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ANL-7618

Reactor Technology

ARGONNE NATIONAL LABORATORY
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REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT

September 1969

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FOREWORD

The Reactor Development Program Progress Report, issued monthly, is intended to be a means of reporting those items of significant technical progress which have occurred in both the specific reactor projects and the general engineering research and development programs. The report is organized in accordance with budget activities in a way which, it is hoped, gives the clearest, most logical overall view of progress. Since the intent is to report only items of significant progress, not all activities are reported each month. In order to issue this report as soon as possible after the end of the month editorial work must necessarily be limited. Also, since this is an informal progress report, the results and data presented should be understood to be preliminary and subject to change unless otherwise stated.

The issuance of these reports is not intended to constitute publication in any sense of the word. Final results either will be submitted for publication in regular professional journals or will be published in the form of ANL topical reports.

The last six reports issued
in this series are:

February 1969	ANL-7553
March 1969	ANL-7561
April-May 1969	ANL-7577
June 1969	ANL-7581
July 1969	ANL-7595
August 1969	ANL-7606

REACTOR DEVELOPMENT PROGRAM

Highlights of Project Activities for September 1969

EBR-II

On September 18, reactor power was raised to a new high of 62.5 MWt. The increase from the 50-MWt level at which the reactor has been operating was preceded by a two-day period of operation at 56 MWt. The system and its components performed satisfactorily at both higher power levels.

Power level was returned to 50 MWt after 517 MWd of operation at 62.5 MWt and will remain at 50 MWt for about six months. During that period of time, the experimenters using EBR-II will have the opportunity to modify their experiments, wherever necessary, for irradiation under full-time operation at 62.5 MWt. Continuous, full-time operation at 62.5 MWt is expected to start early in 1970.

The reactor was operated for 819 MWd from August 21 through September 20, raising its cumulative operation to 29,013 MWd.

ZPR-3

Experiments with Assembly 57, the plutonium-alpha irradiation core, have been completed and the reactor unloaded.

ZPR-6 and -9

ZPR-6 has been preloaded with the inert components which will be part of Assembly 6A, the rebuilt Assembly 6 which was a 4000-liter ^{235}U core; fuel loading will begin shortly.

Assembly 25, also a ^{235}U system, has been constructed on ZPR-9 with a critical mass of 468 kg; the exact null composition was determined from reactivity worth measurements, and a central test zone was reloaded to this composition.

ZPPR

Verification tests of the containment system indicate that all measured leak rates are entirely satisfactory and that an adequate surveillance program is in effect at ZPPR.

The shield-study portion of the FTR-2 experimental program has been initiated after the B_4C control ring was replaced with a sodium-stainless steel shield.

Sodium Technology Program

Beginning with this issue, all of the work in the Laboratory's Sodium Technology Program will be reported in one section of this report (see Sect. I.C. Sodium Technology). Work on the EBR-II Project covering sodium components, systems, instrumentation development, and operations continues to be reported under EBR-II Research and Development.

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I. LIQUID METAL FAST BREEDER REACTORS--CIVILIAN

A. Physics Development--LMFBR

1. Theoretical Reactor Physics

a. General Reactor Physics

(i) Group-constant Averaging (D. A. Meneley)

Not previously reported.

(a) Analytic Representation of Fast Reactor Spectra. In an effort to represent the fast reactor flux analytically, the infinite medium flux is described by

$$\phi(E) = \frac{1}{[\Sigma(E)][aM(E)]} \int_E^\infty dE_0 \chi(E_0)s(E_0) \exp\left[-\int_E^{E_0} \frac{dE'}{M(E')}\right] + \frac{\chi(E)}{\Sigma(E)}, \quad (1)$$

where

$\Sigma(E) \equiv$ the total cross section at E;

$a(E) \equiv$ the absorption ratio at E, that is, $\Sigma_a(E)/\Sigma(E)$;

$s(E) \equiv$ the scattering ratio at E, that is, $1 - a(E)$;

$\chi(E) \equiv$ the fission spectrum;

$M(E) \equiv$ the moderating ratio:

$$M \approx \left(\frac{s\xi^{(1)}}{a} + \frac{\xi^{(2)}}{\xi^{(1)}} \right) E + \left(\frac{sQ^{(1)}}{a} + \frac{Q^{(2)}}{Q^{(1)}} \right), \quad (2)$$

where $\xi^{(1)}$, $\xi^{(2)}$, $Q^{(1)}$, and $Q^{(2)}$ are defined by an equation of the type

$$sP \equiv \sum_{\lambda=\text{all levels}} s_\lambda P_\lambda, \quad (3)$$

in which P_λ stands for $\xi_\lambda^{(1)}\xi_\lambda^{(2)}$ or $Q_\lambda^{(1)}$ or $Q_\lambda^{(2)}$. Here $\xi_\lambda^{(1)}$ and $\xi_\lambda^{(2)}$ are equal to the first and second lethargy moments in elastic scattering from the element containing the λ 'th level; $Q_\lambda^{(1)}$ and $Q_\lambda^{(2)}$ are the energy loss and half the square of the energy loss by scattering through the λ 'th level. The applicability of Eq. (1) in replacing ultrafine group calculations in the MC² code is being studied.

(ii) Reactor Dynamics (D. A. Meneley)

Last Reported: ANL-7606, pp. 4-6 (Aug 1969).

(a) QX1--Asymmetric Insertion Case. As a selected demonstration problem, a variation has been made of the reactor studied in ANL-7606. That reactor was represented in one-dimensional cylindrical geometry. A slab mockup was produced by maintaining the same core, blanket, and reflector dimensions as given in Progress Report for July 1969, ANL-7595, pp. 5-7, but considering these diametral dimensions as slab thicknesses. The insertion consisted of a linear increase in plutonium density in one of the two slabs representing the outer core zone. This case is obviously more susceptible to tilting from such an insertion than is the real cylindrical system, but the amount of this difference is difficult to estimate with a one-dimensional code.

The initial enrichments of the core zone were varied slightly from those in ANL-7595 in order to match peak power densities.

Fig. I.A.1. Amplitude Function after Asymmetric Insertion into Slab Mockup of an LMFBR Reactor

The values of $\text{Pu}/(\text{U}+\text{Pu})$ were 0.160 and 0.204 in the inner and outer core zones, respectively. Seventy mesh intervals and 26 energy groups were used in the problem. Initial criticality was achieved by adjustment of the transverse bucklings. In the point kinetics sense, the insertion rate was 14.517 dollars per second; the isothermal Doppler coefficient between 1300 and 2500°K was $-T dk/dT = 0.00667$.

The flux amplitude functions for this case, as calculated by both point and spatial kinetics, are shown in Fig. I.A.1. The spatial case is plotted only up to 0.115 sec, at which time the peak fuel temperature was above 2600°K. Peak fuel temperature at 0.15 sec in the point kinetics case was 2340°K. Over the range from 0 to 0.15 sec, the average true insertion rate was 21.147 dollars per second, as calculated from the static k_{eff} at 0.15 sec, with initial temperatures and total delayed-neutron fraction. The average value of the Doppler coefficient, $(-T dk/dT)$ over the range

cannot be defined easily because of the uncertain definition of effective temperature. The amplitude of the spatial case at 0.15 sec was 169.55. Figure I.A.2 shows the power shape functions for the initial state and at 0.115 sec.

Fig. I.A.2
Power Shape Functions during the
Asymmetric Insertion into the Region
between $x = 30.5$ cm and $x = 65.2$ cm

(b) QX1--Dependence of Accuracy on the Number of Shape Functions. The criteria for selection of the time step were relaxed in two of the cases described previously. The first was the uniform insertion case described as the " $0.01 \times P_0$ " case in Fig. I.B.1 of ANL-7606. The criteria were relaxed in several stages from the reference-case values, until 14 shape functions (including the initial) were calculated in 0.15 sec, an average spacing of over 10 ms. (Prompt-neutron generation time at $t = 0$ equals $0.355 \mu\text{s}$.) The errors in amplitude at 0.15 sec and excess energy integral over the range were +0.07 and +0.01%, respectively.

This case might be expected to converge with few recalculations, since the shape function changed very little during the transient. The test was repeated for the asymmetric slab described in Sect. (a) above. The final case required 16 shape functions over the excursion. The errors in the flux amplitude and energy integral at 0.15 sec were +0.2 and 0.05%, respectively.

It was noted that these errors fluctuated slightly as a function of the number of shapes when that number is small, due to variation in the recalculation times selected by the automatic time-step selector. The number of shape functions could be reduced further in this case, but in practice this could be done only by turning off the step selector and picking specific recalculation times directly. This illustrates two features of the model at the low-accuracy end of the scale. First, when the recalculation times of the few shape functions are preselected this model bears a very close relationship to the adiabatic model of Henry and Curlee.* Second, in such a case it is possible to save the shape functions from one run and use them for another similar case. "Use" in this context could mean either directly or as first approximations to be improved during the iterative solution procedure. In the new case they would not necessarily be assigned to the same reactor times, but rather to the same or similar reactor states.

In summary, the model includes great flexibility in terms of simplification and run-time savings, while it retains the self-checking

*Ott, K. O., and Meneley, D. A., Nucl. Sci. Eng. 36, 402 (1969).

features of an "exact" reference calculation. This continuous and consistent range of approximation within a single model is an important feature when neutronics is part of a large code system such as the one outlined below.

(c) Planning of the Two-dimensional Multigroup Kinetics System. This planning began with the objective of building a production version of a two space-dimensional multigroup diffusion-theory kinetics module, suitable for use in fast reactor dynamics analysis. This goal was presumed to require complete compatibility with other elements of the SAS package. SAS is intended for complete analysis from normal conditions through slow transients, coolant voiding, fuel displacement, and rapid disassembly. Such a range of problems requires a very flexible neutronics model. The flux factorization model developed in the QX1 program was the starting point for this development.

One new problem that must be faced in two dimensions is mesh distortion. This problem has been treated in hydrodynamics codes by adopting a Lagrangian mesh and tolerating the additional spatial truncation error coming from mesh distortion. Mesh readjustment must be carried out periodically during the problem if the system distortions are large. When this operation is carried out, the Lagrangian mesh option loses all of its advantages as far as neutronics are concerned. Mesh-to-mesh interpolation routines must be written, and compositions can no longer be defined for large regions, then modified locally by density factors. Compositions thus become mesh-interval dependent and produce storage problems. In this work it has been decided to maintain a stationary mesh grid for neutronics and allow the reactor to move over it. The ramifications of this are broad. For example, as a direct result no homogenized cross sections are stored, but only microscopic quantities plus mesh-dependent atom-density vectors and temperatures. The resulting flexibility makes the overall design of the system significantly simpler. This moving-mesh system may well be useful in reactor statics analysis, since irregular reactor shapes arising from thermal expansion, bowing, and the like can be approximated very much more easily than with the region-composition mesh concept.

Planning and module design are proceeding with emphasis on the requirements of linkage to the other physical models of the final system. Storage requirements and execution times are being studied in detail. The system is planned for incorporation into the ANL-ARC system.

2. Experimental Reactor Physics

a. Fast Critical Experiments--Experimental Support--Illinois

(i) Selected Differential and Integral Data (R. Gold)

Last Reported: ANL-7561, p. 4 (March 1969).

(a) Survey of ^{99}Mo Fission Yields. Plutonium-239 capture-to-fission ratios have been measured by the reactivity-reaction rate method.

In the analysis of these data, the most important fission product nuclide used in determining fission rates by radiochemistry is ^{99}Mo .^{*} The technique was established at ANL and developed into a tentative standard.^{**} The accuracy of the measurement is limited by the uncertainty in the value of the ^{99}Mo fission yield.

In order to gain a clear picture of the current status of knowledge regarding ^{99}Mo fission yields, the literature was searched for all reported measurements of this important quantity. Although several surveys of fission yields are available, a special effort was made to examine only the original data (which in a few instances was corrected at a later date) in this study and so avoid secondary references to the same measurement.

Table I.A.1 lists the reported ^{99}Mo fission yields in percent. The errors are those quoted by the authors and in most cases refer to the standard deviation in the measurement.

TABLE I.A.1. Measured ^{99}Mo Fission Yields (%)^a

Neutron Energy	^{233}U		^{235}U		^{238}U	^{239}Pu	
	Thermal	Fission	Thermal	Fission	Fission	Thermal	Fission
	$5.1 \pm 0.5(17)$	$4.75 \pm 0.35(11)$	$6.2 \pm 0.6(1)$	$6.10 \pm 0.16(3)$	$5.9 \pm 0.6(1)$	$6.13 \pm 0.10(1)$	$5.94 \pm 0.12(1)$
	$4.96 \pm 0.15(6)$		$5.3 \pm 0.3(2)^b$	$6.1 \pm 0.9(7)$	$6.19 \pm 0.15(3)$	$6.02 \pm 0.18(6)$	$5.5 \pm 0.4(9)^b$
	$5.35 \pm 0.14(10)$		$6.14 \pm 0.16(3)$	$5.9 \pm 0.4(8)$	$6.45 \pm 0.16(3)$	$5.61 \pm 0.33(12)$	$5.9 \pm 0.6(11)$
	$5.03 \pm 0.13(10)$		$5.98 \pm 0.18(4)$	$6.4 \pm 0.4(9)$	$6.4 \pm 0.7(16)$		
	$5.23 \pm 0.18(10)$		$5.9 \pm 0.2(5)$	$5.23 \pm 0.14(22)^b$	$6.2 \pm 0.9(7)$		$6.11 \pm 0.37(14)$
	$5.05 \pm 0.15(10)$		$6.25 \pm 0.19(6)$		$6.6 \pm 0.4(9)$		$5.81 \pm 0.15(15)$
			$6.06 \pm 0.13(10)$		$7.0 \pm 0.7(8)^b$		
			$6.20 \pm 0.20(10)$				
			$6.37 \pm 0.25(10)$				
Average Percent Fission Yield	5.12	4.75	6.14	6.12	6.29	5.92	5.94
Standard Deviation	0.059	0.35	0.054	0.103	0.101	0.158	0.063
Error at 95% Confidence Limit	0.12		0.11	0.20	0.20	0.31	0.12
Combined Yields				6.13			5.93
Standard Deviation				0.047			0.069
Error at 95% Confidence Limit				0.092			0.134

^aSee Table I.A.3 for references.

^bThese values are not included in the average since the probability of their occurrence as a legitimate experimental fluctuation is less than 0.001.

No attempt has been made to weigh the available data in order to obtain the "best" value of the yield. Instead, a simple average

* Armani, R. J., "Absolute Determination of Fission Rates in U-235 and U-238 by Radiochemical Techniques," Proc. IAEA Symposium on Standardization of Radionuclides, Vienna, Austria, October 1966, IAEA, Vienna (1967), p. 613.

** Tentative Method of Test for Fast Neutron Flux by Analysis of Mo-99 Activity from U-238 Fission, ASTM Designation E343-67T.

of the data has been made. In a few cases, as indicated in Table I.A.1, an experimental value has been omitted in determining this average yield. A suspected measurement was rejected if its deviation $|r_i - \bar{r}|$ was equal to or greater than four times the average deviation of the set, where the average deviation was computed by excluding the quantity in question. The probability of a deviation of this size resulting from a legitimate experimental fluctuation is about 0.001. Table I.A.1 also lists the standard deviation in the average as well as error estimates at the 95% confidence limits.

Molybdenum-99 falls on the low-mass peak of the fission-fragment mass distribution. As a result, ^{99}Mo fission yields are essentially the same for fission induced by thermal-neutron and fission-neutron spectra. Consequently, the average yield from both thermal and fission spectra neutrons is given for ^{235}U and ^{239}Pu . It should be mentioned, however, that ^{99}Mo fission yields are known to fall off for neutron energies above 4 MeV. Probably these same conclusions apply to the fission of ^{233}U . Unfortunately, only one measurement of the ^{99}Mo yield in the fast fission of ^{233}U has been reported.

Recommended ^{99}Mo fission yields are given in Table I.A.2 along with results from other surveys.

b. FFTF Critical Assembly Experiments--Planning and Evaluation
(A. Travelli)

Last Reported: ANL-7606, pp. 15-17 (Aug 1969).

(i) Calculation of Worth Traverses of ^{10}B in ZPR-3 Assembly 56B. The axial variations of the worth of a small sample of ^{10}B in ZPR-3 Assembly 56B have been calculated. The flux distribution was obtained using the 29-group cross-section set 29004.2 (see Progress Report for March 1969, ANL-7561, p. 7) in one-dimensional slab diffusion theory. The MACH-1 code* was used in the calculations.

TABLE I.A.2. Recommended ^{99}Mo Fission Yields (%)^a

^{233}U		^{235}U		^{238}U	^{239}Pu		^{241}Pu	Reference
Thermal	Fission	Thermal	Fission	Fission	Thermal	Fission	Thermal	
5.12 ± 0.12	4.75 ± 0.35	6.13 ± 0.09	6.13 ± 0.09	6.29 ± 0.20	5.93 ± 0.13	5.93 ± 0.13		This work
			6.1	6.3		5.9		(15)-1968
		6.16	6.1	6.56	5.9	5.88	6.0	(18)-1968
4.96 ± 0.25	4.75 ± 0.35	6.14 ± 0.16	6.14 ± 0.4	6.2 ± 0.15	5.86 ± 0.18	5.86	6.0	(19)-1967
5.12 ± 0.14		6.19 ± 0.16			5.9 ± 0.2		6.0	(20)-1967
						5.78 ± 0.24		(13)-1965
4.80		6.06	6.1	6.3	6.10	6.0		(21)-1960
5.1 ^b		6.2		5.9	6.1	5.9		(1)-1951

^aSee Table I.A.3 for references.

^bSee Ref. 17.

NOTE: Errors refer to the 95% confidence level.

*Meneley, D. A., et al., ANL-7223 (1966).

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The radial leakage was simulated by a DB^2 absorber. The transverse buckling ($1.298 \times 10^{-3} \text{ cm}^{-2}$) was obtained through a series of cylindrical and slab calculations converging by iteration to a pair of calculations corresponding to the same k and to transverse bucklings adding up to the material buckling ($1.939 \times 10^{-3} \text{ cm}^{-2}$). The dimensions and material compositions of the cylindricized reactor are given in ANL-7561, pp. 9-10. The value of k obtained through this procedure was 0.988.

Two worth traverses were computed from the real and adjoint neutron flux distributions of the unperturbed reactor.

A first calculation of the ^{10}B worth at each position was obtained by means of first-order perturbation theory as applied by the DEL option of the MACH-1 code. The 29-group ^{10}B cross sections used in this calculation were obtained from a 2100-group fundamental mode MC^2 * flux averaging for the core composition at criticality.

A second and improved calculation of the ^{10}B worth traverse was obtained by applying a correction for self-shielding to the results obtained from perturbation theory. Group-dependent self-shielding factors were derived for the ^{10}B sample in the same way they were derived for Assembly 51 (see Progress Report for January 1969, ANL-7548, pp. 14-15).

The sample was considered to be an infinitely long, thin, hollow cylinder with an isotropic nonspace-dependent neutron flux impinging on its surface. For each energy group the self-shielding factor was obtained by use of the method described in Progress Report for April-May 1969, ANL-7577, pp. 29-33.

Fig. I.A.3. ^{10}B Worth Traverses in ZPR-3 Assembly 56B

The computed axial worth traverses are compared with the experimental values from ANL-7577 in the upper half of Fig. I.A.3, where the experimental data are indicated by crosses. The value at the center of the assembly is normalized to unity. The worth computed in the first calculation (first-order perturbation theory) is indicated by the dotted curve labeled "1." The worth computed in the second calculation (first-order perturbation theory modified for self-shielding) is indicated by the dotted curve labeled "2." Both calculated curves are also normalized to unity at the center of the assembly. By comparison of curve 1 with curve 2 it can be seen that the effect of self-shielding on the curve shape is not negligible, especially near the reflector where the spectrum is relatively soft. The unnormalized values of experimental and computed worth are compared in Table I.A.4.

*Toppel, B. J., et al., ANL-7318 (1967).

TABLE I.A.4. Axial ^{10}B Worth Traverse

Positions (in. above midplane)	Specific Worths (lh/kg)			Positions (in. above midplane)	Specific Worths (lh/kg)		
	Experiment $\pm 3\sigma$	Perturbation Theory	Perturbation Theory Corrected for Self-shielding		Experiment $\pm 3\sigma$	Perturbation Theory	Perturbation Theory Corrected for Self-shielding
0	-6692	-7782	-7163	15.5	-2457	-2991	-2583
1	-6677	-7749	-7131	16	-2335	-2921	-2471
2	-6557	-7654	-7044	17	-2253	-2877	-2322
4	-6213	-7281	-6700	18	-2113	-3046	-2313
6	-5729	-6689	-6152	19	-1980	-3127	-2251
8	-5085	-5927	-5447	20	-1721	-2825	-1984
10	-4365	-5060	-4638	22	-1174	-1819	-1252
12	-3634	-4175	-3802	24	-607	-907	-627
14	-2906	-3396	-3034	26	-330	-336	-233
15	-2604	-3100	-2717	28	-130	-55	-36

Radial variations of the worth of a small sample of ^{10}B in ZPR-3 Assembly 56B have also been calculated in a completely similar manner, except that the real and adjoint neutron flux distributions were obtained from a one-dimensional cylindrical-diffusion calculation. The traverse buckling ($6.406 \times 10^{-4} \text{ cm}^{-2}$) and the value of k (0.988) were consistent with the previous axial calculation.

The normalized experimental and computed radial ^{10}B worth traverses are compared with each other in the lower half of Fig. I.A.3. The unnormalized values are given in Table I.A.5.

