Conditions: 550 $^{\circ}\text{C}$, $4.10^{-7} \text{ sec}^{-1}$ Initial Strain Rate.)

FIGURE 21. Tensile Ductility of Type 316 Steel as a Function of Helium Concentration (Test Conditions: 650 $^{\circ}\text{C}$, $4 \cdot 10^{-6} \text{sec}^{-1}$ Initial Strain Rate.)

FIGURE 22. Tensile Ductility of Type 316 Steel as a Function of Helium Concentration (Test Conditions: 750°C , $2 \cdot 10^{-4} \text{ sec}^{-1}$ Initial Strain Rate.)

contains 120 lb sodium is in use in the study of automatic plugging meters, cold traps, and the effect of temperature gradients. Analyses for oxygen are done by the vacuum distillation technique where they are concerned about such analyses as a function of sample size, nickel crucible purity, and time for vacuum distillation.

Boiling experiments underway include an experiment to study the formation of vapor nucleation as a function of surface temperature, surface characteristics, and the oxygen in the sodium. It is intended to use X-ray radiography to determine initiation of vaporization.

The theoretical analytical studies work now in progress includes an analysis of irradiation effects on sodium boiling. No difference in boiling characteristics is anticipated in sodium under irradiation.

Dr. Koski discussed water analogue studies for sodium boiling. His work has not been reported; there is an AERE report due out in about a month or so. They are interested in bubble growth rate. They are using an approximate analog water model in conjunction with high speed photography in this work, and have completed an exact mathematical treatment of this phenomenon which was in good correlation with experiments. Photographs shown indicate vapor slugs growing in a capillary tube. In the slug stage all the liquid is pushed out, but there is a thin film left on the tube, which was indicated by calculations.

The morning of July 11, was spent on tours of the laboratory. Eric Wait described the UK philosophy on thermal versus fast flux irradiation. In general, it is not possible to simulate fast reactor fuel performance in a thermal flux. That is, the fast flux geometry and heat rating can be tested, but the ^{235}U enrichment necessary in the DFR irradiations

yields a different chemical system. The differences in fission yield between ^{235}U and ^{239}Pu fissions are pronounced. The chemistry is better in thermal fluxes since small specimens can be used, and the fuel does not have to be enriched. However, the temperature distribution characteristic in a fast irradiation cannot be reproduced in a thermal flux. Therefore, the evaluation of the gross irradiation behavior of fast reactor fuel can only be studied in a fast flux. It is anticipated that a difference in swelling behavior in ^{235}U enriched fuel will be evident. Also, the oxygen potential of ^{239}Pu is higher than that of ^{235}U . Correlations for high burnup fuel are just starting and mixed oxide fuel irradiated in a fast flux to 10% burnup is available.

A laboratory tour showed the use of micro-manipulators, ultrasonic micro-drills, etc., used for the very, very small specimens. The argon atmosphere in the cells is pure to the extent of 50 ppm oxygen and 50 ppm water.

They are also doing micro-scale studies on highly irradiated carbides. They find that highly irradiated carbide is practically inert to water at 80 °C. They are making carbide potential measurements and found that the carbide releases fission gases easily at a 1000 °C at approximately 0.5% burnup.

A discussion was also held on the DFR rigs, designed and built for Harwell experiments. Part of the discussion centered on the reduction of natural convection circulation in static NaK-filled capsules. They have done a fair amount of elaborate analog electrical work in the analysis of this problem. The capsules in this particular case contained only about 2 or 3 in. of fuel.

A fair degree of success has been achieved with 5 tungsten-26 rhenium, aluminum tungsten-coated oxide tantalum

insulated sheathed thermocouples operated up to 2000 °C in carbide fuel. They have not done any analytical work for enhancing the axial natural circulation of NaK or sodium in static capsules. It was concluded that, in some instances where burnout of the fuel specimens occurred, it was due to the sensitivity of gas solubility in sodium under high radial ΔT 's.

Capsule heaters having a two million heat flux at approximately 3 atm pressure have been operated with no burnup. There is a report (not available) by Bob Forgan, Bill Hobbs, and Cyril Braunton on this subject.

In another laboratory an electrolytic jet etching technique for thinning UO_2 specimens was displayed.

Discussions were held with Arnold Wapum, John Sayres, John Lambert, Alan Fudge, John Davies, and Eric Wait. Wait discussed the contributions of solid fission products to swelling. They have also studied the chemical state of fission products in oxide fuels. Wait passed out material on the elemental composition of irradiated mixed oxide fuel. There is a threefold approach involved in his studies: theoretical analysis, experimental analysis, and synthetic fission product mixture for experiments. The principal conclusions are that there are three main groups of fission products: noble metals in the metallic state; gases; cesium as iodide and bromide. Cesium, rubidium (metal), yttrium, and rare earths are in solid solution.

Zirconium is usually associated with barium. Strontium has not been detected as a segregated phase. They assume that strontium is in the oxide form.

The state of molybdenum depends upon the final oxygen potential. With an initial fuel O/M of 2.00, the molybdenum goes to molybdenum trioxide; at an initial O/M less than 2.0,

the molybdenum stays metallic with some dioxide formed. It is believed the clad will oxidize if the chromium oxide film is not protective and if molybdenum metal is formed.

Allan Fudge then gave us a dissertation on the cesium distribution in four MK-I subassembly pins. The can mid-wall temperature range was in the 485 to 260 °C range. They saw no evidence of cesium in the plenums in any of the fuel pins. The $\int_{T_S}^{T_C} k dt$ was 16 to 24 W/cm. There was a marked radial distribution of fission products in the fuel. The center void was devoid of cesium. Cesium was found on the cladding walls. Cerium was more uniformly distributed with a slight buildup at the edge of the columnar grain growth region. Fudge showed us some color prints obtained from his radial gamma scanning system. It is stated that solid fission products represent at least 50% of the total swelling in fuel.

In a discussion of the basic swelling processes in the fuel, a statement to the effect that at less than 100 Å diameter, bubble swelling is fairly constant at 0.64% ΔV per 1% burnup and 900 °C. At 2% burnup with an anneal at 1599 °C, the bubbles start to agglomerate; this is attributed to Brownian motion. At 10% burnup there was observed 10% swelling, 10^{17} bubbles per cm^3 , 200 Å apart. At high burnup it is expected that coalescence of the bubbles and an increase in swelling will occur.