TABLE I.B.5. Radial ^{10}B Worth Traverse

Positions (in. from core center)	Specific Worths (lh/kg)			Positions (in. from core center)	Specific Worths (lh/kg)		
	Experiment $\pm 3\sigma$	Perturbation Theory	Perturbation Theory Corrected for Self-shielding		Experiment $\pm 3\sigma$	Perturbation Theory	Perturbation Theory Corrected for Self-shielding
0	-6617	-7782	-7163	9.662	-4489	-5261	-4817
0.498	-6649	-7770	-7158	10.039	-4243	-5103	-4668
0.998	-6622	-7750	-7134	10.914	-4031	-4733	-4315
1.498	-6561	-7714	-7070	13.094	-3231	-3890	-3487
2.186	-6515	-7632	-7022	15.278	-2674	-3521	-3085
2.683	-6514	-7556	-6954	16.366	-2447	-3312	-2783
3.183	-6371	-7463	-6869	17.460	-2228	-3365	-2638
3.683	-6311	-7357	-6769	18.550	-2284	-3799	-2738
4.058	-6155	-7273	-6686	19.644	-1956	-3699	-2455
4.368	-6163	-7195	-6614	20.734	-1453	-2379	-1587
4.865	-6045	-7056	-6488	21.826	-924	-1283	-881
5.366	-5919	-6908	-6352	24.008	-242	-359	-266
5.864	-5774	-6745	-6202	26.190	-51	-90	-73
7.606	-5207	-6111	-5612	28.372	-22	-20	-17
8.732	-4766	-5654	-5185	30.554	+2	-2	-2
9.161	-4631	-5475	-5018	31.864	-17	+0.7	+0.7
9.414	-4555	-5368	-4917				

3. ZPR-6 and -9 Operations and Analysis

a. Zoned Critical Experiments (L. G. LeSage)

Last Reported: ANL-7606, pp. 17-19 (Aug 1969).

Assembly 25 of ZPR-9 was constructed. The critical mass was 468 kg, and the final loading pattern is shown in Fig. I.A.4. The loading was

Fig. I.A.4. Loading Arrangement for
ZPR-9 Assembly 25

the same in each half of the assembly. The coverage worth of the dual-purpose control rods was measured as 111 lh, which gave a total shutdown of 2.56% $\delta k/k$. The measured integral worth of the fine autorod was -4.40 lh.

Measurements of reactivity worth were made to determine the exact null composition, and the control test zone was reloaded to this composition. The preliminary null-composition weight ratios were:

Materials	Ratios
Depleted Uranium/Enriched Uranium	15.06
Stainless Steel/Enriched Uranium	1.014
Aluminum/Enriched Uranium	0.091

4. ZPR-3 and ZPPR Operations and Analysis

a. Support for Routine Operation of the Critical Facilities (P. I. Amundson and R. G. Matlock)

Last Reported: ANL-7595, pp. 17-19 (July 1969).

(i) ZPPR Operation. Initial portions of this reporting period were used for safety tests. The entire containment system was tested at the completion of facility construction in 1968. The tests completed during the current reporting period represent the first verification of the containment system since reactor startup. Results indicate that the leak rate is entirely satisfactory and that the scheduling outlined in the ZPPR Operating Limits is indeed adequate to insure the operating safety of the reactor containment system.

The remainder of this period was devoted to the FTR Program (see Sect. d below).

(ii) ZPR-3 Operation. A high-power run concluding the experimental program for Assembly 57 has been completed. A power history of approximately 400 kWhr was achieved. Unloading operations are presently underway.

(iii) Computer Developments. Development has continued on improving the evaluation of the effective source term in inverse kinetics reactivity evaluations. The method developed by Carpenter to determine the source which gives the minimum root-mean-square deviation of the average value of reactivity (over the range of data where reactivity is constant)

has been added to the 840 MP inverse kinetics code. Additional modifications to the program are being made to speed convergence to the correct value of effective source.

This program was used in the 14-step loading change in the FTR-2 loading in ZPPR to give an accurate, rapid determination of the degree of subcriticality at each stage of the loading change.

Among programs completed were a magnetic-tape-handler program for the SEL-840 MP to read tapes written by the SEL-810A computer-analyzer, and a routine to read thermocouple data from the incremental magnetic tape in the ZPPR Data Acquisition System.

(iv) Pu-241 Reactivity Effects. One-dimensional diffusion and perturbation calculations, based on ENDF/B cross sections, have been performed to estimate the worths of Pu-241 and Am-241 in FTR Core I as built on ZPR-3 (Assembly 56B) and ZPPR. The results indicate that the Pu-241 decay contributes significantly to the observed reactivity loss of both reactors. The discrepancy between calculated and observed values in ZPR-3 as shown in Table I.A.6 can be partially attributed to temperature variations in reactor and cell. ZPPR, which has excellent reactor and cell temperature stability and monitoring capability, shows no such discrepancy.

TABLE I.A.6. Comparison of Measured and Calculated Reactivity Losses in ZPR-3 and ZPPR

ZPR-3			ZPPR		
Time (days)	Meas (lh)	Calc (lh)	Time (days)	Meas (lh)	Calc (lh)
5	2.57	1.18	6	1.1	1.42
57	23.05	13.52	7.5	1.86	1.78
61.5	23.72	14.58	26	6.65	6.16

b. Clean Critical Experiments

The experimental program with Assembly 57 has been completed. The final two experiments consisted of 1) ≈ 4 -kWhr irradiation of a foil packet for fission product analysis and 2) ≈ 400 -kWhr irradiation of high-purity fissile foil packets for alpha analysis (the ratio of capture-to-fission cross sections).

(i) Delayed Photoneutron Analysis. An error was discovered in the results of a computer calculation which were used in another problem to calculate the effective beta of the delayed fission product neutrons; although the change is expected to be small, the analysis awaits the results of corrected calculations. At present, the analysis indicates a delayed photoneutron fraction of ≈ 0.10 of the value reported by Keepin.*

*Keepin, G. R., The Physics of Nuclear Kinetics (1965).

(ii) Reaction Rate Traverses. The axial and radial reaction rate traverses were performed with counters described in Progress Report for May 1968, ANL-7457, p. 23. The active length of the B-10 counter is 2 in.; the active lengths of the other counters are 1 in. The positions given in the tables below refer to the location of the center of the counter.

The axial traverses were performed through a 0.5-in.-dia tube located in the center of a 0.75 x 1.0-in. void loaded through 1- and 2-P-14, the central core drawer. The reactor power level was held constant by an autorod. Counts were taken for a fixed time interval, and every fifth or sixth count was a repeat for the core midplane. All midplane counts were within 2σ of the average; hence, the sum of all central counts was used to calculate the uncertainty at the center. In all cases, except for the U-238 counter, the center of the counters was positioned at the midplane before the data were taken. It was noticed during the U-238 traverse that the counter had not been centered. The displacement was measured afterwards and the positions were corrected. In general, the absolute position of the counter centers is known to ± 0.015 in. and the relative traverse positions to within ± 0.002 in. The data are given in Tables I.A.7 and I.A.8, and shown in Figs. I.A.5 through I.A.8.

TABLE I.A.7. Axial Reaction Rate Traverses for ZPR-3 Assembly 57

U-235			U-238			U-235			U-238		
Counter Position (in.)	Relative Countrate	RMS Error (%)	Counter Position (in.)	Relative Countrate	RMS Error (%)	Counter Position (in.)	Relative Countrate	RMS Error (%)	Counter Position (in.)	Relative Countrate	RMS Error (%)
29	0.01190	2.47	29.12	0.000345	8.56	6	0.721	1.05	6.12	0.674	0.83
27	0.02706	1.65				5	0.798	1.01	5.12	0.782	0.79
25	0.0700	1.06				4	0.870	0.97	4.12	0.869	0.77
23	0.1245	0.83	24.12	0.00115	4.72	3	0.929	0.93	3.12	0.931	0.77
21	0.1748	1.16	22.12	0.00209	3.52	2	0.956	0.93	2.12	0.966	0.76
19	0.2427	1.00	20.12	0.00391	2.60	1	0.988	0.91	1.12	0.991	0.75
17	0.310	0.89	18.12	0.00881	5.38	0	1.000	0.37	0.12	1.000	0.75
15	0.386	0.82	16.12	0.0157	4.04	-1	0.989	0.91	-0.88	0.998	0.24
13	0.472	0.76	14.12	0.0333	2.81	-2	0.961	0.92	-1.88	0.982	0.75
12	0.508	0.73	12.12	0.0712	1.96	-3	0.903	0.95	-2.88	0.944	0.76
11	0.537	0.71				-4	0.857	0.97	-3.88	0.883	0.77
10	0.558	1.17	10.12	0.159	1.38	-5	0.787	1.01	-4.88	0.806	0.79
9	0.573	1.17	9.12	0.248	1.15	-6	0.704	1.06	-5.88	0.703	0.82
8	0.608	1.14	8.12	0.383	0.99	-7	0.609	1.13	-6.88	0.581	0.86
7	0.651	1.10	7.12	0.555	0.87	-8	0.584	1.15	-7.88	0.420	0.95

TABLE I.A.8 Axial Reaction Rate Traverses for ZPR-3 Assembly 57

Pu-239			B-10		Pu-239			B-10	
Counter Position (in.)	Relative Countrate	1 σ (%)	Relative Countrate	1 σ (%)	Counter Position (in.)	Relative Countrate	1 σ (%)	Relative Countrate	1 σ (%)
30			0.0134	1.30	7.5	0.637	0.73	0.732	0.64
29	0.01380	1.42	0.0199	1.09	7	0.665	0.72	0.738	0.64
27.5			0.0361	0.84	6	0.734	0.69	0.773	0.63
27	0.0306	0.99	0.0451	1.23	5	0.805	0.66	0.838	0.62
25	0.0727	0.69	0.1144	0.82	4	0.876	0.65	0.891	0.60
23	0.1366	0.56	0.2289	0.63	3	0.927	0.63	0.944	0.59
21	0.1955	1.21	0.340	0.87	2	0.975	0.62	0.972	0.59
19	0.2662	1.05	0.464	0.76	1	0.995	0.61	0.988	0.59
17	0.358	0.92	0.603	0.69	0	1.000	0.27	1.000	0.22
15	0.433	0.86	0.740	0.64	-1	0.987	0.61	0.988	0.59
14	0.475	0.82			-2	0.959	0.62	0.967	0.59
13	0.517	0.79	0.842	0.62	-3	0.929	0.63	0.927	0.59
12	0.549	0.77	0.878	0.61	-4	0.870	0.65	0.876	0.61
11	0.573	0.76	0.870	0.61	-5	0.802	0.67	0.814	0.63
10	0.585	0.75	0.842	0.62	-6	0.715	0.69	0.755	0.64
9	0.594	0.74	0.795	0.63	-7	0.643	0.73	0.714	0.65
8.5	0.606	0.74			-8	0.595	0.74	0.720	0.65
8	0.606	0.74	0.738	0.64

Fig. I.A.5. ^{235}U Axial Reaction Rate Traverse for ZPR-3 Assembly 57

Fig. I.A.6. ^{238}U Axial Reaction Rate Traverse for ZPR-3 Assembly 57

Fig. I.A.7. ^{239}Pu Axial Reaction Rate Traverse for ZPR-3 Assembly 57

Fig. I.A.8. ^{10}B Axial Reaction Rate Traverse for ZPR-3 Assembly 57

The radial reaction rate traverses were performed through the center of a 1.125-in.-dia hole, 1 in. back from the midplane. This was the hole size required for the reactivity samples used in the alpha measurement by the reactivity-reaction rate method. The blanket iron and depleted uranium were solid 2 x 2 in. blocks with the hole drilled through them. All core materials also had the same size of hole, so that the plates were vertical in the traverse drawers as well as the rest of the core. A 0.5-in.-dia tube was held in the center of the larger hole by thin plates, and the counter traverse was performed down this tube. The data are given in Table I.A.9 and shown in Figs. I.A.9 through I.A.12.

c. Doppler Experiments (R. Kaiser)

Last Reported: ANL-7595, p. 20 (July 1969).

The heater-power-indicating circuit for the ZPPR reactivity Doppler equipment has been built and tested. The capsule has been heated to 800°C with ceramic end support pieces, and it was found that a 1/16-in.-thick nonperforated ceramic piece is thermally equivalent to the perforated Invar piece used in the original design. It is possible that a thinner ceramic or one that has been perforated to some degree would reduce heat losses while retaining sufficient strength to support the capsule.

TABLE I.A.9. Radial Reaction Rate Traverses for ZPR-3 Assembly 57

Matrix Location	Counter Position (in.)	U-235		U-238		Pu-239		B-10	
		Relative Countrate	1 σ (%)	Relative Countrate	1 σ (%)	Relative Countrate	1 σ (%)	Relative Countrate	1 σ (%)
P-1/P-2	27.288	0.0120	2.86	0.00073	9.8	0.0130	2.38	0.0174	2.39
P-2/P-3	25.105	0.0307	1.81	0.00107	8.1	0.0303	1.56	0.0445	1.51
P-3	24.013	0.0414	2.22	0.00124	7.5	0.0476	1.76	0.0696	1.20
P-3/P-4	22.922	0.0658	1.27	0.00182	6.2	0.0741	1.42	0.1174	1.31
P-4	21.830	0.1018	1.45	0.00208	8.2	0.1137	1.15	0.1894	1.03
P-4/P-5	20.739	0.1370	1.28	0.00252	7.5	0.1580	0.98	0.2679	0.88
P-5/P-6	18.556	0.2138	0.97	0.00512	5.2	0.2462	0.79	0.433	0.70
P-6/P-7	16.373	0.2951	0.83	0.00103	3.7	0.348	0.67	0.600	0.60
P-7/P-8	14.190	0.386	0.73	0.0239	2.44	0.452	0.60	0.764	0.54
P-8/P-9	12.007	0.485	0.66	0.0558	1.63	0.552	0.54	0.880	0.50
P-9	10.915	0.533	0.63	0.0876	1.32	0.590	0.53	0.918	0.50
P-9/P-10	9.824	0.571	0.61	0.1368	1.08	0.611	0.52	0.917	0.50
P-10	8.732	0.595	0.60	0.2284	1.23	0.628	0.51	0.860	0.51
P-10/P-11	7.641	0.616	0.59	0.363	1.03	0.630	0.51	0.792	0.53
P-11	6.549	0.645	0.58	0.549	0.73	0.668	0.50	0.749	0.54
P-11/P-12	5.458	0.730	0.54	0.671	0.67	0.735	0.47	0.774	0.53
P-12	4.366	0.811	0.52	0.800	0.63	0.825	0.46	0.818	0.52
P-12/P-13	3.275	0.889	0.51	0.871	0.60	0.889	0.44	0.888	0.50
P-13	2.183	0.948	0.49	0.947	0.58	0.946	0.44	0.949	0.50
P-13/P-14	1.092	0.978	0.48	0.973	0.57	0.979	0.43	0.984	0.48
P-14	0	1.000	0.18	1.000	0.22	1.000	0.17	1.000	0.18

Fig. I.A.9.- ^{235}U Radial Reaction Rate Traverse for ZPR-3 Assembly 57

Fig. I.A.10. ^{238}U Radial Reaction Rate Traverse for ZPR-3 Assembly 57

Fig. I.A.11. ^{239}Pu Radial Reaction Rate Traverse for ZPR-3 Assembly 57

Fig. I.A.12. ^{10}B Radial Reaction Rate Traverse for ZPR-3 Assembly 57

d. Mockup Critical Experiments

The experiments with ZPPR Core 1, which corresponds to Assembly II of the FTR Resumed Phase-B Critical Experiments Program, were interrupted to perform the required annual containment tests on ZPPR. Following the completion of the containment tests, the shield-study portion of the FTR-2 experimental program was initiated. The B_4C control ring has been removed and the sodium-stainless steel shield installed. The removal of the control ring was performed in steps, with the resulting reactivity effects compensated by alternately substituting depleted U-238 for the plutonium fuel in the center of the core. Inverse-kinetic-type subcritical measurements were made and neutron count data obtained with the three U-235 fission chambers (see Progress Report for June 1969, ANL-7581, p. 26) as the removal of the control ring proceeded. The subcritical data are presently being analyzed, and the proton-recoil measurements are in progress.

B. Component Development--LMFBR*

1. Instrumentation and Control

a. Instrumentation Development for Instrumented Subassembly

(i) Instrumented-Subassembly Flowmeters (G. A. Forster)

Last Reported: ANL-7595, p. 27 (July 1969).

(a) Procurement of Mark-III (Sodium-calibratable) Flowmeters. An order was placed for four flowmeters to be used in the instrumented subassembly tests in EBR-II; an option was obtained to increase the quantity at a later date. A Quality Assurance Plan of Action was developed that delineates the responsibilities of the ANL inspectors in overseeing manufacturer performance in order to assure conformance to the specifications.

(b) Modification of Calibration Loop for Mark-III Flowmeters. A new dump tank is being designed to ASME Code requirements for service up to 1200°F and 50 psig. An existing tank is being rehabilitated for use up to 700°F. Special vapor traps were designed and are being constructed; they will be used to remove sodium vapor from the cover gas that flows during test runs. The loop heater system and power wiring has been completed and tested.

(ii) Instrumented-Subassembly Coolant Thermocouples (R. J. Dickman)

Not reported previously.

(a) Tests of Prototype Coolant and Spacer-wire Thermocouples. Coolant and spacer-wire thermocouples are being tested in preparation for use in the Test-I and Test-II instrumented subassemblies for EBR-II. Collection, storage, and distribution of thermocouple data from the EBR-II tests are being planned. Test II will produce 23 sensor records, each in both analog and digital forms, during the three-month in-core test.

(b) Improved Specifications for Coolant and Spacer-wire Thermocouples for Future 19-pin Assemblies. Results of vendor and ANL quality-assurance tests for 172 thermocouples have been entered on IBM cards. A simple, flexible computer program is being written to analyze these data statistically.

*Sodium Technology Development is now reported under Sect. I.C.

b. EBR-II In-core Instrument Test Facility (E. Hutter)

(i) Preliminary Studies of Various Schemes (O. S. Seim and R. C. Brubaker)

Last Reported: ANL-7606, pp. 39-40 (Aug 1969).

One of the goals for the facility is the capability to provide a high-temperature sodium environment in the upper portion of the irradiation zone. A relatively small quantity of sodium is heated when passing through the thimble of the facility and is discharged from the thimble at low pressure into the bulk sodium. To avoid potential thermal-stress problems, the sodium must be cooled within the thimble before it is discharged. This cooling can be obtained by terminating an insulating gas barrier in the thimble just above the reactor-vessel cover and allowing heat conduction to the bulk sodium. Calculations indicate that cooling to within 100°F of the bulk-sodium temperature (equivalent to an 800°F thimble outlet temperature) can be obtained in a length of 2 ft for all expected operating conditions.

2. Fuel Handling, Vessels and Internals

a. Core Component Test Loop (CCTL) (R. A. Jaross)

Last Reported: ANL-7606, pp. 40-41 (Aug 1969).

(i) Loop Modifications to Accomodate Second FFTF Subassembly. The loop modification work is proceeding on schedule. All welding on the 4-in. flow tube and pump-bypass piping has been completed. Only pipe heaters and thermal insulation remain to be installed on these additions to the loop. Radiography of all loop pipe welds made accessible during the modifications has been completed; no deterioration of the welds was found. Installation of continuity-type leak-detection wires around each of the five loop-drain valves has been completed.

The new inlet piping, plenum, and grid structure have been fabricated and installed in the test vessel. The remaining fabrication work is approximately 80% complete.

C. Sodium Technology

Beginning with this issue, all of the work in the Laboratory's sodium technology program will be reported in this section. It will include items previously reported under:

- (1) Liquid Metal Fast Breeder Reactors--Civilian, C. Component Development--LMFBR (ANL-7606, p. 31);

- (2) Other Fast Reactors--Other Fast Breeder Reactors--
Fuel Development, B. Coolant Effects on Cladding Alloys
(ANL-7606, p. 86);
- (3) General Reactor Technology, B. Reactor Fuels and Materials
Development (ANL-7577, p. 153), and D. Chemistry and
Chemical Separations (ANL-7606, p. 118).

In this report, titles for the sodium technology entries will also be shown in the position they previously occupied, with reference to their present location.

The major objectives of the ANL program are to:

- (1) provide base sodium technology support to EBR-II and other ANL-operated national test facilities;
- (2) demonstrate selected processes and equipment using EBR-II as a test facility;
- (3) advance sodium technology in selected fundamental areas of sodium chemistry and materials compatibility.

This work is planned to meet the needs of the LMFBR Program Plan and provide support for the EBR-II and FFTF Projects.

1. Sodium Chemistry

The basic work on sodium chemistry is directed toward the development of a sound scientific foundation for understanding the behavior of common nonmetallic contaminants in sodium, for interpreting and evaluating existing corrosion data, and for predicting potential corrosion problems in sodium systems.

a. Studies on Carbon Transport in Sodium-Steel Systems (F. A. Cafasso)

Last Reported: ANL-7577, p. 176 (April-May 1969).

It has been postulated* that disodium acetylide (Na_2C_2) may participate in the decarburization of ferritic steels by liquid sodium. In an effort to substantiate this postulate, the dependence of acetylide concentration in sodium on temperature, time, and activity of carbon in the ferritic steel is being studied.

*Luner, C., Feder, H. M., and Cafasso, F. A., "Carbon Transport in Liquid Sodium," in Proceedings of the International Conference on Sodium Technology and Large Fast Reactor Design, ANL-7520, Part I, p. 455 (1968).

As a first step, experiments are being made to determine whether an equilibrium involving acetylide exists between sodium and selected ferritic steels or iron-carbon alloys. Approximately 1 g of steel or alloy foil and 10 g of sodium are sealed in copper crucibles and heated for various periods of time in the temperature range from 500 to 700°C. The acetylide content of the sodium retrieved from the crucibles is determined by gas chromatographic analysis of the off-gas from aqueous dissolution of the sodium. Other carbon analyses performed as needed include the total carbon content of the sodium and the change in carbon content of the steel.

The initial experiments were made with an iron-carbon alloy containing 0.11 wt % carbon. Significant amounts of acetylide were found in the equilibrated sodium: 6 ± 2 ppm at 650°C and 0.08 ± 0.02 at 400°C. These levels were obtained within 24 hr and did not vary significantly over periods up to 5 days. Apparently, an equilibrium or steady-state concentration of acetylide was established rapidly in the liquid sodium.

Similar equilibration experiments are now being made with Croloy 2 1/4. Preliminary results show that at 400°C a small but significant amount of acetylide is produced.

b. Determination of the Solubility of Gases in Liquid Sodium

(i) Determination of the Solubility of Helium in Sodium
(F. A. Cafasso)

Last Reported: ANL-7581, p. 110 (June 1969).