Results of carbide irradiations in DFR showed swelling which was approximately 1% ΔV /1% burnup which agrees quite well with Strasser's results of thermal flux irradiated carbides. Arc cast carbides were found to swell more rapidly than sintered carbide. There appears to be a possible incubation period of approximately 2% burnup in the arc cast material and 4.5% in the sintered material.

In some stainless steel clad UO_2 irradiations to 8.0 at.% total burnup in DIDO, enhanced sintering during irradiation was found in the high density fuel at 890 to 900 °C at a burnup of 0.85 at.%. At 1.86% burnup gas bubble formation appeared. Gas bubbles were observed at grain boundaries at approximately 3% burnup. At 2% burnup no ΔD of the pins was observed. The fuel clad gap after irradiation to 1% burnup increased. At 2% burnup the gap was closing rapidly and at 5% burnup the gas gap was detectable only with an electron microscope.

John Sayres described the small particle work and the effect of restraint on swelling being done at Harwell. They use 500 μ UO_2 spheres of 92% theoretical density which were restrained by a pyrolytic carbon coating. At 2.4% burnup, less than 1% in the restrained particles was found. At 12.1% burnup the unrestrained particles had fragmented. Gas release in the restrained sphere was less than 0.5%; but after annealing at 1200 °C for a few minutes they obtained a large gas release.

Studies on the mechanical properties of uranium carbide were described by John Hedges. These studies include hardness as a function of alloying and stoichiometry and compressive creep behavior. Cast UC creep behavior appears to be well defined. U-Pu carbide work is planned in about three months time. $(\text{U}, \text{Pu})\text{O}_2$ work in this area is also planned. In a question about oxide versus carbide work, it was indicated that six to seven years ago efforts were placed on the oxide. The carbide work was then built up. They are now initiating work related to basic oxide irradiation performance in deference to the questions the DFR irradiations have raised. The carbide fabrication processes are well established and are amenable to an oxide fabrication plant. In work related to nitride fuels, ^{15}N was used to eliminate absorption cross sections problems.

There has been some evidence of nitrogen penetration in stainless steel cladding grain boundaries. The bulk of nitride work in the UK is at Harwell.

We had a short discussion on the Fast Materials Test Reactor (FMTR) the British are planning. This has been in discussion for about two years in their organization. It would be scheduled to go on line about 1975. It is principally a research reactor and would be for small tests for single irradiations of materials and advanced fuels materials. Harwell had suggested a modification of DFR in order to make it more amenable to testing which was not accepted. Some of the target parameters for the FMTR are: a 10^{16} n/cm²-sec hard spectrum flux, low power, approximately 100 to 150 MW and "adequate" room for users with instrumentation ability. Plutonium carbide core, 70 to 80 liters in volume. The core would be 12 in. high and 20 to 22 in. in diameter and would be sodium cooled. Minimum inlet temperatures of 150 to 180 °C and an outlet temperature of 280 °C are planned. Instrumentation would include capabilities of temperature, flow, in-pile creep experiments, fission gas release transducers, and so on. The testing volume would be less than 10%. Approximately a 0.2 in. diameter carbide pin clad in 316 stainless steel clad would be used. Burnup target for the fuel is 5%. Their parameters include a 0.050-in. gap between pins, 550 W/g heat generation average, 40 psi ΔP , and a 25 ft/sec coolant velocity. This work is principally centered with V. S. Crocker and Iver Everson. It is an interesting concept in that they intend to drop or lower the core to refuel. The control would be by reflectors. This concluded the Harwell visit.

FONTENAY-AUX-ROSES - JULY 12, 1967

The remainder of the trip report was supplied by J. E. Hanson who continued on with the AEC members of the team to France.

During discussion of the agenda it became immediately obvious that J. E. Hanson's planned visit to Cadarache was desirable because most of the fast reactor fuel fabrication work is now done at Cadarache. Sebilleau was not in attendance on July 12th, as he was at Cadarache on this day. He did attend portions of the meetings on the 13th. Meetings were conducted by M. Bussy. The FFTF Project was reviewed. Neal Todreas covered core and reactor design, John Yevic presented the FFTF and AEC organizations, and J. Hanson presented the proposed irradiation testing program for FFTF driver fuel development. Francois Anselin who had spent approximately two years at G.E./Sunnyvale is now back at Fontenay. He is responsible for the postirradiation examination of the Rapsodie driver fuel.

M. Rozenholz on Rapsodie and Phenix Instrumentation

M. Rozenholz presented a description of the instrumentation in Rapsodie and what is planned for Phenix. He stated, in Rapsodie the temperature distribution map fits very well with flow and power distribution calculations (within 1 °C). The thermocouple for each subassembly in Rapsodie is built into a lid which fits over the outlet of the subassembly. There is a small gap for sodium flow to diffuse from the top of the subassembly. Since the thermocouple is in a pocket above the subassembly there is a time lag of about 7 sec. Each subassembly in Rapsodie is orificed. The orifice is located in the top of the subassembly.

The fuel failure instrumentation in Rapsodie includes delayed neutron and cover gas activity monitors. The delayed

neutron monitor system samples sodium from the whole core. They have a sample tap in each outlet pipe for the core flow. There is also a reactivity meter installed in Rapsodie which is felt to be a very good instrument.

In the Phenix reactor they plan to place three thermocouples in each subassembly; to have fuel failure localization instrumentation through sampling from each subassembly for the delayed neutron monitor system. There will also be cover gas activity monitors and a reactivity meter. They also plan to measure the total flow from each pump.

Dr. A. Lallement on Preparation of Mixed Oxide Fuel and its Properties

Work is being done on the mixed oxide phase diagrams and physical properties. Mixed oxide room-temperature phase diagrams and homogeneity have been studied. These data were presented in a Vienna paper in October, 1966. Extension of the measurements into higher temperature ranges for the U, Pu, O ternary phase diagram is planned. Some mixed oxide fuel for irradiation tests is made at Fontenay. Hypostoichiometric mixed oxides are prepared by carbon reduction.

Preparation of Pellets at Fontenay

Two phase formation in fuel with greater than 40% PuO_2 content has been observed. Fuel is prepared by mechanically blending, as it is believed to be a more simple process than that for coprecipitated fuel. Homogeneity is verified through the use of the electron microprobe, ceramography, and alpha autoradiography.