The solubility in sodium of potential cover gases is needed by reactor designers and engineers to evaluate the possibility that exsolution of gas in the coolant sodium circuit will lead to cavitation and/or bubble formation and thus result in diminished sodium flow. Measurements of the temperature dependence of argon, nitrogen, and helium solubilities were reported previously.* Recently, the dependence of helium solubility on pressure was measured, and the results are summarized below.

The experimental procedure for measuring helium solubility as a function of pressure was basically the same as that described in the Progress Report for June 1969, ANL-7581, pp. 110-111. Fourteen experiments were carried out at ~500°C and at pressures ranging from

*Veleckis, E., et al., in Chemical Engineering Division Semiannual Report July-December 1966, ANL-7325, p. 128 (argon); Proceedings of the International Conference on Sodium Technology and Large Fast Reactor Design, ANL-7520, Part I, p. 295 (nitrogen); Progress Report for July 1969, ANL-7581, p. 110 (helium).

2 to 9 atm. The data (see Fig. I.C.1) have been corrected to exactly 500°C with the aid of the previously determined temperature coefficient of solubility. The experimental points were fitted by the method of least squares

Fig. I.C.1. Pressure Dependence of the Solubility of Helium in Sodium at 500°C

to a linear equation. The slope of this line ($1.56 \times 10^{-7} \text{ atm}^{-1}$) is in very good agreement with the Henry's law constant ($K_H = 1.52 \pm 0.06 \times 10^{-7} \text{ atm}^{-1}$) obtained for 500°C from the temperature study. On the basis of the statistical F-test made on the value of the intercept, the displacement of the line from the origin is insignificant. This can be seen in Fig. I.C.1 by the position of the 95% confidence limits. The solubility data, therefore, obey Henry's law and may be represented by a straight line passing through the origin.

The study of helium solubility in liquid sodium is now complete. Preparations for studying the solubility of xenon, a fission product gas, in liquid sodium are being made.

2. On-line Monitors

The ANL program includes the evaluation and improvement of on-line monitors for oxygen, carbon, and hydrogen.

Particular emphasis will be on checking out United Nuclear Corporation (UNC) oxygen and carbon meters before their installation in the Radioactive Sodium Chemistry Loop (RSCL) at EBR-II.

a. Evaluation and Improvement of Oxygen-activity Meter (J. T. Holmes)

Last Reported: ANL-7581, p. 27 (June 1969).

The cell housing of the UNC oxygen meter is being modified so as to facilitate its insertion and replacement in sodium systems and thus avoid solid-electrolyte failures due to thermal and mechanical shock. Procedures will be developed for filling the assembled probe and housing with sodium prior to insertion into the calibration apparatus. After being calibrated, two complete units (i.e., probes, housings, and sodium) will be installed in the RSCL that is being designed for EBR-II.

Oxygen cells for future use on the RSCL will be tested and calibrated on the Corrosion and Mechanical Properties Loop, now in operation. The techniques of distillation sampling and equilibration with refractory-metal wires will be used to check the oxygen concentration.

D. Systems and Plant Development

1. Plant and Design

a. Contract Management, Technical Review, and Evaluation (L. W. Fromm and K. A. Hub)

Last Reported: ANL-7606, p. 41 (Aug 1969).

The AEC Division of Technical Information Extension (DTIE) plans to have the Tasks-II, -III, and -IV reports by Combustion Engineering and General Electric ready for distribution by the end of September. All remaining reports of Babcock & Wilcox and Westinghouse have been patent cleared and shipped to DTIE for publication. DTIE plans to publish the remaining nine B&W volumes and five Westinghouse volumes at an average rate of one volume per week, which will extend publication to December 19, 1969.

The AEC California Patent Group has cleared all Atomics International Follow-on Study reports, except one volume of the Task-II report. With the present schedule, publication of the AI reports will be completed by February 13, 1970.

(i) Evaluation of 1000-MWe Follow-on Study Task Reports.

The evaluation report on the Westinghouse Task-I report has been edited. It will be sent to the contractor for comment. Evaluation of the Tasks-II, -III, and -IV reports of all contractors is proceeding; approximately 20% of the evaluation reports have been drafted.

E. EBR-II--Research and Development

1. Equipment--Fuel Related (E. Hutter)

a. Higher-worth Control Rod (O. S. Seim and T. Sullivan)

Last Reported: ANL-7581, p. 45 (June 1969).

The design calculations, flowtests, and fabrication drawings for the prototype version of the higher-worth control rod are complete.

A study has been made to determine the feasibility of a higher-worth safety rod. Although such a rod can be designed for the EBR-II, the

amount of B_4C that can be installed within its standard hexagonal tube at 1.2 at. % fuel burnup is less than the amount of B_4C in the higher-worth control rods. Two factors limit the amount of B_4C within the safety rods:

- (1) The length of hexagonal tube above the fuel section is approximately 16 in. less than that of the control rod.
- (2) The safety rods require about 85 gpm of coolant flow, whereas the control rods require only 61 gpm.

Factor (1) limits the length of B_4C that can be installed, and factor (2) limits the cross-sectional area of B_4C that can be installed.

Pressure-drop calculations for several B_4C tube arrangements indicate two acceptable arrays of B_4C tubes. The first array would consist of a cluster of six 5/8-in.-OD tubes within the hexagonal tube instead of the seven 5/8-in.-OD tubes that are in the higher-worth control rods. The other possible array would consist of a cluster of seven 9/16-in.-OD B_4C tubes.

To permit a reasonable length of natural B_4C and sufficient gas space for helium release, the upper shield section within the B_4C tube has been arbitrarily set at 6 in. of steel; for the higher-worth control rod, this same shield section is 8 in. long. Six inches of shielding are believed to be sufficient.

Buildup of helium-gas pressure within the B_4C tubes limits the amount of B_4C that can be installed within the tubes.

2. New Subassemblies and Experimental Support (E. Hutter)

a. Experiment-Irradiation Subassemblies (O. Seim, W. Ware, and E. Filewicz)

Last Reported: ANL-7595, p. 41 (July 1969).

(i) Mark A19A. Hydraulic flowtests of an unorificed Mark-A19A subassembly were performed to determine the characteristics of pressure drop versus flow. The tests were made in the stainless steel pressurized-water loop, and data were collected for reactor row positions 1 through 9. Table I.E.1 compares the typical data obtained from the tests with typical

TABLE I.E.1. Typical Results of Flowtests of Unorificed Mark-A19A and Mark-A19 Irradiation Subassemblies

Reactor Row	Effective Pressure Drop (psi)	Sodium Flowrate at 800°F (gpm)		Reactor Row	Effective Pressure Drop (psi)	Sodium Flowrate at 800°F (gpm)	
		Mark A19	Mark A19A			Mark A19	Mark A19A
1, 2	38	64	97	5	34	48	65
3	39	61	91	6, 7	40.8	-	61
4	34.5	52	73	8, 9	4.2	-	14.4

data obtained from flowtests of the unorificed Mark-A19 irradiation subassembly. The flow through the Mark A19A is higher and closer to the flow through a standard EBR-II core subassembly (see Table I.E.2).

TABLE I.E.2. Reference Values of Flow vs Pressure Drop for a Standard EBR-II Core Subassembly

Reactor Row	Effective Pressure Drop (psi)	Sodium Flowrate at 800°F (gpm)
1, 2	38	139
3	39	123
4	34.5	93.5
5	34	78
6	40.8	70
8, 9	4.2	5.9

(ii) Mark-J19 Design and Pressure-drop Characteristics
(O. S. Seim and W. Ware)

The Mark-J19 irradiation subassembly was developed as a result of an experimenter's request for a vehicle to test element-cladding material at higher than normal EBR-II reactor temperatures. An attempt was made to use existing Mark-A19 and Mark-C19 irradiation subassemblies, but the Mark-A (shroud tube) type was eliminated due to the difficult bypass and element coolant-mixing problem. The Mark C19 did not provide adequate gas space for the high burnup desired in this long-term irradiation experiment.

The Mark-J subassembly evolved as a result of the investigations of the two above subassemblies. It uses the same type of coolant-mixing section as the Mark C, but provides additional element length above the core. A more versatile and efficient flow-orificing system also is used. In addition to these improvements, an insulating gas plenum is provided to prevent excessive heat loss from the element bundle to the cool bypass flow.

Design criteria for the Mark J19 are:

- a) Provide for a total of 19 test capsules, each 40 in. long x 0.375 in. OD, and containing a 0.250-in.-OD element wrapped with 0.030-in.-dia spiral spacer wire to give a pitch-to-diameter ratio of 1.62.
- b) Provide a maximum-length gas plenum above the core.
- c) Insulate the capsule bundle from the cold bypass flow by an argon-filled gas annulus.
- d) Provide a capsule-cladding surface temperature of 1250°F (with uncertainty factor).

e) Assure a mixed-coolant outlet temperature of 910°F (without uncertainty factor).

f) Assure maximum structural integrity of the sub-assembly through proper mechanical design to reduce thermal stresses created by temperature extremes.

g) Provide a flow-orificing mechanism to assure accurate and predictable temperature control of the capsule cooling and bypass flows.

h) Include adequate radiation shielding to protect reactor structures.

For this particular experiment, the experimenter will provide fuel with a power generation of 16.3 kW/ft (average for pin with maximum power generation) in reactor row 4 at 50-MWt reactor power. At this power, each capsule will generate approximately 1020 Btu/min , and the subassembly will generate $1,162,800 \text{ Btu/hr}$. The orifice-hole diameters must be sized to provide a total of $\sim 44 \text{ gpm}$ of coolant flow based on the required mixed outlet temperature of 910°F . As with other EBR-II experimental subassemblies, the orifice sizes of the Mark J19 can be adjusted to meet the requirements of the experimenter.

Two series of flow-versus-pressure-drop tests were performed for Row 4, the reactor row position for which this specific experiment is intended. Flowtests for Rows 1, 2, 3, 5, and 6 will be performed later.

The Mark J19 (see Fig. I.E.1) was designed to utilize as many standard hardware items as possible to reduce the fabrication of long-lead-time items. The top fixture, outer hexagonal tube, and lower adapter are standard. New hardware includes the inner hexagonal-tube assembly, the grid with grid support, and the lower shield-and-orifice assembly.

The inner hexagonal-tube assembly is a double-wall structure in which the inner hexagonal tube fits closely over the liner hexagonal tube from the top (mixer section) down to the upper edge of the reactor core. From the upper edge of the core down to the grid support, an annulus between the two tubes provides the insulating gas space. Both tubes are attached to each other by a circumferential weld at the top of the core which traps argon gas in this annulus when the subassembly is loaded into the reactor. The two tubes may expand and contract independently without exceeding the allowable stress.

The circumferential weld is protected from extreme differential temperatures by overlapping the inner hexagonal tube over the weld area in a manner that reduces the temperature differential to

a value almost equal to the temperature differential across the liner hexagonal tube (see section B-B in Fig. I.E.1). The walls of the liner hexagonal tube and the inner hexagonal tube are stepped to provide a 0.020-in.-wide annular gas space between the tubes from the top to the bottom of the core (a distance of about 14 in.) and a 0.040-in.-wide annulus from the bottom of the core to the grid (a distance of about $7\frac{1}{2}$ -in.).

Fig. I.E.1. EBR-II Mark-J19 Irradiation Subassembly

When the subassembly is lowered into the primary sodium, argon gas from the primary blanket gas is trapped in the annulus at ~ 1 atm. At operating temperature and pressure, the sodium level will rise in the annulus to the lower edge of the core. Thus the insulating gas is maintained in the core region to prevent excessive loss of heat from the capsule bundle.

Sodium coolant enters the lower adapter through the standard orifice holes, and passes up through the lower end of the radiation shield and outward, into the annulus between the outer hexagonal tube and the upper end of the radiation shield. At that point, the flow divides into two paths: the bypass (cool) flow and the element (hot) flow.

The bypass flow must pass through the orifice plate that contains orifice holes of the proper size to give the correct exit temperature. The flow then passes through the annular space between the outer and

inner hexagonal tubes and up to the mixing section. The mixing section is immediately above the upper ends of the test capsules.

The element (hot) flow separates from the total flow at the orifice holes located in the upper end of the radiation shield. These holes are sized to provide the correct temperature of the fuel-capsule cladding. After passing through the test section, the element flow enters the mixing section and mixes with the cooler bypass flow. Proper balancing of the element and bypass flows is necessary to provide an exit temperature of 910°F.

Mixing is accomplished when the cool bypass flow leaves the annular passage through 96 holes, each of 1/8-in. diameter, and enters the mixing section. The holes are sized and positioned to create a turbulent, swirling action within the mixing section. Photographic studies made during the development of the Mark-C19 irradiation subassembly, which has the same type of mixing section, indicate complete blending of the element flow with the bypass flow before the mixed flow leaves the subassembly.

Flowtests of the Mark J19 were conducted in the pressurized-water test loop. To ensure accurate data, the loop was first calibrated with the standard EBR-II core subassembly. The Mark J19 was completely assembled with orifice plate and dummy capsules. Appropriate bushings were installed on the lower adapter to simulate Row-4 inlet restrictions. Results for each datum point, as indicated by the digital readout instruments, were converted to values for 800°F sodium, using density and viscosity corrections for each point.

Readouts of the bypass flow velocity were taken by a small pitot tube in the bypass annulus. These readouts were converted to quantity of flow for the corresponding values of total flow and pressure drop. Thus, for each size of bypass orifice hole and each size of element orifice, a family of curves was drawn to show total flow through the subassembly, quantity of flow through the bypass annulus, and quantity of flow through the element section.

The flow values for the tested orifice sizes in Row 4 are shown for the effective pressure drop of 34.5 psi in Table I.E.3. By varying the orifice sizes, an infinite combination of flow variations can be achieved. Though not documented here, this feature was demonstrated and recorded at both extremes (high bypass flow with no element flow and no bypass flow with high element flow). Tests to demonstrate this feature were first performed with the bypass orifice holes plugged while the element orifice holes were open and then with the element orifice holes plugged while the bypass orifice holes were open.

The hydraulic flowtests of the Mark J19 indicate that its design is satisfactory and will provide an infinite number of bypass and element-flow combinations. With the proper information from the

experimenter (i.e., the expected power generation of the test fuel) it will be possible to achieve the desired test-element temperature with reliability.

TABLE I.E.3. Flowtest Values for EBR-II Mark-J19 Irradiation Subassembly^a

	Flow of 800°F Sodium (gpm)		
	Bypass	Element	Total
<u>Test 1</u>			
(Bypass orifice with six 1/16-in.-dia holes; element orifice with six 3/16-in.-dia holes)	14	29	43
<u>Test 2</u>			
(Bypass orifice with six 5/64-in.-dia holes; element orifice with six 5/32-in.-dia holes)	17	23	40

^aAll values are for Row-4 position and 34.5-psi effective pressure drop. Reference flow for a standard EBR-II core subassembly under these conditions is 93.5 gpm of 800°F sodium.

The design criteria have been achieved in all respects and tested to the fullest extent possible in the laboratory. The only new concept which has not been tested is the insulating annulus of trapped argon gas in the inner hexagonal-tube assembly. No tests have been performed as of now to verify the weld integrity and to ensure that the gas insulation will be maintained. Loss of the gas, however, will in no way compromise reactor safety or create any hazard to the reactor or other subassemblies. The worst possible effect will be a reduction in the temperature of the experimental capsules. Possible outleakage of the insulating gas might be determined during interim examination of the subassembly by postirradiation leaktesting of the welds.

3. Instrumented Subassemblies (E. Hutter and A. Smaardyk)

a. Assembly of Test-2 Subassembly (C. Divona)

Not previously reported.

The assembly of the test-2 instrumented subassembly is identical to that of test 1, which was described in the Progress Reports for February 1969, ANL-7553, p. 47, and March 1969, ANL-7561, p. 31. Externally, the subassembly resembles a standard control rod except for the instrument leads protruding from the top adapter. The instrument leads are brazed into a bulkhead that forms a partition between the primary-tank sodium and the drywell space within the extension tube. The sheathed instrument leads are connected to flexible cables within this drywell space, and suitable pressure-transducer junctions are made.

Final assembly of test 2 has progressed to the point in which the pressure-transducer junctions were brazed (see ANL-7561, p. 30).

Some of the major assembly and testing steps completed so far include: assembly of the 19-capsule bundle and the electromagnetic flowmeter into a subassembly; tensile testing of the completed subassembly; assembly of the subassembly and drywell; brazing the instrument leads into the bulkhead; and brazing the pressure-transducer and flux-monitor junctions in the drywell space.

b. The EBR-II Test-3 Instrumented Subassembly

Last Reported: ANL-7581, p. 49 (June 1969).

(i) Design. The test-3 instrumented subassembly incorporates a bundle of 36 fuel elements and one conduit tube installed on a 37-element grid assembly. The conduit tube allows instrument leads to pass through the fuel-element bundle. Therefore, both ends of the tube must have openings large enough to accommodate the leads, and the tube must be supported by the grid assembly.

A method has been developed to anchor the conduit tube to the grid assembly while providing a bottom opening large enough for the flowmeter lead to pass through. (This lead is the only one that will penetrate the grid assembly.) In this method, a section is cut out of the grid bar below the conduit tube. The conduit tube is supported by a special adapter tip that engages the ends of the cut grid bars. A hole for the flowmeter lead passes vertically through the tip and between the ends of the grid bars. The ends of the grid bars are supported by a pair of side plates welded to the hexagonal grid tube.

Special tools for fabricating this modified grid assembly are being built. When the tools are completed, a model of the grid assembly will be built and used in a subassembly flowtest to verify the adequacy of the design.

(ii) Flowtesting. A model of the test-3 instrumented subassembly was tested in the water test loop. A dummy Mark-III flowmeter was included in the model. Conversion of the water-test data to sodium at 765°F gave an overall pressure drop of 30.3 psi at 52.8 gpm (pressure drop through the fuel bundle was 7.85 psi). These data show that the flowrate for the subassembly is adequate.

The flowtest model is being modified to include the conduit-tube assembly and the special grid required to support the conduit. Flowtests will be conducted with the modified model, during which the size of the lower-adapter flow orifices will be established.

4. Coolant Chemistry (D. W. Cissel)

- a. Sodium Coolant Quality Monitoring and Control (W. H. Olson, C. C. Miles, T. P. Ramachandran, E. R. Ebersole, and G. O. Haroldsen)

Last Reported: ANL-7595, pp. 42-45 (July 1969).

(i) Radionuclides in Sodium. Table I.E.4 lists the results of analyses for ^{137}Cs and ^{131}I in primary sodium. The ^{22}Na activity on 9/11/69 was $3.8 \times 10^{-2} \mu\text{Ci/g}$.

TABLE I.E.4. ^{137}Cs and ^{131}I in Primary Sodium^a

Sample Date	Sample Size (g)	Sample Flow (gpm)	Flush Time (min)	Sample Temperature (°F)	^{137}Cs ($\mu\text{Ci/g} \times 10^2$)	^{131}I ($\mu\text{Ci/g}$)
7/16/69	13.3	0.3	15	540	1.3	1.9×10^{-4}
7/21/69	12.9	0.3	15	525	1.3	3.4×10^{-4}
7/28/69	13.5	0.3	20	530	1.2	2.4×10^{-4}
8/12/69	13.4	0.3	10	560	1.1	1.4×10^{-4}
8/21/69	13.7	0.5	10	545	1.1	1.2×10^{-4}
8/27/69	12.8	0.3	10	485	1.2	1.3×10^{-4}
9/2/69	12.6	0.4	10	520	1.2	-
9/8/69	13.3	0.3	10	480	1.2	1.0×10^{-4}
9/11/69	12.5	0.5	13	480	1.4	9.0×10^{-5}

^aSamples taken in Pyrex beakers.

The ^{24}Na activity of secondary sodium on 8/19/69 was $2.2 \times 10^{-2} \mu\text{Ci/g}$. Saturation activity at 50 MWt is $\sim 3 \times 10^{-2} \mu\text{Ci/g}$.

(ii) Trace Metals in Sodium. Table I.E.5 lists the results of analyses for trace-metal impurities in primary and secondary sodium. Metal concentrations were determined by atomic-absorption spectrophotometry following vacuum distillation of the sodium sample. The slightly higher values for tin in primary sodium than previously reported are attributed to the different method of sample preparation (vacuum distillation versus solvent extraction).

TABLE I.E.5. Trace Metals in Sodium (ppm)^a

Sample Date	Sample Size (g)	Sample Flow (gpm)	Flush Time (min)	Sample Temperature (°F)	Al	Bi	Ca	Co	Cr	Cu	Fe	Mg	Mn	Mo	Ni	Pb	Sn
<u>Primary Sodium</u>																	
7/28/69	40	0.3	22	540	<0.6	2.1	0.05	<0.02	0.03	0.02	0.06	0.06	<0.005	<0.07	<0.05	11.7	24.5
8/21/69	57	0.3	10	600	<0.6	1.8	0.02	<0.04	0.02	<0.02	0.1	0.01	<0.005	<0.07	<0.05	11.0	22.4
<u>Secondary Sodium</u>																	
7/29/69	62	0.4	10	470	<0.6	<0.1	<0.01	<0.02	0.03	0.02	0.1	0.02	<0.005	<0.07	<0.05	0.7	<0.5
8/21/69	56	0.4	10	510	<0.6	<0.1	<0.02	<0.02	0.02	<0.02	0.1	0.02	0.008	<0.07	<0.05	0.7	<0.5

^aSamples taken in tantalum crucibles.

(iii) Oxygen in Sodium. Table I.E.6 lists the results of analyses for oxygen in sodium by the mercury amalgamation method. Oxygen concentrations as measured by plugging indicators are listed also for comparison.

TABLE I.E.6 Oxygen in Sodium

Sample Date	Sample Container	Sample Size (g)	Sample Flow (gpm)	Flush Time (min)	Sample Temperature (°F)	Aliquot Size (g)	Number of Aliquots	Average Concentration (ppm)	Oxygen Equivalent of Plugging Run ^a (ppm)
Primary Sodium									
7/7/69	Extrusion vessel	15	0.5	15	560	~1	1	4.6	<0.5
8/12/69	Extrusion vessel	15	0.4	15	540	~1	4	8.7 ± 2.9	<0.5
8/12/69	1/2-in. SS tube	(Total time in system, 8 months)				~1	5	7.9 ± 2.3	<0.5
Secondary Sodium									
8/20/69	Extrusion vessel	15	0.5	10	515	~1	5	6.5 ± 3.3	<0.5

^aBased on solubility data from AI-AEC-12685.

(iv) Carbon in Sodium. Table I.E.7 lists the results of analyses for carbon in sodium by the oxyacidic flux method.*

TABLE I.E.7. Carbon in Sodium

Sample Date	Sample Container	Sample Size (g)	Sample Flow (gpm)	Flush Time (min)	Sample Temperature (°F)	Aliquot Size (g)	Number of Aliquots	Average Concentration (ppm)
<u>Primary Sodium</u>								
7/7/69	Extrusion vessel	15	0.5	15	560	~1	3	0.8 ± 0.2
8/12/69	Extrusion vessel	15	0.4	15	540	~1	3	1.4 ± 0.8
<u>Secondary Sodium</u>								
8/13/69	Quartz vials	~5	0.5	50 ^a	515	~1	5	1.5 ± 0.8
8/20/69	Extrusion vessel	15	0.5	10	515	~1	4	1.2 ± 0.3

^aTotal time sodium was flowing through the sampler; samples were taken over this span of time. The sampling method used here is described in Proceedings of the International Conference on Sodium Technology and Large Fast Reactor Design, ANL-7520, Part I, p. 216.

Recovery tests on the oxyacidic flux method for sodium carbonate in sodium have been completed. Five runs averaged $100.6 \pm 3.2\%$ recovery. Samples were prepared by spiking small quartz vials with sodium carbonate and then filling them with EBR-II secondary sodium by the method described in ANL-7520, Part I.**

*Miles, C. C., An Oxyacidic Flux Method for Determination of Carbon in Sodium, Anal. Chem. **41**(8), 1041-1047 (July 1969).

**Proceedings of the International Conference on Sodium Technology and Large Fast Reactor Design, Part I: Sessions on Sodium Technology, ANL-7520, p. 216.

(v) Hydrogen in Sodium. Table I.E.8 lists the results of analyses for hydrogen in sodium by the isotope dilution method.

TABLE I.E.8. Hydrogen in Sodium

Sample Date	Concentration ^a (ppm)
<u>Primary Sodium</u>	
7/21/69	2.3
	1.1
<u>Secondary Sodium</u>	
7/15/69	1.7
	1.7

^aDuplicate analyses.

(vi) Test of UNC Oxygen Meter.

Two United Nuclear (UNC) oxygen-meter probes have been tested in static sodium at ~600°F. One probe (Serial No. 201) was located in the primary-sodium cold-trap room where the ²⁴Na activity reaches ~10⁴ R/hr during extended reactor operation at full power. The other probe (Serial No. 146) was located in a nonradioactive area. The test was terminated September 8.

The γ -test probe (No. 201) had been operated for 164 days with the dose-rate ranging from 10³ to 10⁴ R/hr. There was no significant change in output signal other than that due to variations in temperatures of the probe and the upper hardware. Typical test data are listed in Table I.E.9. Two temperature coefficients for the probe were noted. One, associated with the electrolyte temperature, was positive and ranged from 1 to 1½ mV/°F. The other, associated with the temperature of the upper hardware, was negative and ranged from 1/2 to 1 mV/°F, based on the wall temperature just below the Conoseal flange.

TABLE I.E.9. Results of γ -test of UNC Oxygen Meters

Days under Test	Probe No. 201 (γ environment)			Probe No. 146 (non- γ environment)		
	Probe Output (V)	Probe Temperature (°F)	Upper-hardware Temperature (°F)	Probe Output (V)	Probe Temperature (°F)	Upper-hardware Temperature (°F)
1	1.095	600	160	1.119	600	184
21	1.085	600	175	1.124	604	195
48	1.086	602	182	1.135	605	194
90	1.090	605	182	1.143	609	193
116	1.091	602	178	1.146	607	192
164	1.091	601	181	1.186	601	187

The output signal of the probe in the nonradioactive area (No. 146) showed an upward drift during this same interval. Temperature coefficients were approximately the same for this probe as for the γ -test probe. Probe No. 146 had previously been operated in the γ environment for 30 days, during which time the dose rate exceeded 10³ R/hr for 16 days, without exhibiting significant drift of signal.

(vii) Hydrogen and Nitrogen in Cover Gas. Table I.E.10 summarizes the concentrations of hydrogen and nitrogen in the primary and secondary argon cover-gas systems. Measurements are made with on-stream gas chromatographs.

TABLE I.E.10. Hydrogen and Nitrogen in Primary and Secondary Argon Cover-gas Systems (ppm)

	Primary			Secondary				Primary			Secondary		
	High	Low	Avg	High	Low	Avg		High	Low	Avg	High	Low	Avg
	August							September					
H ₂	350	4	20	16	4	8	H ₂	244	4	14	8	4	6
N ₂	6500	3500	4500	2300	1400	1700	N ₂	9300	4600	7100	1400	500	1100

5. Experimental Irradiation and Testing (R. Neidner)

a. Experimental Irradiations

Last Reported: ANL-7606, pp. 42-45 (Aug 1969).

Table I.E.11 shows the status of the experimental subassemblies in EBR-II as of September 15.

TABLE I.E.11. Status of EBR-II Experimental Irradiations as of September 15, 1969

Subassembly No. and (Position)	Date Charged	Capsule Content and (Number of Capsules)	Experimenter	Accumulated Exposure (MWd)	Estimated Goal Exposure (MWd)	Burnup ^b
XG03 ^a (7D1)	7/16/65	UO ₂ -20 wt % PuO ₂ (2)	GE	24,310	28,700	6.1
XG04 ^a (7B1)	7/16/65	UO ₂ -20 wt % PuO ₂ (2)	GE	25,506	45,000	6.4
XO20 ^a (6B5)	1/13/67	UO ₂ -PuO ₂ (9)	GE	15,469	13,000 ^c	6.2
		UC-20 wt % PuC (3)	UNC			6.8
		Structural (4)	PNL			4.3
		Structural (2)	ANL			4.3
		Graphite (1)	PNL			4.3
XO21B ^d (5D2)	2/23/69	Structural (6)	PNL	14,870	23,200	2.5 + 4.1 ^e = 6.6
		Structural (1)	PNL	5,906	9,000	2.5
XO27 ^a (4B3)	11/21/67	UO ₂ -25 wt % PuO ₂ (18)	GE	12,792	16,000	7.1
		Structural (1)	PNL			5.0
XO33 ^a (5E2)	12/22/67	UC-20 wt % PuC (19)	UNC	11,939	12,200	5.9
XO35 ^d (7B2)	4/13/68	Structural (7)	ORNL	11,814	44,800	2.7
XO36 ^a (7E1)	7/25/68	UO ₂ -25 wt % PuO ₂ (19)	GE	10,190	33,300	2.8
XO38 ^d (7D1)	5/7/68	Structural (7)	INC	11,396	17,700	2.5
XO41 (7A3)	7/24/68	Structural (7)	PNL	9,856	16,700	2.1
XO43 ^a (4D2)	2/20/69	UO ₂ -25 wt % PuO ₂ (37)	GE	5,906	6,100	3.2
XO44 (7A1)	9/28/68	Oxide Insulator (1)	LASL	8,531	12,000	1.4
XO50 ^a (4C2)	2/23/69	UO ₂ -20 wt % PuO ₂ (4)	GE	5,906	7,500	3.3 + 7.7 ^e = 11.0
		UO ₂ -28 wt % PuO ₂ (4)	GE			3.2
		UO ₂ -20 wt % PuO ₂ (5)	ORNL			3.2
		UC-18 wt % PuC (2)	W			3.2
		Structural (4)	GE			2.2 + 5.3 ^e = 7.8
XO51 ^a (3A2)	12/16/68	UO ₂ -25 wt % PuO ₂ (37)	PNL	6,451	16,400	1.3
XO54 ^a (4E1)	3/31/69	UO ₂ -25 wt % PuO ₂ (37)	PNL	5,306	10,000	2.8
XO55 ^d (7D3)	2/23/69	UC-15 wt % PuC (19)	UNC	5,906	20,000	1.9
XO56 ^a (5C2)	4/2/69	UO ₂ -25 wt % PuO ₂ (37)	GE	5,306	10,600	2.6
XO57 ^d (5B2)	2/23/69	Structural (7)	PNL	5,906	15,000	2.5

TABLE I.E.11 (Contd.)

Subassembly No. and (Position)	Date Charged	Capsule Content and (Number of Capsules)	Experi- menter	Accumulated Exposure (MWd)	Estimated Goal Exposure (MWd)	Burnup ^b
XO58 ^a (6F1)	4/24/69	UO ₂ -25 wt % PuO ₂ (37)	GE	4,730	16,000	1.7
XO59 ^a (4A1)	4/23/69	UO ₂ -25 wt % PuO ₂ (37)	PNL	4,730	17,500	1.6
XO61 ^d (7B1)	4/23/69	Structural (7)	INC	4,730	18,000	1.0
XO62 ^a (6F3)	5/23/69	UO ₂ -25 wt % PuO ₂ (37)	GE	3,598	13,400	1.6
XO63 ^d (7F1)	6/29/69	Magnetic Materials (7)	ANL	2,396	5,400	0.5
XO64 ^a (4F2)	5/28/69	UO ₂ -25 wt % PuO ₂ (19)	GE	3,598	10,700	2.0

^aTemporarily removed to storage basket for Run 38A.

^bApproximate accumulated center burnup on peak rod (fuels, at. %; nonfuels, nvt x 10⁻²²).

^cSpecial extension on XO20 through Run 37.

^dTemporarily relocated for Run 38A.

^ePrevious exposure from another subassembly.

The following five experimental subassemblies were removed from the reactor for interim inspection or because they had reached their terminal exposure after Run 37:

Subassembly No.	From Position	Content of Capsule or Element	Number of Capsules or Elements	Experi- menter	Approximate Burnup* (fuels, at. %; nonfuels, nvt x 10 ⁻²²)
XO12**	4B2	UO ₂ -20 wt % PuO ₂	19	NUMEC	9.0
XO32	6E5	UO ₂ -25 wt % PuO ₂	19	PNL	4.8
XO42A**	7D5	B ₄ C	4	PNL	2.1
		Ta	3	PNL	2.1
XO60	7C3	Structural	7	INC	1.2

General Electric reported that its postirradiation neutron-radiographic inspection of the four mixed-oxide capsules from Subassembly XO10 (about 5.8 at. % average burnup on peak pin) revealed one failed Incoloy-800-clad element.

6. Materials-Coolant Compatibility (D. W. Cissel)

a. Evaluation and Surveillance of EBR-II Materials (S. Greenberg)

Last Reported: ANL-7606, pp. 46-47 (Aug 1969).

(i) Materials Exposed to EBR-II Primary Sodium. In the Progress Report for April 1968, ANL-7445, Table I.E.26, p. 88, materials

*Average burnup of lead pin.

**Requested for reconstitution.

directly exposed to liquid sodium in the primary system of EBR-II were listed. Errors have since been found in the list, some of which have been noted in subsequent progress reports. At this time, it is believed desirable to publish a revised list incorporating the results of continuing scrutiny of the list during the last 18 months. The revised list of materials is in Table I.E.12.

TABLE I.E.12 Materials in Contact with Liquid Sodium in the EBR-II Primary System

Material	Approximate Sodium Temperature (°F)
Ampco 18-13 Aluminum Bronze	700, 800-850, 900
Ampco 18-23 Aluminum Bronze	700
Borated ($1\frac{1}{2}\%$) Type 304 Stainless Steel	700
Carbon Steel, ASTM 325-55T	700
Chromium-plated Types 304, 410, and 420 Stainless Steel	700 and 900
Calmonoy 4 and 5	900
Inconel	700
Inconel X	700
Stainless Steel Types 303, 304, 316, 321, and 347	300 to ~1000
Stainless Steel Types 410 and 420	700 and 900
Steel, SAE 4340	700
Stellites No. 3 and 6B	700 and 900
Tantalum ^a	700
Tool Steel Types 18-4-1, 18-4-2, and Rex AA	700 and 900

^aTantalum is used as the cladding for the antimony-beryllium neutron sources. These sources are being replaced by sources clad with stainless steel.

(ii) In-reactor (Piggy-back) Corrosion Tests. Cylindrical samples of Type 304 stainless steel, Inconel 600, Hastelloy X, and Incoloy 800 were exposed to reactor sodium for 413 effective full-power days (278 days at 45 MWt and 135 at 50 MWt) in the core of EBR-II, using techniques that have been described.* The exposed surface of each sample was 5.7 cm². Sodium flowed past the samples at a rate of 13.2 fps, and the sodium temperature at full power was 844°F. The samples were in primary sodium at temperatures of $\geq 700^\circ\text{F}$ for 1297 days. The weight change of all samples was less than 1 mg and can probably be accounted for by weighing errors.

b. Examination of Materials from EBR-II Surveillance Subassemblies (S. Greenberg)

Last Reported: ANL-7561, p. 43 (March 1969).

The graphite cans from SURV-II have been received, and examination has started. Measurement of external dimensions showed no appreciable dimensional changes (~ 0.001 in.) or distortion.

*Greenberg, S., Cheng, C. F., and Ruther, W. E., In-reactor Sodium Corrosion of Vanadium and Vanadium-Titanium Alloys, Nucl. Appl. 4, pp. 170-172 (March 1968).

c. Effect of Interstitial Elements in EBR-II Sodium (T. D. Claar)

Last Reported: ANL-7606, p. 47 (Aug 1969).

(i) Interactions between Type 304 Stainless Steel and Nitrogen in Liquid Sodium. It has been reported (see Progress Report for July 1969, ANL-7595, p. 50) that Type 304 stainless steel foil was nitrified to a depth of 7×10^{-4} in. during a 1-week exposure to 550°C sodium under a nitrogen cover gas.

To approximate the conditions encountered in EBR-II more closely, Type 304 stainless steel foils have been exposed to 550°C sodium for 2 weeks under an Ar-0.1 wt % N₂ cover gas. Metallographic examination of the electropolished surface that was exposed to the sodium revealed massive nitrides at the grain boundaries and smaller nitride precipitates within the matrix. These microstructural features were evident without etching. Study of a transverse section of the foil exposed to the Ar-0.1 wt % N₂ mixture showed nitrides at the grain boundaries but none within the grains.

Sections of 11-mil-dia Type 304 stainless steel wire exposed along with the foils were tested for ductility after exposure. The wires could be bent around a 0.042-in. mandrel without breaking. However, the wires broke in a brittle fashion after three 180° bends.

Type 304 stainless steel foils and wires are being exposed to liquid sodium under two sets of conditions: (1) 550°C sodium under an argon cover gas and (2) 650°C sodium under an Ar-0.1 wt % N₂ cover gas.

7. Systems Engineering (B. C. Cerutti)

a. Surveillance, Evaluation and Studies of Systems

Last Reported: ANL-7606, pp. 47-48 (Aug 1969).

(i) Rotating-plug Freeze Seals (L. P. Cooper)

The large-plug seal trough was brush cleaned and replenished with new alloy. Between the time of the previous brush cleaning of the large-plug trough on May 24, 1969, and the time of the present brush cleaning, 171 round-trip transfers took place. Nine transfers occurred while the seal alloy was above its normal molten temperature of 350°F. During the cleaning operation, 555 lb of alloy were added to the large plug trough, bringing the final alloy depth to $7\frac{7}{8}$ in.

(ii) Electrical-machinery Brush Problems (R. E. Seever)

Tests of electrical brushes were completed in EBR-II electrical machinery that had previously experienced unsatisfactory brush performance. This machinery consisted of the exciter of the main turbine-generator, and the exciters and slip rings of the primary-pump motor-generator sets. The only brush of the several tested that performed satisfactorily was the National Carbon Company No. 258. This brush is an electrographitic grade manufactured to obtain the following characteristics:

- (1) resistivity of 0.007 ohm-in.;
- (2) transverse strength of 3500 psi;
- (3) scleroscope hardness of 48;
- (4) carrying capacity of 40 Amp/in.² (commutators) and 75 Amp/in.² (slip rings);
- (5) maximum surface speed of 5500 fpm;
- (6) 1.3- to 1.9-V contact drop (positive plus negative);
- (7) coefficient of friction below 0.225;
- (8) recommended pressure of $1\frac{3}{4}$ to 2 psi (commutators) and $2\frac{1}{2}$ to 4 psi (slip rings);
- (9) commutator undercutting required.

No abrasive material is used in the brush material.

The brushes are made from a blend of ungraphitized carbon, natural or artificial graphite, and lampblack with a suitable bonding material (e.g., pitch), and are baked in a gas-fired furnace at high pressure and temperature. After the brush material has been baked, it is given a graphitizing heat treatment at temperatures above 4500°F to convert the amorphous forms of carbon to a crystalline form of graphite. This grade of brush is recommended for moderately loaded electrical equipment that is operated normally but where exceptional film-forming properties are required to prevent threading or streaking.

The exceptional film-forming properties are required at the EBR-II site because of environmental conditions that are different from those assumed by the brush manufacturers. These conditions are:

- (1) five-thousand-foot elevation (12.3-psi atmospheric pressure), which reduces the available oxygen for film forming;

(2) low average humidity, which reduces the available moisture for film forming;

(3) operation at approximately 50% of full load, which reduces the available heat for film forming.

Sufficient oxygen, moisture, and heat are necessary to obtain good film formation from electrical brushes. Most industrial brushes are designed for use at sea level to 500-ft elevation, 50% to 90%+ relative humidity, and full-load operation. Since these conditions cannot be met at EBR-II, an electrical brush must be used that has good film-forming properties under the existing conditions.

The results of the tests of the National Carbon No. 258 brushes are summarized in Table I.E.13. These brushes have been selected as the ones to use in the electrical machinery listed in the table.

TABLE I.E.13 Results of Performance Tests of National Carbon No. 258 Electrical Brushes^a

Electrical Machine Used for Testing	Hours Operated	Brush Wear (in.)	Estimated Obtainable Average Brush Life (hr)
Exciter of Main Turbine-generator	1616	1 worn from original $2\frac{3}{4}$	2500 to 3000
Exciter of No. 1 Primary-pump Motor- generator Set	2976	$\frac{3}{8}$ worn from original $1\frac{1}{2}$	5000
Slip Rings of No. 1 Primary-pump Motor- generator Set	3286	$\frac{5}{16}$ worn from original 2	6000
Exciter of No. 2 Primary-pump Motor- generator Set	3022	$\frac{1}{2}$ worn from original $1\frac{1}{2}$	4500
Slip Rings of No. 2 Primary-pump Motor- generator Set	2924	$\frac{1}{2}$ worn from original 2	5000

^aOne or more repeat tests were run with comparable results.

(iii) Incoming-power-line Monitor (R. O. Haroldsen)

An incoming-power-line monitor was installed at EBR-II in July 1969. The monitor is designed to anticipate incoming-line voltage and the system demand for reactive power. If the incoming-line voltage or reactive-power demand suggests pending failure of the incoming line,

the monitor trips open the incoming-line breakers, provided that the EBR-II turbine-generator is operating in parallel with the incoming line.

Since its installation, the monitor has tripped three times in response to low incoming-line voltage. (August is the traditional month when thunderstorms in the area are likely to cause failure of the incoming line.) The first trip did not prevent reactor scram because of a separate problem in the constant-power-supply system. In the second and third trips, the EBR-II system separated from the incoming line, and scram was avoided. The two systems were reparalleled after the incoming line restabilized.

8. Leak Detection and Location--Xenon Method (R. E. Rice)

Last Reported: ANL-7548, pp. 55-57 (Jan 1969).

a. Xenon-tag-sampling System (C. W. Wilkes and P. B. Henault)

The system shown in Fig. I.E.2 has been designed for sampling the primary-tank argon cover gas for xenon tag. Components for the system are being procured. The sampling station will be outside and adjacent to the southeast side of the reactor building, in a small sheet-metal enclosure, or building, which has been fabricated. The sample will be tapped from the gas system of the fission-gas monitor (FGM), downstream of the reactor-cover-gas monitor (RCGM), in the return line to the N-1 nozzle. The xenon fission-product gas will be cold trapped by an activated-charcoal filter at the sampling station.

Fig. I.E.2. Schematic Diagram of EBR-II Xenon-tag-sampling System

The sample-gas line will be routed outside the reactor building through spare mechanical penetration No. 2. Four capped tubes as well as several spare electrical and instrumentation leads will be included in the penetration for future use. The electrical and mechanical feedthroughs are designed to allow pressure testing of the penetration. Two building-containment solenoid isolation valves, controlled from the sampling station, will be in the sampling line outside the reactor building. They will close if electrical power is lost or if a reactor-building ventilation-isolation signal is received. A key switch allows bypassing of the building-isolation signal so that a xenon sample can be taken when the building is isolated.

At the sampling station, the sampling line terminates with two valves and a fitting for connecting to the portable filters. After passing through the filters, the exhaust gas will be routed to the underground suspect exhaust line outside the reactor building and thence to the stack. A flowmeter, pressure gauge, and metering valve at the sampling station will be used for system control. An indicator for pressure of the primary-tank cover gas will be in the sampling station. This indicator will have a low-pressure alarm.

b. Improved Cover-gas-purging System (P. B. Henault and C. L. Livengood)

One of the potentially limiting factors of the xenon-tag technique for failed-fuel identification is that simultaneous or near-simultaneous failure of elements in two or more tagged subassemblies will result in a tag mixture, thereby making it difficult or impossible to identify the individual subassemblies holding the failed elements. Though the possibility of simultaneous failures in more than one subassembly is extremely remote, the possibility of a second failure soon after the first is considered credible.

To be able to identify a second failure occurring soon after the first, the capability to "clean up" the cover gas in a finite time after the first failure has been proposed to be added to EBR-II by increasing the capacity of the cover-gas-purging system. In support of this effort, the increase in gas activity resulting from the failure of the cladding of a typical experimental oxide fuel element, and the subsequent rate of activity release from the EBR-II facility once the purging system is put into operation, were calculated.

The assumption was made that all gaseous fission-product isotopes had reached equilibrium concentration before the cladding failed. This assumption is conservative and would be valid for the end of an irradiation run except for ^{85}Kr , which has a 10.6-yr half-life. The equilibrium ^{85}Kr contribution, however, is only a small part (~1.0%) of the total activity.