The UO_2 is calcined in argon plus 10% hydrogen at 1000 °C, and the PuO_2 is calcined in air at 700 to 800 °C. This is done in 100 g quantities. Blending time is approximately 4 hr. Following blending, a binder is added, then the fuel is pressed and sintered (1650 °C for 4 hr in argon plus 10% hydrogen). A pellet density of greater than 95% theoretical

is obtained. For low density pellets they add a greater quantity of binder. Particle sizes before blending were not precisely known but the BET was $3 \text{ m}^2/\text{g}$ for the UO_2 and $10 \text{ m}^2/\text{g}$ for the PuO_2 .

Fuel has also been prepared by another method which includes U_3O_8 , UO_3 and PuO_2 . A solid solution of $(\text{Pu}, \text{U})\text{O}_2$ is obtained by sintering. This method requires an additional grinding step. First, the U_3O_8 , UO_3 , and PuO_2 are blended. Then it is sintered, crushed, and ground. The mixture is blended again, fresh powder and binder are added followed by pressing and sintering. Greater than 95% theoretical density is obtained. It is felt that this is an ideal PuO_2 recycling method. In order to get low stoichiometry fuel, two methods were tried: pure hydrogen reduction in a metal furnace (this method is no longer used), and the carbon reduction method with which it is claimed the lower limit of O/M 1.90 can be obtained. It is felt this is not satisfactory because a local mixed oxide-CO phase is also obtained. 1.92 O/M was stated to be easily obtainable. Sintering in vacuum is used in this case. O to M ratio has been correlated with lattice parameter. The data are presented in the attached graph (Figure 23).^{*} It was stated that hyperstoichiometric oxide is not compatible with sodium. In-pile fuel-sodium compatibility experiments are in progress.

Properties of Mixed Oxide Fuel

The discussion included plans and current work. The study of the thermal stability of mixed oxide fuel and its vapor pressure disclosed that there is no decomposition of the fuel at high temperatures. The rig used in this study is now being used for the PuO diagram. $\text{PuO}_{1.5}$ to $\text{PuO}_{1.6}$ is being studied.

^{*} Subsequent communication with Lallement reveals these data were in error.

FIGURE 23. Variation of Parameter of $(U_{0.8}Pu_{0.2})O_{2\pm x}$ as a Function of O/M

Simulation of nuclear heating by high frequency heating to study grain growth and oxygen migration is being pursued.

Some work on synthetic fission products to measure diffusion coefficients has been done.

Their work on properties includes measurements of thermal conductivity, resistivity, magnetic susceptibility, and specific heat.

Two types of thermal conductivity measurements are being made. Room temperature to 1000 °C they used the Angstrom method; from 1000 °C up they are limited by decomposition of the fuels at high temperatures since the measurements are made in vacuum. The thermal conductivity of $(U,Pu)O_{2+x}$ up to 1600 °C has been measured. Lallement released a preliminary plot of the data which is included in this report (Figure 24). The maximum thermal conductivity for hypostoichiometric fuel occurs in the 1.95 to 1.97 O/M range. The specific heat for UO_2 is used to calculate thermal conductivity from the thermal diffusivity measurements (Figure 25). Changes in stoichiometry during these measurements are ruled out since their stoichiometry measurements correlate with their resistivity measurements.

FONTENAY-AUX-ROSES - JULY 15, 1967

In the morning Hubert Mikailoff discussed fuel irradiation results from the Rapsodie reactor and presented their plans for fuel development for the Phenix reactor. The information was presented in tabular form on lantern slides so that it was not possible to copy a great deal of detail. Most of the planned irradiations were to be conducted in thermal test reactors. These tests are capsule irradiations to evaluate a

FIGURE 24. Thermal Conductivity of $(U, Pu)O_{2+x}$

FIGURE 25. Thermal Diffusivity of $(U,Pu)O_{2+x}$

large number of the variables affecting fuel performance. I would estimate that greater than 100 capsule experiments are planned. In addition, prototypic Phenix Fuel arrangements are to be tested in Rapsodie.

Mikhailoff showed a drawing of the Rapsodie fuel pin: 340 mm in length with a spring at the top, sintered pellets of 26% PuO_2 , 96% theoretical density, 1.96 to 1.99 O/M, $230 \pm 80 \mu$ fuel-clad diametral gap, 6.7 mm OD, 5.8 mm ID clad with 0.8 mm high spiral rib on a 12 cm pitch. Cladding is 316L, but replacement fuel will use 316 cladding for better creep strength. The heat rating was 1310 W/cm^3 and $\int_{T_S}^{T_C} k dt = 26 \text{ W/cm}$.

The reactor inlet temperature is 410°C and the outlet temperature is 500°C . Hot spot temperature of the clad is 650°C . In the Rapsodie core there are 5 rows of fuel, 4 rows of blankets and 5 rows of reflector. The blanket is UO_2 , the reflector is SS. Each subassembly is orificed so that the same outlet temperature is obtained from each fuel bundle.

Following is some information from their thermal flux irradiations:

1. 304 and 316 SS cladding has been used in their thermal flux irradiations.
2. It was stated that good dimensional stability of the fuel has been observed.
3. Their Pu contains 3 to 5% ^{240}Pu .
4. At the present time no fuel irradiations up to or beyond center melting have been made.
5. They believe that thermal growth (ΔL) of a fuel column saturates.
6. Radial fuel clad gaps from 30 to 40μ after irradiation have been observed. Fuel cracking occurs consistently at center temperatures less than 1800°C . At temperatures greater than 1800°C unstable cracks and center void formations are observed.

(Mikailoff appeared to be reading from a report that was in a very preliminary form during this presentation.) We were given some reports from the CEA on fuel irradiations (Table IV).

Discussion on Fuel-Clad Gap Conductivity

It was stated that at the beginning of life, the gap conductivity depends upon the gas film. The values quoted were: 0.65 W/°C-cm in helium, and 0.30 to 0.40 W/°C-cm in argon.

It was stated that the fuel-clad gap closed after 1000 to 2000 MWd/tonne burnup.

Fission Gas Release

It was stated that fission gas release is temperature dependent at low burnups. 100% gas release at fuel temperatures greater than 1700 °C and 30% release in the equiaxed grain growth region has been observed. The release of fission gas increases with burnup. Defected fuel has been observed to release all the gas. In addition, in defected fuel they have seen intergranular attack on the fuel and degradation of the heat transfer within the fuel. At 7,000 MWd/tonne burnup 16% gas release occurred. At 22,000 MWd/tonne burnup 26% gas release occurred. In each case the temperature distribution was the same.