Table I.E.14 lists all the detectable gaseous fission-product isotopes that may be found in a sample of the EBR-II reactor cover gas, their fission yields, and the contribution of each to the total cover-gas activity at time zero. Fast fission yields were taken from a recent Idaho Nuclear Corp. study* of EBR-I fuel pins and were adjusted for a fuel composition having 80% uranium and 20% plutonium. (This composition is typical of oxide fuel being irradiated in EBR-II.) Because cover-gas samples, purge gas, etc., are cooled before being analyzed or exhausted to the stack, iodine and bromine isotopes were not considered in the total activity even though these isotopes contribute to the activity of the hot cover gas at 700°F. Also excluded from the list were the very short-lived isotopes, since these will decay to undetectable levels before they can be monitored in a gas sample or exhausted from the stack.

TABLE I.E.14. Isotopic Contributions to an Increase in EBR-II Cover-gas Activity Resulting from Failure of a Typical UO_2 - PuO_2 Fuel Element

Isotope	Fission Yield (%)	Half-life	Activity at Time Zero (Ci)	Isotope	Fission Yield (%)	Half-life	Activity at Time Zero (Ci)
$^{83}\text{Kr m}$	0.57	114 min	72	$^{131}\text{Xe m}$	0.03	12 days	4
^{85}Kr	0.29	10.6 yr	36	^{133}Xe	6.57	5.27 days	826
$^{85}\text{Kr m}$	1.04	4.4 hr	131	$^{133}\text{Xe m}$	0.16	2.3 days	20
^{87}Kr	2.46	76 min	309	^{135}Xe	4.71	9.2 hr	592
^{88}Kr	3.19	2.8 hr	401	$^{135}\text{Xe m}$	2.02	15.3 min	254
$^{129}\text{Xe ma}$	-	8 days	-	^{138}Xe	6.27	17 min	787

^aThough this isotope has been detected in EBR-II cover-gas samples, no data for its fission yield are available.

The activities listed in Table I.E.14 are actually the total inventories in the fuel at equilibrium conditions. Not all of this activity gets out of the fuel and into the plenum, and not all of the activity in the plenum is likely to escape during a failure. The theoretical upper limit for gas release from oxide fuel has been set at 80%. Measured gas release from a number of encapsulated irradiated oxide pins, however, varied from only 5 to 65%, with the average around 40%.** In addition, an oxide element is likely to fail below the bottom of the gas plenum. With such a failure early in life, only about half of the plenum gas would be released; late in life, most of the gas should be released. Based on these considerations, a more realistic value for the total activity released at time zero would be 2000 Ci, or about 60% of the total value shown in Table I.E.14. This activity, of course, would be further reduced by radioactive decay before sampling or purging operations commenced.

*Lisman, F. L., and Maeck, W. J., "Burnup Determination of Nuclear Fuels," Project Report for the Quarter April 1-June 30, 1968 and Final Report, IN-1277 (March 1969).

**Hull, G. R., and Craig, C. N., Experimental Description and Hazards Evaluation for the Group F9A Fast Ceramic Reactor Irradiation in EBR-II, General Electric-Sunnyvale (Aug. 8, 1968).

In calculating the rate of gas-activity release from the EBR-II exhaust stack after the start of purging, it was assumed that purging did not begin until 4 hr after the failure. This time interval is required to recover and identify a tag sample. The purge rate was assumed to be 3 scfm. The EBR-II cover gas has a volume of 625 ft³ and an average temperature of approximately 630°F. Table I.E.15 lists the calculated cover-gas activity, the release rate to the stack, and the total amount of activity released for various times after failure.

TABLE I.E.15. Calculated Release Rate of Cover-gas Activity, and Total Activity Release at Various Times after Failure of an Oxide Fuel Element When the Proposed Improved Purging System Is Used

Time after Failure (hr)	Cover-gas Activity (Ci)	Rate at Which Activity Is Released to Stack (Ci/min)	Total Activity Released from Stack (Ci)
0	3500	0	0
1	2900	0	0
2	3230	0	0
3	1890	0	0
4 (purging starts)	1550	15.5	0
5	780	7.8	710
6	420	4.2	995
7	225	2.3	1090
8	102	1.0	1135
9	58	0.6	1165
10	31	0.3	1180
11	17	0.2	1185
12	9	0.1	1190

9. Reactor Analysis, Testing and Methods Development

Last Reported: ANL-7606, pp. 48-61 (Aug 1969).

a. Response-time Measurements (G. S. Brunson, J. R. Karvinen, and R. A. Call)

During the week of Sept. 8, a comprehensive testing program was undertaken to measure the response times of those instrument channels in EBR-II considered to be of particular importance to the safety and control of the reactor. All measurements were made by scaling times off a chart on which was recorded the output from a Brush Clevite Model 260 6-channel recorder. Chart speed was always 125 mm/sec and was shown to be accurate to better than 0.2% in a preliminary test. In general, the time values cited below are average values for three measurements. Time did not permit all combinations of channels and tests.

(i) Nuclear Channels

(a) Startup Channels (1, 2, and 3). These channels record information derived from uranium-loaded fission chambers. The tests were carried out by disconnecting the regular fission chamber, which was left in place. The test chamber was placed in a paraffin barrel, and suitable beginning and final count rates were obtained by maneuvering two neutron sources in the barrel. The step was obtained by dropping the larger source. Time was measured from the appearance of neutron signals at the preamplifier to the tripping of the control-power (CP) relay. Since these channels feed into a two-out-of-three circuit, one other channel had to be pretripped to get a measurement. The channels are designed to trip on both period and level signals. However, only one of the step inputs was in the range for operating the level trip. The results of the test are summarized in Table I.E.16.

TABLE I.E.16. Response Times of Channels 1, 2, and 3^a

Range of Test Step (counts/sec)	Time to Period Trip (ms)			Time to Level Trip (ms)		
	Channel 1	Channel 2	Channel 3	Channel 1	Channel 2	Channel 3
100-500	2020	2000	2100	-	-	-
500-2000	1880	1980	1850	-	-	-
1000-5000	1470	1640	1460	1470	730	600
2000-9000	1610	1570	1470	-	-	-

^aAll three channels were set on trip points of 25-sec period and 1500-count/sec level. This level trip functioned only on the third step, as shown above.

(b) Intermediate Channels (4, 5, and 6). These channels register the signals from compensated ion chambers, which were disconnected

TABLE I.E.17. Response Times of Channels 4, 5, and 6^a

Range of Test Input Step (A)	Time to Operation of CP Relay (ms)		
	Channel 4	Channel 5	Channel 6
0.3×10^{-9} to 0.9×10^{-9}	310	-	-
0.3×10^{-7} to 0.9×10^{-7}	360	250	340
0.3×10^{-5} to 0.9×10^{-5}	250	250	340
0.3×10^{-4} to 0.9×10^{-4}	-	-	295
0.3×10^{-3} to 0.9×10^{-3}	260	245	-
10^{-9} to 10^{-6}	120	95	-
10^{-8} to 10^{-5}	-	-	112
10^{-7} to 10^{-4}	-	-	105
10^{-6} to 10^{-3}	100	125	-

^aAll three channels were set on a trip point corresponding to a 25-sec period.

for these tests. The test input was a square-step current signal between the levels indicated in Table I.E.17. The signal was supplied by a signal generator built for this purpose and was introduced at the connector to which the chamber cable is normally connected. These channels trip on period but not level.

(c) High-power Channels (9, 10, and 11). These channels use compensated

ion chambers and were tested in the same way as the intermediate channels. However, in this case the trip results from high level rather than rate of change. The results of these tests are given in Table I.E.18.

TABLE I.E.18. Response Times of Channels 9, 10, and 11^a

Range of Test Input (A)	Time to Operation of CP Relay (ms)			Time to Scram Clutch on No. 8 Control Rod (ms)		
	Channel 9	Channel 10	Channel 11	Channel 9	Channel 10	Channel 11
Step from 10^{-8} to 10^{-4}	32	29	32	26	23	26
Step from 10^{-6} to 10^{-4}	34	31	31	25	26	23
Step from 10^{-5} to 10^{-4}	36	31	29	26	25	24
Step from 0.3×10^{-4} to 10^{-4}	33	29	29	27	22	23

^aFor all tests, the channels were set at their normal operating trip points, as follows: Channel 9-- 0.49×10^{-4} A; Channel 10-- 0.519×10^{-4} A; and Channel 11-- 0.44×10^{-4} A.

(ii) Thermocouple Channels. These measurements were made by introducing a step-voltage input corresponding to the temperatures shown in Table I.E.19. The times indicated apply only to the electronic channels and do not include the effects of the thermocouple time constants. Thermocouples 503V, 503Z, 503M, and 503Q measure the temperatures of core subassemblies. Thermocouple 507BK monitors reactor outlet temperatures.

TABLE I.E.19. Response Times of Thermocouple Electronic Channels Exclusive of Thermocouple Time Constants^a

Simulated Temperature Step Referred to Normal Operating Trip Point (°F)	Time to Trip (ms)				
	TC 503V	TC 503Z	TC 503M	TC 503Q	TC 507BK
-55 to +10	2230	-	-	-	-
-55 to +5	-	2680	2690	1950	1010
-45 to +5	2740	2410	2480	785	965
-35 to +5	2600	2190	220	1670	870
-25 to +5	2170	1900	1900	1220	1100
-15 to +5	1710	1480	1380	980	920

^aNormal operating trip points are: TC 503V--905°F; TC 503Z--870°F; TC 503M--875°F; TC 503Q--861°F; and TC 507BK--862°F.

(iii) Flow-sensing Channels. These channels consist of two types: those that measure flow and those that measure the rate-of-change of flow. First, measurements were made by cutting pump power at full flow and timing the interval from power-off by CP-relay operation. The simulated loss-of-flow measurements were made by introducing into the sensor channel a signal representing a sudden 10% loss of flow, time being measured from signal to operation of the CP relay. The results of these tests are given in Table I.E.20.

TABLE I.E.20. Response Times of Flow-sensing Channels

Test	Unit	Trip Point (% of Normal Flow)	Response Time (sec)
Low Flow	Pump No. 1	-6	3.52
	Pump No. 2	-6	3.78
	Total	-6	3.75
Rate of Change	Pump No. 1	3%/sec ^a	2.3
	Pump No. 2	3%/sec ^a	2.65
Rate of Change with Simulated Loss of Flow	Pump No. 1	3%/sec	1.15
	Pump No. 2	3%/sec	1.40

^aActual flow at the time of activation of these trips was about 97%.

(iv) FERD (Fuel Element Rupture Detector). These channels are fed by a BF₃ pulse chamber embedded in a moderating volume of graphite. They were tested in place using a small (1.4×10^5 neutrons/sec) Am-Be neutron source. In each case, the source was suddenly and fully inserted. The response time was measured from the appearance of neutron pulses at the amplifier to the actuation of the CP relay. Since the three channels feed into a two-out-of-three coincidence circuit, one channel had to be pretripped to make the measurement. The results of the test are given in Table I.E.21. The low count rate in Channel 3 is the result of having only one detector on

TABLE I.E.21. Response Times of FERD Channels^a

	Channel 1	Channel 2	Channel 3
Count Rate with Source Fully Inserted, counts/sec	1350	1200	370
Trip and Scale Setting for Test	Time to Trip (ms)		
60% on 100-count/sec scale	1240	1490	2480
60% on 200-count/sec scale	-	1630	3180
60% on 500-count/sec scale	1710	2080	-
60% on 1000-count/sec scale	-	2620	-

^aTimes were measured from arrival of source neutron to trip on CP.

that channel, whereas the other channels have two in parallel. One neutron detector in Channel 3 had been temporarily replaced by a gamma chamber for special background measurements.

b. Analog Reactivity Meter (J. R. Karvinen and R. A. Call)

The precursors were recalibrated, and the results of zero-power tests show that the problem of overshoot has been corrected by this refinement. Figure I.E.3 shows the results of testing before and after the recalibration. The logic section of the meter has been fabricated, and control-rod-position indicators have been connected to this section. Burn-up is being recorded on the output of this section.

Fig. I.E.3. Comparison of Reactivity Traces from the Analog Reactivity Meter during Runs 36A and 37, before and after Neutron-precursor Calibration

As in the past, the experimental reactivity meter has been kept on-line during reactor operation, including the scram of August 14, 1969. The meter indicated that a line power transient caused a false indication of scram. No unusual reactivity change was recorded during this period.

c. Test of Shielded Twisted-pair Cables (J. R. Karvinen)

A 15-pair cable consisting of foil-shielded pairs of signal leads was installed between the north column of the reactor (which contains 12 subassembly thermocouples and 3 spares), the recorders in the control room, and the 100-point scanner. The original wiring was left intact to allow comparative testing of the original and new wiring under identical circumstances. An approximate 40-db improvement in impressed signal on the leads (from pickup in the connected condition) was noted.

The signal-lead noise was reduced to a fraction of a millivolt, and because of the in-phase condition of the impressed ground-to-line voltages, a differential amplifier should provide common-mode rejection to reduce the effective noise even further. This was not the case for the unshielded wires, where phase shifts of approximately 20° were recorded between the voltages impressed on the two leads. Where the voltages were not identical in amplitude, the cause could be attributed to the connection internal to the reactor. This is proved by the symmetry that was present in the open-circuit tests. When open circuited at the reactor end, as in a fuel-handling mode, the noise level was reduced from damaging potentials (up to 70 V) to an inconsequential 50 mV, which represents at least a 60-db reduction.

d. Analysis of Subassembly Bowing (D. Mohr)

The present version of the EBR-II bowing computer code BOW-V is operating successfully. During the past month an addition was made to the program so that it now calculates simplified "hysteresis effects" when necessary. For instance, if the core goes from low power to high power and returns to low power, the subassemblies go from a condition of noninteraction to extensive interaction to essentially noninteraction. As the temperatures of the hexagonal cans are reduced and return to zero, however, the subassemblies do not necessarily return to their initially assumed, zero-lean positions. Instead, they are left somewhat askew at zero power because no restoring forces are assumed present to cause the subassemblies to become perfectly vertical. BOW-V is the first ANL bowing code to treat this hysteresis effect.

e. Temperature Distribution in Mechanically Mixed Oxide Fuel
(R. K. Lo)

A transient-heat-transfer computer code (THTB) was used to determine the temperature distribution in heterogeneous UO_2 - PuO_2 fuel for the case of a power burst following an accidental insertion of the safety rods into the just-critical EBR-II core. The reactor was assumed to be tripped on a 12.8-sec period. The fuel rod considered was 0.25 in. in diameter, the cladding was 0.015 in. thick, the PuO_2 particles were $20\ \mu$ in size, and the UO_2 particles $500\ \mu$. The contact conductance between PuO_2 and UO_2 particles was taken as $4000\ \text{Btu/hr-ft}^2\text{-}^\circ\text{F}$ and that between UO_2 particles and the cladding as $1500\ \text{Btu/hr-ft}^2\text{-}^\circ\text{F}$. The heat-transfer coefficient between flowing sodium and cladding was $1400\ \text{Btu/hr-ft}^2\text{-}^\circ\text{F}$, and the linear heat generation was $16\ \text{kW/ft}$.

Fig. I.E.4. Calculated Normalized Power Trace for Power Burst Following a Hypothetical Accidental Insertion of Safety Rods into a Just-critical EBR-II Core

Assuming that the fuel was located at Row 4 and that all the uranium was ^{238}U , the relative power density in the plutonium was calculated to be 79.9% and that in the uranium to be 20.1%. (The volumetric heat generation of PuO_2 particles was $5.803 \times 10^8\ \text{Btu/hr-ft}^3$, and that of UO_2 particles was $1.461 \times 10^8\ \text{Btu/hr-ft}^3$.) Using the normalized power trace shown in Fig. I.E.4 (as computed from the AIROS II-A code) and physical data outlined above as input to the THTB code, the temperature distribution in the PuO_2 and UO_2 particles was obtained. The calculated volume-weighted

average temperatures are shown in Fig. I.E.5. It is observed that for this case the volume-weighted temperature of PuO_2 is almost exactly the same as that of the UO_2 .

Fig. I.E.5

Calculated Transient Average Temperature of PuO_2 and UO_2 for Power Burst Following a Hypothetical Accidental Insertion of Safety Rods into a Just-critical EBR-II Core

f. Sensitivities of Neutron Fluxes and Currents in Structural Subassemblies in the Energy Range of Resonance Scattering
(D. Meneghetti and K. E. Phillips)

Stainless steel-rich structural subassemblies containing neither fissile nor fertile materials are present in some positions in the EBR-II core. Studies of spectral details and computational techniques for structural subassemblies are underway. In the August 1969 Progress Report (ANL-7606, p. 60), the calculated fine-energy-detailed variations of neutron-flux spectra in the energy range of the resonance-scattering interactions were reported for the idealized situation wherein the structural subassembly is assumed to be at the central subassembly position in an EBR-II type core. Both a possible nickel-rich structural subassembly and the current stainless steel-rich structural subassembly were studied.

Figure I.E.6 shows the fine-energy-detailed variations of the corresponding currents at various positions.

By use of the fine-flux and corresponding fine-current solutions as weighting factors, coarse-group cross-section sets have been derived for: (a) each mesh interval, i.e., 63 sets, 15 of which are in the structural region, and (b) spatial averages in various subregions, i.e., 6 sets, one of which constitutes the complete structural-subassembly region.

Fig. I.E.6. Neutron-current Spectrum at Various Spatial Positions in an EBR-II Core with a Nickel-rich and a Stainless Steel-rich Central Region

Comparisons of coarse-group calculational results using sets (a) and (b) relative to the reference calculation can be helpful in understanding the resonance-scattering effects across the interface of a core subassembly and a structural subassembly. These results for the cores of the stainless steel-rich and the nickel-rich subassemblies are shown in Tables I.E.22 and I.E.23, respectively. Analogous comparisons of coarse-group fluxes by these approximations usually agree within 1% relative to the summation of reference fine-group flux encompassed within the corresponding coarse group.

TABLE I.E.22. Relative Values of Currents at Interface of Stainless Steel-rich Structural Subassembly and Core Subassembly^a

Coarse-energy Group J	$E_{J, \text{Lower}}$ (MeV)	Reference Fine-group Sums	Coarse-group Sums	
			63 Sets	6 Sets
4	0.825-1.35	-0.000686	-0.000686	-0.000686
5	0.5	0.000498	0.000497	0.000499
6	0.3	0.000744	0.000744	0.000742
7	0.18	0.000240	0.000261	0.000259
8	0.11	0.000566	0.000582	0.000581
9	0.067	0.000350	0.000349	0.000349
10	0.0407	0.000230	0.000217	0.000223
11	0.025	0.000154	0.000151	0.000145
12	0.015	0.000161	0.000159	0.000160
13	0.0091	0.000050	0.000049	0.000049
14	0.0055	0.000012	0.000012	0.000012
15	0.0021	0.000018	0.000018	0.000018

^aBased on a total fission source of unity per centimeter of height in cylindrical geometry.

TABLE I.E.23. Relative Values of Currents at Interface of Nickel-rich Structural Subassembly and Core Subassembly^a

Coarse-energy Group J	$E_{J, \text{Lower}}$ (MeV)	Reference Fine-group Sums	Coarse-group Sums	
			63 Sets	6 Sets
4	0.825-1.35	-0.000414	-0.000414	-0.000414
5	0.5	-0.000142	-0.000132	-0.000129
6	0.3	0.000006	-0.000001	0.000032
7	0.18	0.000926	0.000942	0.000914
8	0.11	0.000566	0.000600	0.000599
9	0.067	0.000213	0.000207	0.000204
10	0.0407	0.000787	0.000802	0.000808
11	0.025	0.000152	0.000132	0.000131
12	0.015	0.000020	0.000015	0.000007
13	0.0091	0.000272	0.000263	0.000264
14	0.0055	0.000136	0.000121	0.000137
15	0.0021	0.000051	0.000050	0.000051

^aBased on a total fission source of unity per centimeter of height in cylindrical geometry.

g. Intercomparison of Some Cross-section Sets for ZPR-3 Assembly 5 (B. R. Sehgal)

This work is related to that reported for the ZPR-3 Assembly 6F in the April-May 1969 Progress Report (ANL-7577, pp. 64-68). The core of ZPR-3 Assembly 5 had approximately the same composition as that of As-

sembly 6F, but unlike Assembly 6F had a lower-density blanket sandwiched between the core and the depleted-uranium blanket and was of cylindrical geometry. A homogeneous spherical model for the measured configuration was derived using the shape factor, the heterogeneity correction, and the center-gap corrections of the observed critical mass.* This model is shown in Table I.E.24.

TABLE I.E.24. Description of Model of ZPR-3 Assembly 5

Geometry: Sphere
Radius of Region 1 (core): 24.21 cm
Radius of Region 2 (light blanket): 42.31 cm
Radius of Region 3 (depleted-U blanket): 60.41 cm

Isotope	Region Compositions (atoms/barn-cm)		
	Region 1	Region 2	Region 3
²³⁵ U	0.00673	0.0000384	0.0000913
²³⁸ U	0.0076474	0.0190949	0.0400656
Al	0.0189787	0.0148215	0.0013676
Fe	0.0075029	0.0044956	0.0044956
Cr	0.0019763	0.0011842	0.0011842
Ni	0.0009393	0.0005628	0.0005628

(3) the Argonne set No. 238.[†] Some details of the procedure of generating these multigroup sets and the differences in the cross-section data between them were reported in ANL-7577.

The calculation of the physics parameters was performed with the SNARG-1D code^{††} in the S_8 approximation. The results of calculations

*Davey, W. G., k Calculations for 22 ZPR-3 Fast Reactor Assemblies Using ANL Cross-section Set 635, ANL-6570 (May 1962).

**Pitterle, T. A., Page, E. M., and Yamamoto, M., Analysis of Sodium Reactivity Measurements: Volume I: Cross Section Evaluation and Testing, APDA-216 (1968).

[†]Madell, J. T., A New Multigroup Cross Section Set for EBR-II Analysis, memo to P. Persiani, January 25, 1968.

^{††}Duffy, G. J., et al., SNARG-1D, A One-dimensional Discrete Ordinate Transport Theory Program for CDC-3600, ANL-7221 (1968).