Rapsodie Subassembly Postexamination

This was actual driver fuel burned to 150 MWd/tonne. The ΔL was 0.2 to 0.8 mm (the fuel column was spring loaded). The initial fuel-clad gap was 190 μ . A large eccentricity of the pellet in the cladding was observed. When questioned whether or not there was concern about this in terms of hot spot formation and circumferential temperature distribution in the cladding, the reply was "no." The large eccentricities were not studied or considered in the design of the fuel.

TABLE IV. *List of Materials Received by J. E. Hanson*

From UK Workers:

1. "Some Modern Developments in Nuclear Power" by F. J. Barclay & T. J. Ledwidge. Paper presented at I.E.E. meeting, North Midlands Centre, February 21, 1967.
2. Tabulated data, "Composition of Irradiated Mixed Oxide Fuel at 9.66% Burnup." Received from Dr. Eric Wait at Harwell.
3. Brochure, "Radiometric Scanner with Polychromatic Display." Received at Harwell.

From French Workers:

1. Brochure, "The Fontenay-Aux-Roses Nuclear Research Center."
2. Paper, "The New Fontenay-Aux-Roses $\alpha\beta\delta$ Radiometallurgy Laboratory Design and Construction," by A. Valentin and A. Portal.
3. Paper, "Le Laboratoire de Radiometallurgie de Fontenay-Aux-Roses," by A. Valentin and J. L. Faugere.
4. Mikailoff, Hubert, et al, "Resultats D'Irradiation D'Elements Combustibles En Oxyde Mixte UO_2 - PuO_2 ," Rapport CEA-3066, September 1966.
5. Mikailoff, Hubert, et al, "Compartement A L'Irradiation De Carbures, Nitrures Et Carbonitrures Mixtes D'Uranium Et De Plutonium," Rapport CEA-R-3223, March 1967.
6. Graph, Variation of Lattice Parameter with O/M Ratio in $(U_{0.8} Pu_{0.2})O_{2+x}$ Fuel.
7. Photomicrographs of mixed oxide fuel.
8. Unpublished manuscript, "Part III - Vapor Pressure of Plutonium and Uranium Oxides."
9. Unpublished data, "Thermal Diffusivity of Mixed Oxide Fuel."
10. Unpublished data, "Thermal Conductivity of Mixed Oxide Fuel."

11. Processing diagram of Rapsodie Fuel.
12. Two pictorial drawings of Cabri Reactor.
13. Brochure, "The Cadarache Nuclear Research Center."

No grain growth was observed in the Rapsodie fuel. The center temperatures were stated to be less than 1700 °C (the subassembly was from the center of the reactor core, position 01-01). The estimated gap conductivity was 0.62 W/°C-cm. The pellets were badly cracked. The measured fission gas release was 1%. There were 37 pins in this fuel assembly. All were gamma scanned and radiographed. Eleven pins were pierced for fission gas release and four pins were examined ceremographically. A 300 kV X-ray machine is used for X-raying the fuel pins. The quality of their X-rays appeared to be much better than those the British showed us. The next Rapsodie subassembly for examination will be removed at 3 to 5000 MWd/tonne burnup at about the end of July. 30,000 MWd/tonne burnup is a maximum target burnup in Rapsodie. One gets the impression that the French feel the smeared density in the Rapsodie fuel is too high.

Thermal Flux Tests

The tests being done in thermal fluxes include the following parameters: Vipac and annular pellets, the effect of gap variations, smeared density, plutonium content, lineal power, sodium bonding (which is really sodium compatibility), migration of fission products, and fission product plugging, and burnup (80,000 MWd/tonne maximum).

Phenix Fuel Design

The Phenix fuel design was discussed briefly. The Phenix reactor power will be 250 MW. The fuel pellet density is 85% with 79% smear density, 5 1/2 mm pellet diameter, 850 mm column length, 0.2 mm diametral gap, 5.7 mm ID cladding, 6.6 mm OD, 0.350 mm blanket length, 420 mm clad length at the bottom of the pin (plenum at the bottom), 92 mm

plenum at the top of the pin (contains a spring). The total length of the fuel pin is 1,722 mm. The linear heat generation of the fuel is 430 W/cm. The fuel swelling is taken as 2% $\Delta V/10,000$ MWd/tonne burnup. Goal burnup in the Phenix fuel is 100,000 MWd/tonne.

Some irradiations are also being done in the DFR with the target burnup in these tests being 80,000 MWd/tonne. A 70% smeared density fuel specimen is used with 316 SS cladding. Heat rating is 420-450 W/cm. Some Hastelloy-X clad fuel pins are being irradiated in Rapsodie.

It is believed that a solid pellet of low density is better for accommodation of swelling than a high density annular pellet.

It is planned to do Phenix fuel development work in Rapsodie in approximately eight subassemblies of prototypic Phenix fuel arrangement and geometry. These include 316 SS and Hastelloy-X cladding. This is defined at the second phase of Rapsodie testing.

Discussions by M. Weisz (Saclay) on Cladding

This discussion was very brief due to lack of time and a certain reticence on M. Weisz's part to present any information. The fragmentary information obtained seems to indicate only a very small amount of cladding development work is being pursued by the French.

It was stated that a tensile in-pile creep rig had been designed, but no details were given. Also, pressurized tubing tests were being prepared for irradiation in Rapsodie.

Out-of-pile burst tests on Rapsodie clad showed rupture occurred 4 to 5 mm from the fin and followed the spiral of the fin.

Creep tests in sodium are being done (by EDF).

Weisz did not consider eccentricity of pellets in cladding a problem.

Inconel-X and Hastelloy-X are being evaluated as high temperature clad materials. Cold-worked 316 cladding is being evaluated. An 800 to 900 °C heat treatment is said to decrease creep strength, but increase ductility.

Sodium corrosion work is being done at Saclay and Grenoble, and the EDF is also doing work in this field. A "CCTL" is being constructed at Cadarache.

Phenix construction will start in 1969. The plant will be located at Marcoule.

Tour of New RML Facility

The visit to the new RML facility at Fontenay disclosed an up-to-date, well-equipped facility which utilizes the alpha box concept. Magnetic manipulators are used in conjunction with the alpha boxes.

CADARACHE - JULY 17, 1967

Discussions with M. Francois Stosskopf

Mr. Guillet is the Chief of Service in the Plutonium Technology Division. There are two sections in the Division. One is responsible for fabrication of fuel elements for core loadings and for irradiation tests. This is managed by M. Stosskopf. The other section is responsible for the design of irradiation tests and fuel elements and is managed by M. Ratier. There is a reprocessing plant at Cadarache for the reprocessing of cold scrap and which is also responsible for chemical analyses associated with fuel production.

There is a hot cell facility associated with Rapsodie which is used principally for the disassembly of subassemblies (the pins are then shipped to Fontenay for postirradiation examinations).

Since the plutonium laboratory was opened in 1962, they have made approximately one hundred subassemblies. These were

either standard Rapsodie subassemblies or experimental fuel subassemblies. They also make fuel for thermal flux irradiations for Mustellier at Fontenay. The capacity of the fuel fabrication line for Rapsodie was 70 pins per day. They anticipate that they will make the 33,000 pins for the Phenix reactor in 12 to 18 months.

They have lead-shielded glove boxes with lead glass windows and they feel that this is adequate for fuel containing up to 30% ^{240}Pu . They are going to use some simple, automated equipment to reduce exposure to personnel.

Stosskopf made the comment that they were not satisfied with the grid bar and spacer arrangement in the Rapsodie fuel assembly. The pins are not welded to the grid bars. They accidentally dropped a subassembly; the rivets where the grid bar assembly is attached to the hex wrapper were actually pulled out of the wrapper. They are now thinking about modifying the Rapsodie subassembly, and incorporating in the Phenix subassembly design pin bundles which are supported from the top of the fuel subassembly so that the connection can be made to a heavier section.

Stosskopf took us on a tour of his laboratory where they produced fuel pins. In many instances it appeared that it was a single pin operation. They have automatic pellet presses and a large number of sintering furnaces which are operated on a 24 hr/day basis.

Stosskopf displayed some cut away models of the Rapsodie fuel bundle and subassembly. There are 12 in. long blanket rods at the top and the bottom (about 5/8 in. diameter) of the fuel bundle. These are spiral wire wrapped. The orifice is at the top of the subassembly. There is a thermocouple which fits in a lid that is located above the top of the subassembly. The hex wrapper for the Rapsodie subassembly is 316 SS. It is welded. The total subassembly length is approximately 5 1/2 ft.

Regarding fuel fabrication, Stosskopf explained that the fuel process for Rapsodie fuel involves plutonium oxalate which is calcined in air at 700 to 900 °C to obtain PuO_2 . The UO_2 , (60% ^{235}U enriched) is received as ADU. It is also calcined in air to U_3O_8 and then reduced to UO_2 in a hydrogen-argon atmosphere. The fuel manufacturing process flow sheet for Rapsodie fuel is appended.

A test subassembly for irradiation in Rapsodie containing 80 to 85% theoretical density pellets was being fabricated.

The plutonium fuel fabrication facility is on a cell arrangement. It is very similar to the PNL 308 building although the cells or the rooms seem to be a little smaller. They are very well equipped and the equipment appeared to be quite modern and up-to-date. It was interesting to note that color coding on the glove boxes is used. The boxes containing hydrogen atmospheres are painted red.

In the fuel fabrication cells for the Phenix fuel, the glove box shielding seemed to be heavier than that for the Rapsodie fuel.

A centerless grinder that was used in the production of the Rapsodie fuel was pointed out. Stosskopf made the comment that 2% of the Rapsodie first core pellets were ground. They do not use the grinder now except for irradiation test fuel, where they want to precisely control the fuel-cladding gap.

There was also a glove box setup used for determining the gas content of the fuel pellets. I tried to get some information on what their sampling techniques are. Apparently there are about 500 pellets per sintering batch. I don't think Stosskopf quite understood me, when I asked how many pellets they take for analysis of Pu and U and impurities and so on. He mentioned a number of about five pellets which would be a 1% sampling.

He stated that 10% of the pellets were inspected for length, diameter and weight, and X-ray lattice parameter for O/M ratio. When we started to discuss homogeneity, and how they control and measure this during production of the fuel, I think we either had a language barrier problem, or that he didn't have a good answer for it. Apparently, they feel that if they can control homogeneity in the step where they grind or crush the pellets after they have been pre-pressed, and get less than a 25 μ Pu particle size they are happy. There was some allusion that this number came from the physicists.

They have a semiautomatic pellet loader which looked to be fairly sophisticated. It was a funnel arrangement, with a hopper for pellets -- the pellets are moved by vibration. The fuel column length and weight is taken automatically. The cladding protrudes through the side of the glovebox with a sealing arrangement so that they get no contamination on the end of the clad. Here again the lead shielding and lead glass was very heavy, and it appeared to be a single pin operation.

Stosskopf's people were in the process of putting together a Phenix prototype fuel bundle for irradiation in Rapsodie.

Discussions with M. Ratier

Mr. Ratier's responsibilities include conceptual design for irradiation experiments and the conceptual design for the Phenix fuel assembly. Ratier receives functional specifications for the Phenix fuel assembly requirements from Vendreyes, who is in charge of fast reactor projects.

Ratier showed us drawings of the conceptual design for the Phenix fuel subassembly. The important points are that the gas plenum is at the bottom of the fuel pin and the fuel pin will be fabricated in three sections (apparently to fit into their glove boxes). Ratier showed us the machine being developed to put the spiral wire wrap on the Phenix fuel. This is a single

pin operation. The work was being done in a glove box, but we were told they were only doing this while they were developing the process. The top end cap of the fuel pin is made in two portions to facilitate the wire wrapping. One piece is free to rotate while the wire is being wrapped on. The two pieces are tack welded after wire wrapping.

Ratier then briefly discussed the design of the Phenix fuel subassembly. He stated that 1% ΔD of the fuel pins is equivalent to a 2 to 3% flow reduction. This is included in the hot spot calculations.

In design criteria for the cladding stresses Ratier stated that the summation of the thermal stresses plus the pressure stresses has to be less than the relaxation stress or the elastic limit stress of the 316 SS cladding.

Some creep and fatigue tests to evaluate the effects of clad defects on cladding strength are planned. Ratier stated that he did not think that tensile tests and burst tests were really meaningful in evaluating the effect of defects on cladding strength.

They also use secondary creep as a criteria for the deformation they expect in the fuel simply due to fission gas pressure. Under close questioning, Ratier admitted that they are using a very low density fuel in the Phenix fuel in order to accommodate swelling in the fuel without fuel-cladding interaction at the end life.