TABLE I.E.25. Results of Reactivity Calculations for ZPR-3 Assembly 5

Cross-section Set	k_{eff}	% Difference from Experiment
ENDF/B	1.0106	1.06
238	0.9968	-0.32
MENDF/B	1.0102	1.02

for k_{eff} and for the fission and capture ratios are shown in Tables I.E.25 and I.E.26, respectively. The ENDF/B and the MENDF/B sets are overreactive by $\approx 1\%$ in k_{eff} , and the 238 set is slightly underreactive. There is a reasonable agreement with the measurements of fission and capture ratios calculated with the three sets except for the ^{234}U fission/ ^{235}U fission ratio, where the difference is $\approx 18\%$. It is quite possible that the ^{234}U fission cross section in each of the three sets is incorrect.

TABLE I.E.26. Calculated Fission and Capture Ratios for ZPR-3 Assembly 5

Cross-section Set	^{238}U Fission/ ^{235}U Fission	% Difference from Experiment	^{238}U Capture/ ^{235}U Fission	% Difference from Experiment	^{239}Pu Fission/ ^{235}U Fission	% Difference from Experiment	^{234}U Fission/ ^{235}U Fission	% Difference from Experiment
ENDF/B	0.0719	-3.48	0.1087	+1.87	1.1949	-3.64	0.4617	-17.55
238	0.0720	-3.36	0.1027	-3.75	1.2050	-2.82	0.4548	-18.78
MENDF/B	0.0752	+0.94	0.1029	-3.56	1.2214	-1.50	0.4610	-17.68

The relative ^{235}U and ^{238}U radial fission rates are shown and compared with the measurements in Fig. I.E.7. There is a reasonable agreement between the measured and calculated values. The discrepancy between the measured and the calculated (from the ENDF/B set) ^{238}U radial fission distribution is not as large as that for Assembly 6F.

Fig. I.E.7. Relative ^{235}U and ^{238}U Radial Fission Rates for ZPR-3 Assembly 5

Figure I.E.8 compares the radial ^{235}U fission distribution calculated by Set 238 and MENDF/B to those calculated by ENDF/B. Figure I.E.9 makes the same comparison for the radial ^{238}U fission distribution.

Again, it is seen that the 238 set gives a large difference for the ^{238}U fission distribution in the blanket. This is a consequence of the lower ^{238}U inelastic-scattering cross sections in the 238 set.

Fig. I.E.8. ^{235}U Fission-rate Distribution for ZPR-3 Assembly 5

Fig. I.E.9. ^{238}U Fission-rate Distribution for ZPR-3 Assembly 5

The general conclusions from this set of calculations are the same as reported in ANL-7577.

h. Flux Asymmetry in EBR-II (L. B. Miller and R. E. Jarka)

The fact that the asymmetric loading of experimental subassemblies in EBR-II results in an asymmetric flux distribution has been recognized for some time. The capability of the DOT5 code on the IBM 360/75 to model the entire reactor lattice in x-y geometry has been utilized to investigate the degree of flux asymmetry to be expected in a typical loading.

The horizontal midplane of the reactor as loaded for Run 32D was represented in detail, with each of the 271 subassemblies in Rows 1 through 10 divided into four volume elements. The rest of the blanket was represented with approximately one mesh point per subassembly. The hexagonal subassemblies were represented as rectangles, as shown in Fig. I.E.10, to facilitate the calculation with rectangular coordinates.

Fig. I.E.10. Mathematical Model for Run 32D

The relative power along two directions through the center of the reactor is shown in Figs. I.E.11 and I.E.12. Figure I.E.11 is a plot of the power distribution along the diameter from position 6E4 through position 1A1 to position 6B3. The driver in position 3B2 produces 4.5% more power than the driver in position 3E2. Control-rod No. 10, in position 5B3, produces 4% more power than control-rod No. 4, in position 5E3.

Figure I.E.12 shows the power distribution along the diameter from position 5A1 through position 1A1 to position 5D1. The asymmetry in

this direction is limited to local effects. Although the power in the No. 1 safety rod, in position 3D1, is 1.8% higher than the power in the No. 2 safety rod, in position 3A1, the power in the adjacent driver fuel elements in 4D1 and 4A1 varies in the opposite direction. The power in 4D1 is 2.8% less than the power in 4A1.

Fig. I.E.11. Power along X Coordinate 21

Fig. I.E.12. Power along Y Coordinate 21

i. Safety Studies

The MELT-I code has been received from BNWL and made operational on the IBM 360/75 computer. Input for the first problem is being prepared. The core under consideration is the wet-critical configuration containing 57 driver and 14 control and safety subassemblies. Preliminary one-dimensional radial-geometry diffusion calculations were made with the MACH-1 code, with compositions homogenized row-wise. The BAILIFF subroutine was used to compute $\beta_{\text{eff}} = 0.00717$ and $\ell_p = 0.793 \times 10^{-7}$ sec.

10. Driver Fuel Development (C. M. Walter)

a. High Burnup Encapsulated Irradiations (W. N. Beck and R. V. Strain)

Last Reported: ANL-7606, p. 62 (Aug 1969).

(i) Postirradiation Examination of Encapsulated Mark-II Elements Irradiated to ~4 at. % Burnup. Diametral measurement of six Mark-II elements irradiated to ~4 at. % burnup shows a diameter increase of 0.5 to 0.6%, which is consistent with the amount of stainless steel swelling expected at these fluences. Figures I.E.13 and I.E.14 are representative diameter profiles of elements having two different fuel enrichments, i.e., 93 and 80%, respectively.

Fig. I.E.13. Diameter Profile of Mark-II Element 213 (93% enrichment; ~4.3 at. % peak burnup)

Fig. I.E.14. Diameter Profile of Mark-II Element 279 (80% enrichment; 4 at. % peak burnup)

The gamma scans of the fueled portion of the elements showed flux peaking at 5 to 5.5 in. from the bottom of the pin. The gamma-scan profiles are similar to those obtained for elements gamma scanned after a burnup of 2 at. %.

The weight measurement of the deencapsulated elements showed that the weight of fuel element No. 279 had increased 0.9 g. Optical examination of the surface of the cladding of that element revealed a circumferential crack at an indentation in the cladding which holds the ferrule restrainer in position. The crack, which is next to the end of the ferrule that is away from the fuel column, was most probably a direct result of crimping the cladding against the sharp edge of the ferrule. This type of restrainer is one of three initially selected to be evaluated in these series of irradiations. It is not used in the six unencapsulated Mark-II driver subassemblies being irradiated in EBR-II. In those subassemblies, a

triple-indent type of restrainer is employed. The fuel is restrained by three indentations in the cladding; no restraining ferrule is used. The triple-indent design has shown no tendency to crack.

b. Fuel and Cladding Surveillance (J. P. Bacca)

Last Reported: ANL-7606, p. 62 (Aug 1969).

(i) 70%-enriched Fuel Experiment (J. P. Bacca and G. C. McClellan)

Postirradiation examinations of the elements from Subassembly C-2184S, which had attained a peak burnup of 1.71 at. % by the end of Run 36, were conducted. Fuel-swelling results are summarized in Table I.E.27 and depicted in the swelling-versus-burnup curve of Fig. I.E.15. As observed for earlier irradiations of 70%-enriched fuel, irradiation swelling of the 70%-enriched fuel is somewhat higher than that observed for normal 52%-enriched fuel following comparable burnup. This increase is attributed to the higher irradiation temperatures experienced in the 70%-enriched fuel.

Precision measurements of the diameter of selected elements from Subassembly C-2184S did not reveal definite straining of the cladding.

TABLE I.E.27. Fabrication and Irradiation Data for
70%-enriched Driver-fuel Subassembly C-2184S

(Calculated burnup: 1.71 at. % max; 1.45 at. % avg)

Injection-casting Batch No.	Silicon Content of Fuel (ppm)	Number of Elements	Element Burnup Range (at. %)	Total Volume Swelling of Fuel ($\Delta V/V$, %)	
				Average	Range
<u>70% Enriched</u>					
0700	260	1	1.66	17.5	-
0701	533	2	1.59-1.68	13.4	10.2-16.6
0702	362	1	1.63	16.7	-
0703	407	10	1.57-1.71	16.2	14.1-17.1
0704	409	10	1.57-1.69	14.6	10.1-16.9
0705	437	6	1.59-1.69	12.8	10.0-16.5
0706	520	25	1.56-1.71	12.2	11.1-15.9
0708	520	25	1.56-1.71	<u>10.2</u>	7.2-12.1
				Average	12.6
<u>52% Enriched</u>					
4329	449	1	1.32	5.4	-
4335	321	7	1.29-1.42	8.1	5.7-10.6

Fig. I.E.15. Swelling of 70%-enriched Fuel and of Fuel from Controlled-flow Subassemblies as a Function of Burnup

(ii) Controlled-flow Experiment (J. P. Bacca and G. C. McClellan)

Postirradiation examinations of the elements from controlled-flow subassembly C-2187S, which had attained a calculated maximum burnup of 1.5 at. % by the end of Run 36A, were conducted. Fuel-swelling results are summarized in Table I.E.28 and illustrated in Fig. I.E.15. Irradiation

TABLE I.E.28. Fabrication and Irradiation Data for Controlled-flow Driver-fuel Subassembly C-2187S

(Calculated burnup: 1.50 at. % max; 1.32 at. % avg)

Injection-casting Batch No.	Silicon Content of Fuel (ppm)	Number of Elements	Element Burnup Range (at. %)	Total Volume Swelling of Fuel ($\Delta V/V$, %)	
				Average	Range
Mark IA					
095 II	290	8	1.43-1.49	14.8	13.1-16.6
096 II	308	10	1.43-1.50	14.5	11.8-15.9
097 II	276	10	1.45-1.49	15.6	13.7-17.1
098 II	289	16	1.43-1.50	14.9	7.8-17.1
Mark IB					
095 II	290	8	1.43-1.49	17.0	14.3-18.7
096 II	308	10	1.43-1.49	16.1	12.0-20.6
097 II	276	9	1.44-1.50	17.0	13.4-20.7
098 II	289	17	1.43-1.50	17.7	13.7-21.0

swelling of elements in this controlled-flow subassembly is somewhat greater than that observed for 70%-enriched elements irradiated to the same burnup level. This tendency was observed in earlier controlled-flow subassemblies C-2185S and C-2186S (see Fig. I.E.15). Although the irradiation temperatures of the fuel in the two experiments should have been similar, they could have been higher in the controlled-flow elements. These higher temperatures would have caused higher swelling in those elements. From Table I.E.28, it can be observed that irradiation swelling of the U-5 wt % Fs fuel in the Mark-IB elements is greater than that in the Mark-IA elements. This increased swelling is attributed to the greater gas-plenum volume of the Mark-IB element. Because of this larger volume, less pressure develops in the plenum, and, therefore, there is less pneumatic resistance to fuel swelling as irradiation of the element proceeds.

Precision measurements of the diameters of elements from Subassembly C-2187S did not reveal any definite straining of the cladding.

(iii) Driver-fuel Extended-burnup Program (J. P. Bacca and G. C. McClellan)

Postirradiation examinations of Subassemblies B-397 and B-3037, both of which had attained a calculated maximum burnup of 1.74 at. % during Run 36, were conducted. The results of these examinations are summarized in Tables I.E.29 and I.E.30. Fuel swelling was as expected.

TABLE I.E.29. Fabrication and Irradiation Data for
Extended-burnup Driver-fuel Subassembly B-397

(Calculated burnup: 1.74 at. % max; 1.36 at. % avg)

Injection-casting Batch No.	Silicon Content of Fuel (ppm)	Number of Elements	Element Burnup Range (at. %)	Total Volume Swelling of Fuel ($\Delta V/V$, %)	
				Average	Range
035 HS	313	57	1.42-1.74	9.84	5.4-17.0
048 IH	335	34	1.39-1.64	8.41	5.3-13.9

TABLE I.E.30. Fabrication and Irradiation Data for
Extended-burnup Driver-fuel Subassembly B-3037

(Calculated burnup: 1.74 at. % max; 1.38 at. % avg)

Injection-casting Batch No.	Silicon Content of Fuel (ppm)	Number of Elements	Element Burnup Range (at. %)	Total Volume Swelling of Fuel ($\Delta V/V$, %)	
				Average	Range
071 IH	330	1	1.62	13.8	-
072 IH	410	30	1.51-1.64	6.4	5.7-7.1
073 H	330	59	1.37-1.64	7.1	6.1-8.1
Average for all elements				6.9	

Subassembly C-2169 had attained a calculated peak burnup of 1.88 at. % by the end of Run 36. Postirradiation-examination results for this subassembly are summarized in Table I.E.31. Precision measurements of the diameters of selected elements indicated no positive evidence of jacket straining. Burnup of this subassembly, a "leader" in the extended-burnup program, was the highest attained by a normal driver subassembly up to the time the postirradiation examinations were made. By the end of Run 37, a new "leader" subassembly (B-3038) had attained a calculated maximum burnup of 1.95 at. %. Postirradiation examinations of elements from this subassembly will begin shortly. Also by the end of Run 37, seven additional subassemblies had attained calculated peak burnups in the range from 1.7 to 1.8 at. %.

TABLE I.E.31. Fabrication and Irradiation Data for
Extended-burnup Driver-fuel Subassembly C-2169

(Calculated burnup: 1.88 at. % max; 1.60 at. % avg)

Injection-casting Batch No.	Silicon Content of Fuel (ppm)	Number of Elements	Element Burnup Range (at. %)	Total Volume Swelling of Fuel ($\Delta V/V$, %)	
				Average	Range
060 II	330	9	1.82-1.84	16.9	15.6-18.2
062 IH	437	9	1.60-1.85	16.1	15.3-17.1
063 IH	369	70	1.76-1.83	16.4	14.0-17.8

(iv) Mark-II Driver Subassemblies (J. P. Bacca and
G. C. McClellan)

Mark-II driver subassemblies C-2201S, C-2203S, and C-2205S had attained calculated peak burnups of approximately 0.95 at. % by the end of Run 37. The subassemblies were transferred to the reactor storage basket and will reside there during the 62.5-MWt operation of Run 38.

(v) Metallographic Studies of Cladding from Driver-fuel
Extended-burnup Subassemblies and Controlled-flow
Subassemblies (R. V. Strain and J. P. Bacca)

Samples from driver-fuel extended-burnup subassemblies C-2147 and C-2065 and controlled-flow subassemblies C-2185S and C-2186S have been examined for changes in the microstructure of the cladding. No interaction between the fuel pins and the cladding was found in the samples examined; however, only the top end of the fuel pin was in contact with the cladding because the fuel had not swelled completely to the cladding. The only change that occurred in the microstructure of the cladding was sensitization in the samples from the midlength and from

near the top of the fuel pins. The degree of sensitization did not appear to be more severe than that observed in cladding from standard driver-fuel subassemblies irradiated to a burnup of 1.2 at. %.

Both of the extended-burnup subassemblies had been irradiated to a maximum burnup of 1.47 at. %. The controlled-flow subassemblies had been irradiated to a burnup of 1.2 at. % under sodium-flow conditions that allowed maximum fuel temperatures to reach 1170°F, as compared to maximum temperatures of 1039°F for similar subassemblies during 50-MWt operation under normal flow conditions.

Micrographs of the cladding samples in the as-polished and as-etched conditions revealed no evidence of grain-boundary corrosion, reduction in wall thickness, or other defects that could be attributed to residence in the reactor.

Cladding sensitization was more severe at the midlengths than at the tops of the fuel pins. Although this phenomenon has been noted previously in normal driver-fuel cladding, it was very pronounced in these elements. Calculations indicate that the temperatures at the midlengths are significantly lower than at the tops of the pins. The midlengths of the fuel pins experience greater neutron exposure than the ends of the pins. This indicates that the sensitization of the cladding may be a function of neutron exposure as well as temperature. The increased sensitization in the high-flux region of the elements may be indicative of enhanced diffusion as a result of irradiation. The enhanced diffusion may deplete chromium near the grain boundaries and accelerate corrosion at the grain boundaries. Based on these results, grain-boundary corrosion would be expected to be most severe in the region of highest fluence rather than in the regions of highest temperature near the tops of the fuel pins. This phenomenon of enhanced diffusion caused by neutron exposure also may be a factor in the reduction in cladding ductility with increased irradiation.

11. Operation with Failed Fuel (R. R. Smith)

Last Reported: ANL-7606, pp. 62-71 (Aug 1969).

a. Fission-product Release of May 13, 1969 (R. M. Fryer)

Following a routine shutdown at the end of EBR-II Run 34A on May 13, 1969, it was noted that the ^{135}Xe activity in the cover-gas plenum was decaying at a rate slower than that expected from half-life considerations. Accordingly, experimentally measured values were compared with those expected from a simple decay of the ^{135}I - ^{135}Xe fission-product pair. The results of the comparison are summarized in Fig. I.E.16. Subtraction of calculated values from experimental values showed a time-dependent buildup of an additional ^{135}Xe component after reactor and primary-pump shutdown. Since no evidence was noted of the step increase in ^{135}Xe activity that is typical of a gas release, it was concluded at that time that the increase in ^{135}Xe activity was the result of a continuous extrusion of bond sodium containing ^{135}I , which subsequently decayed to its daughter, ^{135}Xe .

Fig. I.E.16. ^{135}Xe Activity in the Cover-gas Plenum during the May 13, 1969 Fission-product Release

Qualitatively, the release closely resembled those associated with a bond-type defect in a driver fuel element in Subassembly C-2008

(see Progress Report for April-May 1969, ANL-7577, pp. 99-101). However, since C-2008 had been removed from the core after the Run-33B shutdown on April 22, 1969, it was evident that another subassembly was involved.

At the time of the release, the release was tentatively concluded to be associated with a driver element, because the signal was small and because the nature of the release closely resembled the nature of the release from driver Subassembly C-2008.

Attempts were made to narrow the list of suspects from a knowledge of the signal strength. In principle, the magnitude of the signal strength associated with a bond-type defect depends on the size of the defect (i.e., sodium-extrusion rate), the specific power-generation rate, and the change in coolant pressure following pump shutdown. Of these, the latter two depend strongly on row position in the core. Also affecting the signal strength is the duration of the time period in which sodium is permitted to extrude during pump shutdown. Obviously, a long pump-shutdown period will lead to a larger signal increase than a shorter period.

Assuming that the defect was comparable in size to that associated with the C-2008 leaker, the signal strengths noted from specific C-2008 releases were used as bases of comparison. To reduce the effects of power-shutdown schedule and pump downtime on the comparison, two relatively unambiguous C-2008 releases were used as criteria. For both of these releases, the shutdowns were the results of reactor scrams, and the primary pumps were turned off for periods of approximately 10 hr. With C-2008 in Row 2, a maximum saturated signal strength of $3.3 \times 10^{-3} \mu\text{Ci/ml}$ was noted. In Row 3, the corresponding signal strength was $2.5 \times 10^{-3} \mu\text{Ci/ml}$.

By taking the ratio of the relative fluxes and pressure differentials occurring after pump shutdown for the various row locations and using the Row 2 and 3 signal strengths defined above, values for the maximum saturated signal strength for other core locations were estimated. The results are summarized in Table I.E.32.

TABLE I.E.32. Maximum Releases of ^{135}Xe from Single Mark-IA Fuel Elements

Reactor Position	Peak Release ($\mu\text{Ci/ml}$ of ^{135}Xe)
Row 2	3.3×10^{-3a}
Row 3	2.5×10^{-3a}
Row 4	1.5×10^{-3b}
Row 5	1.1×10^{-3b}
Row 6	1.0×10^{-3b}
Row-5 Control Rod	6.0×10^{-4b}

^aMeasured.

^bEstimated.

An analysis was made of the signal strengths associated with the leaker for all shutdowns after May 13. Post-shutdown releases were noted for Runs 34B, 35A, 35B, 36A, and 36B. In most of these instances, however, the primary coolant pumps were not turned off sufficiently long to establish the maximum signal

strengths. On one occasion--that following a shutdown on June 23 during Run 35B--reactor- and pump-shutdown conditions were such that a clear

Fig. I.E.17. ^{135}Xe Activity in the Cover-gas Plenum during the Post-shutdown Fission-product Release of June 23, 1969

maximum signal could be established. A graphical summary of measured and calculated ^{135}Xe activities during this shutdown is given in Fig. I.E.17. By comparing the indicated peak value ($1.06 \times 10^{-3} \mu\text{Ci/ml}$) with the values given in Table I.E.32, a decision was reached that a Row-5 subassembly was the most likely suspect.

By the end of Run 36B, the suspect list had been reduced to that given in Table I.E.33. Subassembly C-2078 was considered to be the most-suspect assembly. Since it was removed from the core at the end of Run 36, the tentative conclusion was reached that Run 37 might be free of bond leakers. A power and pump shutdown on August 14, 1969, during Run 37 confirmed this conclusion. As a result, the suspect list was further reduced to C-2078, C-2081, and C-2079, with C-2078 still the prime suspect. All other subassemblies listed in Table I.E.33

TABLE I.E.33. Suspect List of Subassemblies as of the End of Run 36B

Subassembly No.	Core Position during Releases	Position for Run 37	Release Correlation	Release-magnitude Correlation
C-2078	5A4	Basket	Yes, every release	Good
C-2188S	2C1	2C1	Yes, every release	Fair ^a
C-2081	4D1	Basket	Yes, every release	Fair
L-475	5C3	5C3	Yes, every release	Fair
L-476	5B3	5B3	Yes, every release	Poor
L-477	5B1	5B1	Yes, every release	Poor
L-478	5E3	5E3	Yes, every release	Poor
L-479	5F3	5F3	Yes, every release	Poor
L-480	5A1	5A1	Yes, every release	Poor
C-2084	3C2, 5E4 one run only	5E4	Yes, every release	Poor
C-2079	3E1	Basket	Yes, every release	Poor
S-617	3D1	3D1	Yes, every release	Poor

^aC-2188S had a fission rate corresponding to Row 2, but was orificed to simulate a Row-5 sodium flowrate.

are still in the reactor core. The three suspect assemblies are scheduled to be transferred to the Fuel Cycle Facility, where they will be checked for sodium-bond level.

b. Studies of Loss of Bond in EBR-II Mark-IA Fuel Elements
(R. R. Smith)

In general, defects in the cladding of EBR-II Mark-IA driver elements may be classified under two broad categories: one in which the defect occurs in the gas plenum of the element, and the other in which the defect occurs in the fuel region. For defects in the gas-plenum region, the immediate and long-range consequences are nothing more than the release of rare-gas fission products to the primary-sodium and cover-gas systems. As the result of a series of calibration experiments involving unclad fuel material in the reactor core, it is now clear that the radiological effects of such a release are trivial.* No evidence, either of an analytical or experimental nature, has ever indicated the existence of potentially hazardous effects resulting from a defect in the cladding in the vicinity of the gas plenum.