CADARACHE - JULY 18, 1967

Tour of Nondestructive Testing Laboratory with M. Ratier

Nondestructive testing techniques employed for cladding are eddy current, dye penetrant, and burst tests. Their standard defect for eddy current testing is 2 mm long, 0.4 mm deep 0.2 mm wide.

We were shown an eddy current testing machine being developed which employs frequencies of 300 to 600 kHz in two coupled eddy current coils. The coils are coupled so that a single disturbance or an indication of a defect from one coil, is amplified by the coupling with the other coil. The amplification appears in both the amplitude and the frequency changes.

An interesting comment made by Ratier was that the straightening process for cladding resulted in diameter changes which gives rise to defect signals. He then stated that they no longer permit vendors to straighten tubes as the tubes straighten themselves when they are placed in the subassembly.

A capacitance-type transducer setup is used to measure OD and ID of the cladding.

In general, Ratier stated, they get 60% acceptance on a good batch of tubing and 20 to 30% acceptance on a bad batch of tubing. He attributed defects in tubing to poor handling at the mill. This is basically the principal cause for inclusions in the cladding. Surface defects account for the majority of the rejects.

Another high capacity testing machine for production testing of cladding was shown. This machine uses 10 kHz and 100 kHz frequencies in the eddy current coils. Ratier said they can inspect one hundred-fifty tubes per hour with four people operating this machine.

A dye penetrant test was used to inspect the ID of the Rapsodie cladding. They rejected tubes with any sign of illumination from the dye penetrant. This testing technique reveals surface defects on the ID. It employs an ultraviolet light source and a light tube arrangement with a TV camera looking through the opposite end of the tube. This technique is not planned to be used for testing the Phenix cladding. Eddy current testing will be the principal NDT technique used on Phenix cladding.

Another interesting machine was being used to pressure-test cladding tubing. This machine has the capacity to simultaneously test about 12 or 15 tubes. 350 bar pressure is used for testing the Rapsodie cladding. Tubes are rejected if they show more than 5 μ deformation. 1 or 2 μ deformation is usually obtained. Ratier was asked if he considered this cold-working the tubes and he said, "Well actually it was, but it was but only by a very small amount."

Eddy current testing techniques are being developed for industrial use. Ratier said that they have always been able to correlate signals from the eddy current tests with defects. He mentioned that ultrasonic testing techniques were being developed at Saclay and they will start to use these next year.

Ratier stated they have a good appreciation for the limitations of both ultrasonic and eddy current testing.

Ratier was asked who was responsible for preparing specifications for the cladding materials. He stated that he was and then showed us (a very brief glimpse) a specification he had prepared for cladding materials. He stated that the 316 stainless steel was vacuum melted and the carbon content was 0.055 to 0.065%. The inclusion specification was ASTM E45. The ASTM grain size was 6 to 10. The tubing is not straightened.

Discussions with M. Guy Denielou

The fuel subassembly for the Phenix reactor will incorporate a thermocouple to monitor the coolant exit temperature. The location of the thermocouple will be a bit removed from the subassembly as tests in Rapsodie, with the Rapsodie fuel subassembly thermocouples, has shown that the influence of thermocouple position on coolant exit temperature measurement was less than anticipated.

Tests have been conducted in Rapsodie on subassemblies with reduced flow. (Each subassembly is orificed according to its position in the reactor core). The tests involved flow reductions of 90 to 60% of full flow. The results showed that adjacent subassemblies appear to be unaffected by a subassembly with reduced flow. The increased coolant exit temperature in the reduced flow subassembly was not seen until the reactor reached 20% of full power due to:

1. Thermocouple insensitivity
2. Low ΔT across the subassembly.

Denielou was very adamant in stating that for precise evaluation of reactor performance the coolant inlet temperature must be very accurately measured. This requires good position (very close to the bottom of the core), and high precision instrumentation such as resistance temperature detectors which are easily replaceable. The coolant exit temperature is scanned by data logging equipment which gives round figures for temperature. There were uncertainties in what the precise value of the ΔT and exit temperature should have been.

In Rapsodie the effect of the control rods was underestimated. This was determined by a power tilt which was revealed by the subassembly thermocouples. Variations in flow rate did not change the temperature skewing. They now intend to use only the shim rods in Rapsodie and are planning on doing the same in Phenix since they feel this will result in better flux and power distribution in the core.

Denielou stated that since the coolant exit temperature in each subassembly is quite insensitive to what occurs in the adjacent subassemblies, it is mandatory that each subassembly coolant exit thermocouple be operative. Thus, these thermocouples must also be easily replaceable.

Another very important aspect of thermocouple instrumentation for precise measurement of temperatures is proper attention to the cold junctions of the thermocouples. Denielou said that this had gotten them into difficulty due to inattention to this aspect of the instrumentation system.

Thermocouple noise analysis work is being done at both Cadarache and Grenoble. Denielou feels it is more important to develop the noise analysis of the ΔT rather than the exit temperature. He feels this would be a more sensitive measurement of reactor performance.

Reactivity and coolant exit temperature measurements from each subassembly will be the principal safety circuit trips in the Phenix reactor.

During tests of the failed element detection system in Rapsodie, it was noted that the exhaust stack gas activity could detect fission product gases faster than the cover gas or delayed neutron monitoring systems.

Denielou stated he did not think much of boiling detection systems since the range between local boiling and vapor formation was too small.

Steam generator development work is being done in Paris under M. Robin.

Discussions with Messrs. Millot, Ferrarri, and Lion on Cabri Loss-of-Flow Tests

In-reactor testing for evaluation of gross behavior of fuel elements under loss-of-flow conditions is felt to be the most direct way of studying this problem. Out-of-pile testing is limited by capacity of electrically heated rods and electrical interference in the instrumentation.

They have done some preliminary work on chemical heating (thermite mixtures) but are not satisfied with the results.

A pictorial description of the Cabri reactors is given in Figures 26 and 27. The reactor in which the sodium loop is to be installed is capable of operation at 100 MW power for 10 min.

A visit to the reactor site was not possible since they had only recently completed a down-flow DNB test and had experienced a fuel element (MTR-type) meltdown and were in the process of cleaning up. The meltdown, it was stated, was an intentional effort to establish the upper limit of heat transfer capability of the core.

A small sodium loop is being designed and fabricated for the initial loss of flow tests in Cabri. This loop will accommodate up to three fuel pins. Tests will be conducted with UO_2 of 3 1/2 to 20% ^{235}U in 0.57 to 0.58 cm diameter pins. Testing is planned to start the end of this year. The program will involve approximately 100 single pin experiments.