For the case of a cladding defect in the vicinity of the fuel, however, the immediate and long-term hazards implications are less obvious. Depending on the location of the defective element in the core and on the elevation of the defect, sufficient bond sodium may be forced through the defect to cause the unbonding of a significant section of fuel material (see Progress Report for January 1969, ANL-7548, pp. 39-43). Under these circumstances, fuel material could melt. What actually happens after fuel material melts in a relatively intact jacket has been simulated in flat-topped transient experiments in TREAT.** As a result of these experiments, it is now believed that the immediate consequence of fuel melting is a recasting of the melt as it contacts the cladding, with the important result that heat transfer is reinitiated.

For simplicity, the initial TREAT studies were conducted on EBR-II driver elements which were completely unbonded, i.e., the annulus between the fuel material and the cladding was filled with argon under room conditions. While the tests conducted with such specimens generated useful and important information, the simulations failed to consider the effects of residual sodium which would remain in the annulus under actual bond-loss conditions.

*Smith, R. R., et al., Exposed Fuel Calibration Studies in EBR-II, Second Series, ANL-7558 (to be published).

**Willis, F. L., Dickerman, C. E., Henault, P. B., Purviance, R., Boland, J. F., and Noland, R. A., TREAT Experiment Study of Survival of EBR-II Fuel Pin in a Loss-of-Bond Incident, Trans. Am. Nuc. Soc. 11(2), 698 (Nov 1968); Dickerman, C. E., Smith, R. R., Willis, F. L., Cohen, A. B., and Regis, J., Analysis of Transient Behavior of Unbonded EBR-II Fuel Pin Incident, *ibid.*, 11(2), 698 (Nov 1968); Dickerman, C. E., Willis, F. L., Smith, R. R., and Cohen, A. B., "Fast Reactor Safety Integral Loop Experiment on an Argon-bonded EBR-II Pin," ANL Reactor Physics Div. Annual Report, July 1, 1966-June 30, 1967, ANL-7310 (1968); Dickerman, C. E., and Robinson, L. E., "Fuel Meltdown Experiments in TREAT," Int. Conf. on the Safety of Fast Breeder Reactors, Aix-en-Provence, France (Sept 1967).

Subsequent investigations of such effects involved both analytical and experimental efforts. In the former, fuel temperatures were evaluated as a function of the amount of circumferential bond loss. Experimental efforts involved an additional flat-topped TREAT transient conducted on a fuel element which contained only enough sodium to bring the bond to mid-plane elevation. An implication of the results of these studies is that, even though a cladding defect in the fuel region may lead to the substantial uncovering of the fuel material, melting may not necessarily occur. Evidence substantiating this belief is given below.

(i) The Effect of Annular Bond Voids on Fuel Temperature
(R. A. Cushman)

The model chosen for these particular studies involved an EBR-II Mark-IA driver element assumed to be operating at the maximum specific core power corresponding to a reactor power level of 50 MWt. It also was assumed that the 0.006-in.-thick sodium bond in the annulus between the fuel pin and the cladding was replaced by argon in incremental angular changes. Parametric variations of 20° of arc were made in the argon gas gap over the range from 60 to 260°.

Two-dimensional geometry (r, θ) was used, and for simplicity, it was assumed that axial heat conduction was negligible. Five radial nodes were used in the fuel region, with radial boundaries fixed at 0.020, 0.040, 0.060, 0.070, and 0.072 in. All nodes were 10° in angle.

Fuel temperatures were computed for each node with the THTB program. As expected, the results showed that, as the circumferential gas gap increased, the maximum fuel temperature increased, with the maximum occurring near the outside of the fuel at the center of the gas gap. The results of the calculations are summarized in Fig. I.E.18, in which the maximum fuel temperature is plotted as a function of the angular width of the circumferential gas gap. To illustrate the temperature distribution in the fuel material, temperatures are also given for various distances from the vertical axis of the fuel pin. The results indicate that axial gas voids up to 260° along the full length of the pin may be tolerated before the fuel temperature at the pin surface exceeds the melting point (1840°F).

(ii) Results of Flat-topped TREAT Transient Studies on a Partially Bonded EBR-II Mark-IA Fuel Element
(R. M. Fryer)

The effects of residual sodium in the annulus of a partially voided EBR-II element were investigated experimentally in TREAT. The tests were conducted in a Mark-I flowing-sodium loop.

Fig. I.E.18. Effect of an Argon Gas Gap in the Sodium-bond Space on Fuel Temperature in an EBR-II Mark-IA Fuel Element

The fuel element used for the studies consisted essentially of a standard driver element with the ^{235}U enrichment decreased to 6%. During the assembly of the element, sufficient sodium to bond the fuel material to midplane elevation was added to the jacket, the fuel pin was settled into place, and the top closure weld was made. The element was heated to 500°C and was manually inverted to wet unbonded portions of the fuel material. This technique results in a complete bond for the lower 4½ in. of fuel material. For the remaining 9 in., the bonded area consisted of a circumferential arc of approximately 200°. Thus, the unbonded area for the upper 9 in. corresponded to an average circumferential arc of approximately 160°.

The test specimen was located at the center of a cluster of six stainless steel dummy elements in the Mark-I flowing-sodium loop and was irradiated under flat-topped transient conditions in TREAT on May 29, 1969. By programming the time dependence and amplitude of the TREAT power transient, the maximum power density in the fuel specimen was made to correspond approximately to the power density expected for standard 52%-enriched fuel material located at midplane in EBR-II under a hypothetical power level of 140 MWt.

The time-dependent power-generation rate in the specimen is given in Fig. I.E.19. Also given is the equivalent EBR-II power. A corresponding record of inlet and outlet temperatures in the loop is presented in Fig. I.E.20. The axial power distribution in the test specimen, as determined by gamma scanning, is given in Fig. I.E.21.

Fig. I.E.19. Power Generation vs Time for Partially Bonded EBR-II Mark-IA Fuel Element Irradiated in TREAT

Fig. I.E.20. Loop Inlet and Outlet Sodium Temperatures during TREAT Test of Partially Bonded EBR-II Mark-IA Fuel Element

Fig. I.E.21. Axial Power Distribution in Partially Bonded EBR-II Mark-IA Fuel Element Irradiated in TREAT

Following the transient, the specimen was transferred to the air cell of the Fuel Cycle Facility for postirradiation tests. A visual examination revealed the absence of any detectable surface effects. From a comparison of neutron radiographs made before and after the transient, it was concluded that fuel melting did not occur. The results of corresponding eddy-current bond tests made before and after the transient were consistent with this conclusion. The results of profilometry measurements indicated no change in the normal 0.174-in. outer dimension of the jacket over the full length of the element.

After performance of jacket profilometry, the jacket was stripped away and the bond quality checked. The sodium was removed and the fuel pin was measured. The average preirradiation diameter had been 0.1436 in. The postirradiation diameters (see Fig. I.E.22) suggest that slight

Fig. I.E.22. Axial Profile of Diameter of Pin in Partially Bonded EBR-II Mark-IA Fuel Element Irradiated in TREAT

phase changes might have occurred during the transient. The lower, cooler end of the fuel pin contracted slightly, whereas the upper, hotter end of the fuel element might have expanded slightly. A reduction of 0.013 in. in length was noted.

A photograph of the test pin with the cladding stripped away is shown in Fig. I.E.23. Tabular data for the extent of unbonded fuel area are summarized in Table I.E.34. From an inspection of the photograph and a review of the tabular data, it may be concluded that, although circumferential gas voids up to 240°

were present, the fuel material failed to melt. The fact that the transient tests were conducted under specific power levels exceeding those for driver fuel material in EBR-II at 50 and 62.5 MWt is encouraging. Such evidence suggests that the consequences of a loss-of-bond type defect in a driver element under normal operating conditions may be safely tolerated for at least short periods of time.

Fig. I.E.23. EBR-II Mark-IA Fuel Element That Had Been Irradiated in TREAT and Whose Cladding Was Subsequently Stripped Away

TABLE I.E.34. Condition of Bond in Partially Bonded EBR-II Fuel Element after TREAT Transient

Distance from Spade End (in.)	Quality of Sodium Bond
0	Complete circumferential bond
1	Complete circumferential bond
2	Complete circumferential bond
3	Complete circumferential bond
4	Complete circumferential bond
4.5	Complete circumferential bond
5	Unbonded over 140°/0.18 in. high on circumference
6	Unbonded over 145°/0.18 in. high on circumference
7	Unbonded over 160°/0.20 in. high on circumference
8	Unbonded over 190°/0.24 in. high on circumference
9	Unbonded over 190°/0.24 in. high on circumference
10	Unbonded over 190°/0.24 in. high on circumference
11	Unbonded over 210°/0.26 in. high on circumference
12	Unbonded over 240°/0.30 in. high on circumference
13	Unbonded over 145°/0.18 in. high on circumference
13.5	Unbonded over 160°/0.20 in. high on circumference

F. EBR-II Operations

1. Reactor Plant (G. E. Deegan)

Last Reported: ANL-7606, pp. 72-73 (Aug 1969).

The reactor was operated for 819 MWd, in Runs 37 and 38A, from August 21 through September 20. In Run 38A, the reactor was operated for the first time at a power level of 62.5 MWt. Initial operation at this level was satisfactory, and the testing program in conjunction with the power increase is proceeding. The cumulative total of EBR-II operation is 29,013 MWd.

Run 37 was completed on August 30. Fuel handling for Run 38 was extensive because of the need for relocating or removing most of the experimental subassemblies in preparation for operation at higher power. An approach to critical was made before completion of fuel handling to check the loading. After the loading was completed, the large-plug seal trough was cleaned and refilled.

Several plant changes were made during the shutdown to prepare for 62.5-MWt operation. The pitches of the cooling-tower fans were adjusted to an optimum setting. Three Annubar flow sensors were installed in the condensate-system piping, and one flow sensor was placed in the steam line to the turbine-driven condensate pump. Pressure in the secondary-sodium surge tank was increased to 7 psig to allow the increased flow rates with the existing pump. The response times of selected shutdown-circuit trips were tested.

Startups for a criticality check and control-rod calibration were conducted before the startup for the power run on September 13.

While raising the power to 50 MWt, components of the sodium boiler plant were monitored for vibration and condensate flow rates were measured. An unusually high rate of condensate flow in the desuperheater system was indicated. Rod-drop tests were performed at 50 MWt and 500 kW, after which the reactor was shut down temporarily for changing the temperature compensation of the reactor-outlet-flow indicator and for investigating the desuperheater system. Two desuperheater spray nozzles were found to have eroded and were replaced. After the reactor was restarted, desuperheater flow was normal.

With power again at 50 MWt, base data for the increase were obtained. The power was then raised slowly to 56 MWt. All systems operated normally, and rod-drop tests were conducted. After almost two days of operation at 56 MWt, the final increase to 62.5 MWt was made. There were no problems; rod drops were performed again when power had stabilized. System and component performance remain satisfactory.

Fuel handling for Run 38 included the following:*

Twenty-two driver subassemblies were replaced. These consisted of 11 spent driver subassemblies, 8 extended-burnup surveillance subassemblies, 2 vendor surveillance subassemblies, and 1 surveillance subassembly containing ANL cold-line elements bonded by the vendor. One depleted-uranium inner-blanket subassembly also was removed for surveillance.

Because of the planned 62.5-MWt operation, the three controlled-flow subassemblies, the three Mark-II subassemblies, and the subassembly containing the known upper-weld defect were removed to the basket. Fourteen experimental subassemblies were relocated in the grid. Four other experimental subassemblies that had completed their irradiation were removed from the grid.

Three fresh subassemblies containing vendor fuel elements that had been impact bonded in the cold line were installed for Run 38. One subassembly containing vendor fuel elements that had been heat treated before bonding was reconstituted from previously removed Subassembly C-2209 and was installed in the reactor for additional irradiation. One subassembly containing tantalum, niobium equilibration-test wires, silicon carbide samples, and melt wires for various test procedures was also installed in the reactor.

The antimony source was removed from the beryllium thimble and placed in a storage thimble. The source and the storage thimble, as an assembly, were removed from the reactor and transferred to the Fuel Cycle Facility.

2. Fuel Cycle Facility (M. J. Feldman)

Last Reported: ANL-7606, pp. 73-81 (Aug 1969).

a. Fuel (D. L. Mitchell)

(i) Cold-line Production and Assembly. Table I.F.1 summarizes the production activities for August 16 through September 15, 1969, and for the year to date.

Impact bonding of the unbonded fuel elements received from the vendor, Aerojet-General Corporation (AGC), is continuing. Of the 941 unbonded elements that have been impact bonded, 868 were accepted and 73 rejected. Ten unbonded elements were set aside for historical samples.

Ten Mark-IA subassemblies were fabricated during the month.

*Loading changes involving experimental subassemblies are described in Sect. I.E.5.a.

TABLE I.F.1. Production Summary for FCF Cold Line

	8/16/69 through 9/15/69		Total This Year	
	Mark IA	Mark II	Mark IA	Mark II
Alloy-preparation runs				
New fuel	3	0	10	8
Remelts	<u>2</u>	<u>0</u>	<u>12</u>	<u>3</u>
Total	5	0	22	11
Injection-casting runs	3	0	28	10
Pins processed				
Accepted	253	0	2,768	941
Rejected	3	0	89	53
Elements welded	203	0	3,143	884
Elements rewelded	0	0	0	69 ^a
Elements leaktested				
Accepted	103	0	2,725	804
Rejected	1	0	51	21
Elements bondtested				
Accepted	425	0	2,893	934
Rejected	33	0	319	9
Subassemblies fabricated (cold-line fuel)	10	0	29	6
Bonded elements received from vendor	0		24,088 ^b	
Inspected and accepted	0		23,407 ^{b,c}	
Inspected and rejected	0		1,240 ^b	
Unbonded elements received from vendor	2,179 ^d		2,179 ^d	
Impact-bonded, inspected and accepted by ANL	868		868	
Impact-bonded, inspected, and rejected by ANL	73		73	
Subassemblies fabricated (vendor fuel)	3 ^e		12 ^f	
Total elements available for subassembly fabrication as of 9/15/69				
Cold-line fuel				
Mark IA	420			
Mark II	234			
Vendor fuel (Mark IA)	22,394			

^aIncludes 64 elements in which visual examination of welds indicated that they were not acceptable for potentially high-burnup experiments.

^bTotal includes figures for 1968.

^cIncludes fuel elements returned to vendor for rework to correct voids in sodium bond and after rework sent back to ANL for reinspection.

^dTen unbonded vendor elements were set aside for historical samples.

^eEach subassembly is made up of a mixture of vendor and cold-line fuel elements.

^fIncludes subassemblies made up of a mixture of vendor and cold-line fuel elements.

(ii) Inspection of Vendor Fuel. A summary of the data relating to receipt and acceptance of the fuel produced by AGC is included in Table I.F.1.

b. Surveillance (M. J. Feldman, J. P. Bacca, and E. R. Ebersole)

(i) Postirradiation Analysis of EBR-II Fuel (J. P. Bacca)

(a) Surveillance of Vendor-produced Fuel

(1) Impact-bonded AGC Fuel Elements. Postirradiation examinations were made of the elements from Subassembly C-2200. This subassembly, which had completed irradiation to a calculated maximum burnup of 0.76 at. % by the end of Run 36, contained elements assembled by AGC but impact-bonded by ANL. In addition, the subassembly contained normal cold-line-produced elements. Tables I.F.2 and I.F.3 summarize results obtained by eddy-current bond testing of the irradiated elements. The data in Table I.F.2 indicate that the fuel pins made from

TABLE I.F.2. Postirradiation Lengths of AGC Fuel Pins from Subassembly C-2200^a
(Calculated burnup: 0.76 at. % max)

AGC Batch 546		AGC Batch 577		AGC Batch 579		AGC Batch 580	
Element No.	Length (in.)	Element No.	Length (in.)	Element No.	Length (in.)	Element No.	Length (in.)
15	13.6	9	13.6	1	13.7	4	13.6
37	13.6	13	13.6	19	13.6	29	13.6
40	13.6	22	13.6	26	13.7	34	13.7
52	13.6	46	13.6	33	13.7	58	13.5
55	13.6	49	13.6	66	13.6	73	13.6
88	13.6	63	13.6	70	13.5	91	13.7
		76	13.6	79	13.6		
				83	13.6		
Average	13.60		13.60		13.63		13.62

^aThis subassembly contained a mixture of elements cast by AGC and impact bonded by ANL and of ANL cold-line elements. Average postirradiation fuel-pin length of the 60 ANL cold-line elements in this subassembly was 13.57 in.

TABLE I.F.3. Fabrication and Irradiation Data for Driver Subassembly C-2200^a
(Calculated burnup: 0.76 at. % max; 0.65 at. % avg)

Injection-Casting Batch No.	Silicon Content of Fuel (ppm)	Number of Elements	Element Burnup Range (at. %)	Total Volume Swelling of Fuel ($\Delta V/V$, %)	
				Average	Range
<u>Vendor Fuel</u>					
AGC 546	460	6	0.70-0.75	4.5	3.6-5.8
AGC 577	440	7	0.71-0.75	3.5	3.0-3.9
AGC 579	440	8	0.70-0.76	3.8	2.9-5.2
AGC 580	390	6	0.69-0.75	3.6	3.1-4.8
<u>Cold-line Fuel</u>					
103 IH	405	3	0.75-0.75	4.5	3.7-4.9
134 IH	320	41	0.71-0.76	3.9	2.6-5.6
136 IH	343	14	0.70-0.72	3.1	1.7-4.2
140 IH	395	2	0.75-0.76	3.2	2.8-3.5
Avg (all elements)				3.8	

^aThis subassembly contained a mixture of elements cast by AGC and impact bonded by ANL and of cold-line elements.

impact-bonded AGC fuel did not shorten. The postirradiation lengths of pins made from this fuel were similar to the postirradiation lengths of pins made from ANL cold-line fuel. Table I.F.3 shows that irradiation swelling of the AGC fuel also was similar to the swelling of the cold-line fuel in the subassembly.

This subassembly is the second of three designed to demonstrate that impact-bonded AGC fuel is dimensionally stable under irradiation in EBR-II. The third subassembly of this group (C-2199), currently in the core, is scheduled to attain a goal peak burnup of 1.5 at. % during Run 39.

(2) AGC Elements Heat Treated before Centrifugal Bonding. Postirradiation examinations were made of the elements from Subassembly C-2209 following irradiation to a calculated maximum burnup of 0.35 at. % attained by the end of Run 36. In addition to cold-line elements, this subassembly contained 30 AGC elements made from three AGC casting batches (615, 616, and 620), which had been heat treated before centrifugal bonding. The heat treatment, which consisted of holding the elements for 1 hr at 500°C in air, was intended to transform the metastable gamma-phase uranium present in the cast alloy to alpha-phase uranium before centrifugal bonding. It is hypothesized that this transformation must be brought about before centrifugal bonding if dimensional instability of the fuel during irradiation is to be prevented.

Tables I.F.4 and I.F.5 summarize the postirradiation data obtained. Table I.F.4 shows that the prebonding heat treatment was effective in preventing shortening of the AGC centrifugally bonded fuel pins. These data for fuel-pin length are similar to those obtained for the cold-line elements in the subassembly. Irradiation swelling of the ANL and AGC fuel in the subassembly, as summarized in Table I.F.5, are reasonably similar.

TABLE I.F.4. Postirradiation Lengths of AGC Fuel Pins from Subassembly C-2209^a

(Calculated burnup: 0.35 at. % max)

AGC Batch 615		AGC Batch 616		AGC Batch 620	
Element No.	Length (in.)	Element No.	Length (in.)	Element No.	Length (in.)
7	13.5	14	13.5	22	13.5
25	13.5	15	13.6	23	13.6
32	13.5	16	13.6	24	13.5
33	13.6	34	13.6	44	13.5
36	13.5	35	13.6	46	13.6
45	13.6	56	13.5	47	13.6
58	13.7	57	13.6	55	13.5
59	13.6	61	13.5	60	13.6
77	13.6	68	13.6	67	13.6
78	13.5	69	13.6	70	13.5
Average	13.56		13.57		13.55

^aThis subassembly contained a mixture of AGC elements heat treated for 1 hr at 500°C in air before centrifugal bonding and of cold-line elements. Average postirradiation fuel-pin length of the 61 cold-line elements in the subassembly was 13.52 in.

TABLE I.F.5. Fabrication and Irradiation Data for Driver-fuel Subassembly C-2209^a
(Calculated burnup: 0.35 at. % max; 0.29 at. % avg)

Injection-Casting Batch No.	Silicon Content of Fuel (ppm)	Number of Elements	Element Burnup Range (at. %)	Total Volume Swelling of Fuel ($\Delta V/V$, %)	
				Average	Range
<u>Vendor Fuel</u>					
AGC 615	500	10	0.32-0.35	1.22	0.4-2.2
AGC 616	420	10	0.32-0.35	1.36	0.8-1.8
AGC 620	470	10	0.32-0.35	1.18	0.6-1.5
<u>Cold-line Fuel</u>					
119 IH	486	22	0.33-0.35	1.72	1.2-2.8
122 IH	418	39	0.32-0.35	1.77	1.1-3.5

^aThis subassembly contained a mixture of heat-treated AGC fuel elements and of cold-line elements. The AGC elements were heat treated by holding them for 1 hr at 500°C in air.

The elements from this subassembly were inspected in the air cell of the Fuel Cycle Facility and then reconstituted into a new subassembly (C-2262). Subassembly C-2262 was transferred to the reactor for additional irradiation beginning with Run 38. Target peak burnup for this subassembly is approximately 1.2 at. %.

(3) Study of the Transformation Kinetics for Vendor-produced Driver Fuel. This study is being made to provide information concerning the physical metallurgy of U-5 wt % Fs driver fuel produced by the Fuel Cycle Facility and by AGC. It will aid directly in the optimization of corrective heat treatment that may be required for AGC-produced elements now at ANL before they are used in EBR-II.