A large loop which would be capable of testing a seven subassembly cluster is planned as a follow-on to the small sodium loop. Coupling of the test section with the thermal driver core is a problem. The fast test section would generate about 10% of the power of the core. Use of flux filters, such as cadmium or stainless steel with boron, is being evaluated in addition to an annular arrangement of enriched UO_2 pins.

Some fast/thermal coupled experiments have been performed in ERMINE. Measured spectrums agree with calculations.

The description of the small sodium loop follows:

1. It will be a reentrant tube type arrangement. Pumps, heat exchangers, storage tanks, etc. are located in a separate building which adjoins the reactor building.
2. A horizontal section through the test section showed: (starting from the center) the test pin, the flow channel (sodium), niobium tubing, sodium annulus (by pass flow), and stainless steel pipe.

FIGURE 26. Cabri Reactor

FIGURE 27. Cabri Reactor

3. The in-core portion of the loop is approximately 8 m long.
4. The by-pass flow is approximately ten times that of the flow through the channel to keep the ΔP across the test section constant.
5. Instrumentation will consist of thermocouples to measure coolant inlet and exit temperatures, perhaps thermocouples on the fuel specimen cladding, electromagnetic flow meters, fission product detection (D/N system) and pressure transducers.

Some notes on the CEA organization for safety:

CEA Organization for Fast Reactor Work

APPENDIX A

AGENDAS OF EACH MEETING HELD DURING THE U.S. FFTF FUELS
AND MATERIALS TEAM TRIP TO THE UNITED KINGDOM AND FRANCE

Agendas

TRG/1/43/67 USA

Discussions will be held in Room E.508

AGENDA

(U.S. topical
agenda item
2 (c)

REVIEW PERFORMANCE OF STAINLESS STEEL CLAD MIXED
OXIDE FUEL SUBASSEMBLIES: DEVELOPMENT OF MODELS
OF FUEL PERFORMANCE AT HIGH BURNUP - led by
Dr. Bishop.

- (i) Fuel swelling
- (ii) Fission gas release
- (iii) Clad deformation
- (iv) Clad
- (v) Clad-fuel interaction
- (vi) Fuel movement
- (vii) Physical and chemical properties of fuel.

Items 2A and 2B of the U.S. topical agenda will be dealt with at Dounreay: see attachment.

12.45 LUNCH - Visitors dining room No. 1.

(U.S. topical
agenda item 3) REVIEW FUEL PIN AND FUEL SUBASSEMBLY DESIGN
OF MIXED OXIDE CORES - led by Mr. Pounder
and Dr. Bishop.

- A. Design bases and criteria
- B. Methods of design
- C. Development programmes and prototype tests for developing and proving core design

(Part of U.S. topical agenda Item 4) PFR CORE REFERENCE DESIGN -
led by Mr. R. H. Campbell, Dr. R. D. Smith,
Mr. R. B. F. Evans

- A. Nuclear performance and measurement
- B. Thermal performance and measurement
- C. System control and stability
- D. In-core surveillance and interpretation methods.
- E. In-core instrumentation.

The other part of U.S. item 4 - DFR operating experience will be dealt with at Dounreay.

19.45 Dinner at Stanneylands Hotel, Wilmslow -
for 20.00 Mr. Frame, Mr. Campbell, Dr. Bishop,
Dr. Smith, Dr. P. T. Nettley (Culcheth
Labs.), Mr. I. P. Bell (Culcheth Labs.)

DISTRIBUTION

Mr. J. G. Yevick	Mr. Alistair G. Frame
Mr. G. W. Cunningham	Mr. R. H. Campbell
Mr. N. E. Todreas	Dr. John F. W. Bishop
Mr. R. E. Patterson	Dr. P. T. Nettley
Mr. T. T. Claudson	Dr. R. D. Smith
Mr. J. E. Hanson	Mr. I. P. Bell
Mr. N. P. Klug	Mr. P. B. F. Evans
	Mr. John O. Pounder

June 27, 1967

Overseas Relations,
Reactor Group,
Risley.

AnnexOther U.S.A.E.C. topical agenda items
will be dealt with as followsCulcheth-Tues., July, 4

1. Review mechanical performance of stainless steel fuel clad and structural parts in fast reactor and sodium environment.
 - A. Mechanical properties as a function of irradiation exposure.
 - C. Examination techniques and interpretation of results (in so far as relevant).
 - D. Corrosion and mass transfer.

Lunch at Risley, Visitors Dining Room No. 1.

Dounreay-Wed., Thur., Fri., July 5, 6, 7

1. as above
 - B. Fuel clad performance and C, as above.
 2. Review performance of stainless steel clad mixed fuel subassemblies.
 - A. Performance versus irradiation exposure fuel form, composition of materials and element configuration.
 - B. Postirradiation fuel examination techniques and interpretation of results.
- 4A - 4E as they apply to DFR operating experience.

UNITED KINGDOM ATOMIC ENERGY AUTHORITY
DOUNREAY EXPERIMENTAL REACTOR ESTABLISHMENT

Programme for the visit of
U.S.A.E.C. Fast Reactor Team
on July 5-7, 1967

U.S.A.E.C. Team

Mr. J. G. Yevick	- U.S.A.E.C.
Mr. N. E. Todreas	- U.S.A.E.C.
Mr. G. W. Cunningham	- U.S.A.E.C.
Mr. R. E. Paterson	- Battelle
Mr. T. T. Claudson	- Battelle
Mr. J. E. Hanson	- Battelle
Mr. N. Klug	- U.S. Embassy - London

U.K.A.E.A.

Dr. H. C. Dunn - Overseas Relations, Risley

Wednesday, July 5, 1967

- 11.15 Arrive Wick Airport.
Transport to Dounreay.
- 12.15 Arrive Dounreay - Conference Room.
Met by Mr. R. R. Matthews, Director,
Mr. J. L. Phillips, Assistant Director, and
Mr. L. R. Williams, Head of Metallurgy Division.
- 12.45 Lunch - Senior Staff Dining Room.
- 13.45 Transport - D1300 to D1120.
Visit to Fast Reactor Control Room and Sphere.
- 14.45 Return to Mr. Henry's Office.
Discussions on DFR Operating Experience.
Mr. J. L. Phillips, Mr. K. J. Henry, Mr. J. G. Walford,
Mr. E. Edmonds, Dr. A. G. Edwards, Mr. T. Ledwidge and
Dr. I. R. Birss.
- 16.45 Depart Site for Pentland Hotel, Thurso.

Thursday, July 6, 1967

09.00 Transport - Pentland Hotel to Dounreay.

09.30 Arrive Dounreay - Conference Room.

Discussions on Experimental work in aid of PFR.
Mr. R. R. Matthews, Mr. L. R. Williams,
Dr. K. M. Swanson, Dr. D. M. Donaldson,
Mr. E. P. Hicks, Mr. N. Parkinson,
Mr. D. Mosedale, Mr. R. Diggle, Dr. P. Brock,
and Mr. J. Bramman.

11.25 Transport - D1300 to D1200.

11.30 Visit D1201 - Fuel Element Plant - Dr. P. Brock.

11.45 Visit D1202 - Irradiation Rigs - Dr. P. Brock.

12.00 Visit D2001 - Equipment and Techniques -
Dr. K. M. Swanson.

12.35 Return to Conference Room, D1300.

12.45 Lunch - Senior Staff Dining Room.

13.45 Visits or discussions as required.

16.45 Depart Site for Pentland Hotel.

Friday, July 7, 1967

Programme to be arranged.

16.00 Depart Wick Airport - Flight B.E.8015

July 3, 1967
D.E.R.E.

(G. M. MALCOLM)
D8538/22
Ext.7242

DistributionFor Information

Mr. R. R. Matthews
Mr. J. L. Phillips
Mr. K. W. Matthews
Dr. S. M. B. Hill
Mr. D. M. Carmichael (2)
Dr. J. S. Broadley
Mr. A. L. Cuthbert
Mr. P. R. Spencer
Mr. B. Durrans
Mr. K. R. Montgomery
Mr. R. F. Thomas
Mr. R. Wright
Mr. J. Dea
Mr. A. Walker
A.E.A.C.
Transport
File

For Action

Mr. L. R. Williams
Mr. K. J. Henry
Mr. J. G. Walford
Mr. E. Edmonds
Dr. A. G. Edwards
Mr. T. Ledwidge
Dr. I. R. Birss
Dr. K. M. Swanson
Dr. D. M. Donaldson
Mr. E. P. Hicks
Mr. N. Parkinson
Mr. D. Mosedale
Mr. R. Diggle
Dr. P. Brock
Mr. J. Bramman
Mr. R. Wilson
Mr. J. Davies
Mr. S. Fulcher
Mr. R. Henderson
Mr. W. Hamilton

VISIT PROGRAMME

U.S.A.E.C. Fast Reactor TeamJuly 10, 11, 1967Monday, July 10

10.20 Arrive Main Gate, met by Mr. W. R. Wiseman, issued with passes and taken to B.220.

10.30 Arrive B.220 Conference Room for Introductory Discussion with

Metallurgy: Dr. R. S. Barnes; Dr. B. R. T. Frost;
Mr. R. G. Bellamy; Dr. D. R. Harries.

Ceramics: Dr. L. E. Russell; Dr. J. Williams.

Chemistry: Dr. L. E. J. Roberts; Dr. E. Wait;
Dr. J. K. Dawson.

Research

Reactors Dr. V. S. Crocker; Mr. I. Everson

Coffee served

(Possible visits to B.459 and B.459.4)

12.30 Lunch

2.00 (1) Discussions on Cladding.

Dr. R. S. Barnes; Dr. D. R. Harries;
Dr. P. C. L. Pfeil; Mr. G. H. Broomfield;
Mr. B. W. Mott and D. G. Long (for corrosion aspects)

Tea to be served at 3.15 p.m.

Tuesday, July 11

9.30 Fuel - Applied Studies and Postirradiation Techniques. Mr. R. Bellamy;
Mr. J. D. B. Lambert; Dr. B. R. T. Frost;
Mr. J. B. Sayers; Dr. A. J. Fudge;
Mr. H. Venables; Dr. E. Wait

(Visits to B.393.6: B.220; B.459; B.459.4 and B.388)

Coffee to be served at 10.15 a.m.

12.30 Lunch

2.00 (1) Basic studies on Fuel.

Dr. R. S. Barnes; Dr. B. R. T. Frost;
Mr. R. Bellamy; Mr. J. B. Sayers;
Dr. A. D. Whapham; Mr. B. L. Eyre; Dr. E. Wait;
Dr. J. M. Davies; Mr. H. J. Hedger;
Mr. J. R. Findlay; Dr. L. E. J. Roberts;
Dr. J. K. Dawson; Dr. L. E. Russell.

(2) Fast Flux Test Reactors (F.F.T.F. & M.F.R.)
B.220.

Dr. V. S. Crocker; Mr. I. Everson;
Mr. G. Constantine; Dr. L. E. J. Roberts;
Dr. B. R. T. Frost.

Tea in B.220 at 3.15 p.m.

4.00 Depart for London Airport

W. R. Wiseman

Public Relations Office,
Building 77, Extension 2704.

July 7, 1967

Distribution:

Dr. R. S. Barnes
Dr. B. R. T. Frost
Mr. R. Bellamy
Dr. D. R. Harries
Dr. L. E. Russell
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The following to take
notes of discussions:

Mr. K. Rose	Monday morning
Mr. P. Higgins	Monday afternoon
Mr. J. Lambert	Tuesday morning
Mr. G. Constantine	Tuesday afternoon

Mr. B. L. Eyre
Dr. J. H. Davies
Mr. H. J. Hedger
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AGENDA

VISIT USAEC FFTF FUELS AND MATERIALS TEAM
TO CEA - FONTENAY AUX ROSES, FRANCEJuly 12, 1967

1. Discussion of FFTF
Over-all Core Design	USAEC	N. Todreas
2. Discussion of FFTF Fuels
Program	USAEC	G. Cunningham
3. Discussion of Fuels
& Materials Program	PNL	J. Hanson
4. Rapsodie and Phenix In-Core
Instrumentation	CEA	M. Rozenholz*
5. Rapsodie Mixed Oxide
Fabrication	CEA	G. Dean
6. Fuels - Physical Properties	CEA	A. Lallement
7. Tour of Physical Measurements
Laboratory and Oxide Pilot
Line		

* From Saclay

July 13, 1967

1. Rapsodie and Phenix Fuels
Irradiation Program	CEA	H. Mikailoff
2. Phenix Clad Program	CEA	Weisz**
3. Tour of Hot Cells		

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