In the initial investigation, samples of AGC-cast and -bonded fuel were annealed at 500 and 600°C for 2, 5, 10, 20, and 40 min, and then quenched in water to determine the minimum time required to bring the phases to equilibrium. Measurements of hardness, density, and electrical resistivity were made after each heat treatment to follow the changes in the amounts of transformation products. Parallel microstructural examinations are being used to corroborate these changes.

(b) Postirradiation Surveillance. Table I.F.6 summarizes the surveillance performed on subassemblies received from the reactor and dismantled (or stored) in the air cell of the Fuel Cycle Facility prior to carrying out the indicated examinations.

c. Fuel Handling and Transfer (N. R. Grant and P. Fineman)

Table I.F.7 summarizes the fuel-handling operations during the month.

TABLE I.F.6. Postirradiation Surveillance Performed

Subassembly No.	Program	Burnup (at. % max)	Surveillance Performed
C-2184S	70%-enriched fuel	1.71	Fuel swelling; cladding straining
C-2187S	Controlled flow	1.5	Fuel swelling; cladding straining
C-2185S	Controlled flow	1.2	Metallographic examination of cladding for irradiation-induced changes in microstructure
C-2186S	Controlled flow	1.2	
B-397	Extended burnup	1.74	Fuel swelling
B-3037	Extended burnup	1.74	Fuel swelling
C-2169	Extended burnup	1.88	Fuel swelling; cladding straining
B-3038	Extended burnup	1.95	Stored in air cell; surveillance planned for next month
C-2147	Extended burnup	1.47	Metallographic examination of cladding for irradiation-induced changes in microstructure
C-2065	Extended burnup	1.47	

TABLE I.F.7. Summary of FCF Fuel Handling

	8/16/69 through 9/15/69	Total This Year
<u>Subassembly Handling</u>		
Subassemblies received from reactor		
Driver fuel (all types)	16	101
Experimental	3	18
Other (blanket)	0	4
Subassemblies dismantled for surveillance, examination, or shipment to experimenter		
Driver fuel	8	99
Experimental	3	18
Other (blanket)	0	2
Driver-fuel elements to surveillance	558	4937
Number from subassemblies	8	99
Subassemblies transferred to reactor		
Driver fuel	8	56
Experimental	2	7
<u>Fuel-alloy and Waste Shipments</u>		
Cans to burial ground	2	22
Skull oxide and glass scrap to ICPP	0	2
Recoverable fuel alloy to ICPP		
Fuel elements	4 (70.12 kg of alloy)	30 (504.79 kg of alloy)
Subassemblies	3 (17.60 kg of alloy)	7 (38 kg of alloy)
Nonspecification material	0	5 (84.3 kg of alloy)

d. Experimental Support (V. G. Eschen and N. R. Grant)

Remote reassembly of Subassembly XO40 was completed. The subassembly, now designated XO40A, is being stored in the air-cell pits until the 62.5-MWt reactor run (Run 38A) is completed.

Subassembly XO19, containing seven GE mixed-oxide fuel capsules, three UNC carbide-fuel capsules, one PNL graphite element, and eight PNL structural elements, was received from the reactor and dismantled. The ten fueled capsules were neutron radiographed, and all of capsules were prepared for shipment to the experimenters.

Subassembly XO60, containing seven INC structural elements, was dismantled, and all of its elements were returned to the experimenter.

Subassembly XO42A, containing four capsules with boron carbide and three with tantalum (all PNL), was dismantled. Five of the capsules were neutron radiographed. One of the remaining two capsules was returned to the experimenter. The five radiographed capsules, the one capsule remaining in the air cell, and one new capsule will be reassembled.

e. Reactor Support (N. R. Grant and V. G. Eschen)

(i) Subassembly Fabrication. Twelve depleted-uranium inner-blanket subassemblies have been fabricated and transferred to the reactor building, and an additional 12 will be fabricated during the next month. The 24 subassemblies should be sufficient for the needs of this fiscal year.

Six special subassemblies were fabricated for Run 38A. Four of these subassemblies (core type) were half-worth, i.e., they were loaded with 46 standard Mark-IA elements and 45 stainless steel dummy elements. The lower adapters of these subassemblies contain an orifice plate to reduce the coolant flow to $2/3$ of that in a standard subassembly. The other two subassemblies were standard Mark-A subassemblies loaded with 19 stainless steel dummy elements. These subassemblies also were orificed to reduce the coolant flow to a small fraction of that in a fueled Mark-A subassembly.

(ii) Blanket Surveillance. Surveillance plans for the depleted-uranium radial-blanket subassemblies have been modified to (1) evaluate the effects of 62.5-MWt operation on the blanket elements, (2) verify the data obtained at the established burnup limit of 0.27 at. %, and (3) determine if the present burnup limit can be increased. The surveillance plans now provide for examination of one blanket subassembly that had achieved approximately 0.27 at. % burnup before the 62.5-MWt reactor operations in Run 38A and another subassembly that will achieve approximately the same burnup after Run 38A. These examinations should provide information on what effects, if any, the 62.5-MWt operation had on blanket-element performance.

Additional subassemblies will be examined at burnup increments of approximately 0.03 at. % above the present established limit of 0.27 at. %. The limiting criteria for the determination of the new burnup limit will be cladding strain of 2% for the elements.

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More on Materiallography

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Thickness Corrections for Neutron-activated Gold Foils

G. S. Stanford and J. H. Seckinger
ANL-7545 (Feb 1969)

*ANL/EBR-003 is the third of a series of "blueback" topical reports to be prepared by the EBR-II Project.
**Georgia Institute of Technology.

II. GENERAL RESEARCH AND DEVELOPMENT

A. Studies and Evaluations

1. Power-system Modeling (T. D. Wolsko)

a. Utility Decision Model for Generating-plant Additions

(i) Model for Reliability Analysis of Electric-generation System (M. E. Stephenson)

Last Reported: ANL-7595, pp. 81-83 (July 1969).

Modifications and improvements have been made to the methods for modeling electrical-generating systems. The changes are:

(a) Previously, forced-outage frequency as a function of forced-outage magnitude was obtained by grouping the large number of individual forced-outage-frequency and forced-outage-magnitude points representing all possible combinations of plants in up and down conditions.* In the improved method, the computer searches through all combinations of plant forced outage and totals the frequencies for all those values whose forced-outage magnitude is equal to or greater than the critical value of forced outage that would cause the system to fail under the conditions corresponding to a particular day. This method avoids the somewhat arbitrary choice of an increment of forced-outage magnitude for grouping purposes. It also ensures that all frequencies corresponding to the critical forced-outage magnitude or greater are included.

(b) The calculations are now done in 26 two-week periods, rather than 12 one-month periods. Typical values of maximum biweekly loads were obtained from a utility.

(c) The program written to schedule maintenance (or re-fueling for a nuclear plant) has been incorporated into the main program. Provisions are available for the selection of 2-, 4-, 6-, and 8-wk maintenance periods for the individual plants.

(d) The variability of day-by-day load over a biweekly period, in terms of the ratio of standard deviation to mean load, has been made variable over the year.

Calculations with the improved model have been made for a generating system having a total system capacity of 10,000 MW, consisting of two 1000-MW plants, three 800-MW plants, four 600-MW plants, five

* Halperin, H., and Adler, H. A., Determination of Reserve-Generating Capability, Trans. AIEE, pp. 530-544 (Aug 1958).

400-MW plants, and six 200-MW plants. The maximum annual load was approximately 8000 MW. The two 1000-MW plants were assumed to be LMFBRs, the others fossil. For the fossil plants, forced-outage probabilities of 0.05 and average repair times of 150 hr were used. A utility indicated that fossil plants require an average shutdown of about 4 wk/yr for overhauling. For the nuclear plants, repair times were assumed to average 150 hr/yr, and plant forced-outage probability, p , was assumed to be 0.03, 0.05, 0.07, and 0.09; refueling outages were taken to be 4, 6, and 8 wk/yr.

Fig. II.A.1. Effect of LMFBR Forced Outage on System Reliability (for 4-wk refueling time)

conditions, the savings or penalties in required system reserves between various cases can be obtained by noting the horizontal increment between curves at the 5-yr level. Moderate changes in refueling times do not significantly change the total annual risk for the assumed generating system.

The reason is that a longer refueling time increases the total annual scheduled planned outage, and thus reduces the margin between capacity and load; however, much of the risk of system failure is in the months that have no planned outage.

A study was made of partial-load operation of the two LMFBR plants. Partial-load operation occurs when failure is not in the central portion of the plant (the reactor, etc.) but rather in a loop,

The results of varying the plant forced-outage probability for the LMFBRs are shown in Fig. II.A.1, where the refueling time was held constant at 4 wk/yr.

The results of varying the refueling time are shown in Fig. II.A.2, where the plant forced-outage probability for the LMFBRs was held constant at 0.05. With a plant forced-outage probability of 0.05 and a refueling time of 4 wk/yr for the LMFBRs, the mean time between system failures is 5 yr. For other

Fig. II.A.2. Effect of LMFBR Refueling Time on System Reliability (for plant forced-outage probability $p = 0.05$)

and the remaining loops can continue to operate. This condition was treated by assuming a forced-outage rate due to failure of the central portion of the plant and a separate rate due to failure of a loop. If there are three loops, the other two can continue to run when one is down. In general, if b is the fraction of the overall plant forced-outage probability due to a loop, and c is $(n - 1)/n$, where n is the number of loops, then the program multiplies the plant capacity by $(1 - bc)$ to calculate its average outage magnitude. A hand calculation of system forced-outage probabilities versus reserve margin for a sample case confirmed that this method is a good approximation. The possibility of the failure of more than one loop at a time was considered to be a second-order effect and so was neglected.

Figure II.A.3 shows the relation between failure frequency and system capacity for 1-, 2-, and 3-loop systems. In this case the forced outage was 0.03 for the central plant plus 0.04 for the loops, regardless of their number. This is an extreme set of assumptions and tends to set a bound.

Figure II.A.4 is similar to Fig. II.A.3, except that in place of 0.04 forced-outage rate regardless of the number of loops, a value of 0.02 was used for each loop. A comparison of the figures indicates that, for the system studied, the needed system capacity for a given failure frequency is not a strong function of loop forced-outage rate for a 3-loop plant.

Fig. II.A.3. Effect of Number of LMFBR Coolant Circuits on System Reliability

Fig. II.A.4. Effect of Number of LMFBR Coolant Circuits on System Reliability

2. Power-system Data (T. D. Wolsko)

Not reported previously.

An analysis of the cost of the fuel cycle for startup of Atomics International's Advanced LMFBR design for the LMFBR Task Force was performed using the CYCOST code. With the one-dimensional synthesis code STARBURN, startup mass-balance data were generated for the three startup strategies: (a) Pu_f startup, (b) ^{235}U startup, reloading with Pu_f , (c) ^{235}U startup, reloading twice with ^{235}U and then with Pu_f . A summarized cost comparison of the three startup strategies for two fuel prices is given in Table II.A.1. The cost penalty for ^{235}U startup for this reactor is not large on a 10-yr levelized basis. Depending on capital and fuel-cycle cost differentials between LWRs and LMFBRs, the ^{235}U startup of LMFBRs might be a preferred way of generating additional plutonium in a plutonium-scarce economy.

Table II.A.1. Fuel-cycle Costs during Reactor Life (mills/kWh)

	$\text{Pu}_f = \$10/\text{g}; \text{U}_3\text{O}_8 = \$8/\text{lb}$			$\text{Pu}_f = \$15/\text{g}; \text{U}_3\text{O}_8 = \$15/\text{lb}$		
	A	B	C	A	B	C
Cycle No. 1	1.14	2.00	2.00	1.30	2.26	2.26
2	0.86	1.35	1.60	0.97	1.53	1.80
3	0.82	1.05	1.52	0.95	1.20	1.73
4	0.81	0.80	1.27	0.94	0.93	1.45
5	0.79	0.79	1.03	0.92	0.91	1.18
Equilibrium	0.797	0.797	0.797	0.918	0.918	0.918
10-yr levelized	0.823	0.904	0.977	0.946	1.037	1.12
20-yr levelized	0.814	0.867	0.915	0.936	0.996	1.05
30-yr levelized	0.811	0.856	0.896	0.933	0.983	1.03

III. GENERAL REACTOR TECHNOLOGY

A. Applied and Reactor Physics Development

1. Theoretical Reactor Physics--Research and Development

a. Theoretical Reactor Physics

(i) Evaluation of Cross-section Data (D. A. Meneley)

Last Reported: ANL-7606, p. 89 (Aug 1969).

The work has been completed on deriving unresolved resonance parameters for U-235 and Pu-241 for the new ENDF/B compilations, and the data have been sent to Brookhaven.

A new ENDF/B library containing 76 materials has been prepared for the IBM-360 version of MC² by adding ten materials to the existing library. The new library contains all of the materials present on the CDC-3600 MC² library except deuterium, which the 360 system refuses to process correctly.

(ii) Reactor Computations and Code Development (B. J. Toppel)

Last Reported: ANL-7595, p. 84 (July 1969).

(a) The ARC System. The use of catalogued procedures for the IBM-360 largely eliminates the burden of the job-control language from the casual user. The facility provided by the symbolic parameters is particularly attractive in that they provide a largely user-oriented language as opposed to the less palatable job-control language.

Catalogued procedures have been provided in the ARC system for the existing standard path capabilities and for various utility purposes, such as modification and updating of the program modules.

User-execution catalogued procedures are named ARCSPXXX, where SPXXX corresponds to the standard path STPXXX which the procedure invokes, and XXX may take on any numeric value from 001 to 999.

ARC system-utility catalogued procedures are named ARCXXXX, where XXXX may take on any alphanumeric value except SPYYY, where YYY is an integer.

Use of the catalogued procedure also eases the user's burden from a purely mechanical point of view. For example, for a typical two-dimensional diffusion-theory problem, the user now must provide 8 control cards, whereas without the catalogued procedure 111 control cards were needed to execute the same problem.

The following catalogued procedures are currently available in the system:

Procedure	Use
ARCSP001	One-dimensional diffusion theory
ARCSP002	One-dimensional transport theory
ARCSP003	Two-dimensional diffusion theory
ARCSP004	Two-dimensional diffusion burnup
ARCSP005	Cross-section preparation
ARCSP006	One-dimensional diffusion perturbation
ARCSP007	Two-dimensional diffusion perturbation
ARCUCLP	Update, compile, and link edit to a private library
ARCCLP	Compile and link edit to a private library
ARCLP	Link edit to a private library

(b) MC² Capability in the ARC System

Last Reported: ANL-7577, pp. 130-131 (April-May 1969).

The standard path capability for multigroup cross sections, STP005,* is a driver path of the ARC system. This standard path invokes the modules CSI001, CSC001, CSC002, and CSC003 of the ARC system to generate the multigroup neutron cross-section data set XS.ISØ, using the algorithms of the MC² code. Record No. 2 of File 1 of data set XS.ISØ has been modified to include a temperature tag for each isotope that is added to the data set. The structure and content of this modified segment follows the format proposed by A. Lindeman *et al.*,** and is given below:

Record Type 2. (ISONME(I),I=1,NISO),(LOCA(I),I=1,NISO1),
(TEMP(I),I=1,NISO),(TNAME(I),I=1,NISO)NISO1=NISO+1

Variable Name

ISONME(I)	Name of Ith isotope (Real*8). Must be unique.
LOCA(I)	Number of records in second file to be skipped to read data for isotope I. (LOCA(1)=0). (Integer*4).
TEMP(I)	Temperature of Ith isotope (Real*8).
TNAME(I)	Ith isotope name as given on ENDF/B files (Real*8).

*Toppel, B. J., internal memorandum, Sept. 8, 1969.

**Lindeman, A., *et al.*, internal memorandum, Aug. 11, 1969.

This alteration was proposed* due to temperature-dependent effects of isotope cross sections which become significant in general nonstatic neutron problems.

The current testing on the IBM-360 of the four modules comprising the MC² capability in the ARC system continues. To date 85% of the testing of the input options available with the MC² standard path has been validated. A correction was made in CONTROL2** to handle properly the user-supplied fine-group weighting spectrum.

A catalogued procedure ARCSPO05 which invokes the standard path STPO05 has been added to the Applied Mathematics System Library, SYS1.AMDPRØC, and is currently being tested. By using this procedure, the number of job-control language (JCL) cards in the JCL deck required to run a job is greatly reduced.

The maximum code dimensions available for the calculations of an MC² problem in ARC are dictated by the available space in the main core of the computer. At present, the maximum dimensions of pertinent quantities are 20 materials, 4 isotopes for each material, 70 broad energy groups, 70 fine energy groups, 2100 ultrafine energy groups, 2000 resolved resonances, and 100 energy points in the unresolved resonance region. The next effort, just begun, in the refinement of MC² in ARC is to vary the dimensions of the arrays in modules CSI001, CSC001, CSC002, and CSC003 by the use of BPØINTR.† With this refinement the user can utilize bulk memory as well as the main core memory to expand the code dimensions. Another option is to expand some of the code dimensions at the expense of other dimensions by utilizing only the main core available in the computer.

(c) Resonance Absorption

Last Reported: ANL-7527, pp. 88-89 (Dec 1968).

A somewhat more accurate formulation for the regional slowing-down source in the RABBLE code†† has been devised and is being tested. In RABBLE, the lethargy range of interest is divided into many, extremely narrow intervals, or fine groups. Because a fine group is so narrow, the assumption has been made that the quantities of interest are constant within a fine group. In the current formulation, the refinement is made that the collision density has a linear dependence on lethargy within a fine group.

* Lindeman, A., et al., internal memorandum, Aug. 11, 1969.

** Toppel, B. J., et al., ANL-7318 (June 1967).

† Kennedy, A. S., A Dynamic Storage Allocation Program, Applied Mathematics Division ANL Technical Memorandum No. 98 (Feb. 28, 1967); Corrections to PØINTR for LCS, Applied Mathematics Division, ANL Internal Technical Memorandum (Jan. 29, 1969).

†† Kier, P. H., and Robba, A. A., ANL-7326 (1967).

The basic quantity of interest is S_k , the rate with which neutrons are scattered into the k^{th} fine group, which extends from u_0 to $u_0 + \Delta u$, in some region. For simplicity of notation, the region index is surpassed and only one species of scatterer is assumed. S_k is given formally by

$$S_k = \int_{u_0}^{u_0 + \Delta u} s(u) du, \quad (1)$$

where $s(u) du$, the rate with which neutron are scattered into du about u , is given by

$$s(u) = \frac{1}{1 - \alpha} \int_{u - \epsilon}^u f_s(u') e^{-(u - u')} du'; \quad (2)$$

here $f_s(u') du'$ is the scattering rate in du' about u' , and α and ϵ have their usual meaning in slowing-down studies.

The first step in the derivation is to differentiate Eq. (2) to obtain

$$\frac{ds(u)}{du} = \frac{1}{1 - \alpha} [f_s(u) - \alpha f_s(u - \epsilon)] - s(u). \quad (3)$$

Equation (3) is readily solved by multiplying by the integrating factor, e^u , and then integrating between u and $u - \Delta u$:

$$s(u) = e^{-\Delta u} s(u - \Delta u) + I(u) e^{-u}, \quad (4)$$

where

$$I(u) = \frac{1}{1 - \alpha} \int_{u - \Delta u}^u e^u [f_s(u') - \alpha f_s(u' - \epsilon)] du'. \quad (5)$$

Upon substitution of Eq. (4) into Eq. (1) we obtain

$$S_k = e^{-\Delta u} S_{k-1} + \int_{u_0}^{u_0 + \Delta u} I(u) e^{-u} du. \quad (6)$$

To ease the evaluation of $I(u)$, we assume that the scattering densities have a linear dependence on lethargy, that is, for $u \geq u' \geq u - \Delta u$,

$$f_s(u') = f_s(u - \Delta u) + (u' - u + \Delta u) f'_s(u - \Delta u), \quad (7)$$

where in the last term on the right the prime denotes the derivative. Upon substituting for $f_s(u')$ into Eq. (5), we obtain

$$I(u) = \frac{e^u}{1-\alpha} \{ (1 - e^{-\Delta u}) [f_s(u - \Delta u) - \alpha f_s(u - \Delta u - \epsilon)] (1 - e^{-\Delta u - \Delta u}) [f'_s(u - \Delta u) - \alpha f'_s(u - \Delta u - \epsilon)] \}. \quad (8)$$

Upon substituting from Eq. (8) into Eq. (6) we obtain

$$S_k = e^{-\Delta u} S_{k-1} + \frac{(1 - e^{-\Delta u})}{1-\alpha} [F_{s_{k-1}} - \alpha F_{s_{k-L-1}}] - \frac{(1 - e^{-\Delta u - \Delta u})}{1-\alpha} \left\{ f_s(u_0) - f_s(u_0 - \Delta u) - \alpha [f_s(u_0 - \epsilon) - f_s(u_0 - \epsilon - \Delta u)] \right\}, \quad (9)$$

where f_{s_j} is the scattering density integrated over the j th group. Now, since it is assumed that the collision density is linear within a fine group,

$$f_s(u_0) - f_s(u_0 - \Delta u) = \Delta u f'_s(u_0 - \Delta u) \quad (10a)$$

and

$$f_s(u_0 - \epsilon) - f_s(u_0 - \epsilon - \Delta u) = \Delta u f'_s(u_0 - \epsilon - \Delta u). \quad (10b)$$

At these points the derivative of the scattering density is discontinuous and hence will be approximated by

$$\begin{aligned} \Delta u f'_s(u_0 - \Delta u) &\approx f_s(u_0 - \frac{1}{2} \Delta u) - f_s(u_0 - \frac{3}{2} \Delta u) = \bar{f}_{s_{k-1}} - \bar{f}_{s_{k-2}} \\ &= (F_{s_{k-1}} - F_{s_{k-2}}) / \Delta u; \end{aligned} \quad (11a)$$

similarly

$$\Delta u f'_s(u_0 - \epsilon - \Delta u) = (F_{s_{k-L-1}} - F_{s_{k-L-2}}) / \Delta u. \quad (11b)$$

Upon substituting from Eqs. (11) into Eq. (9) we obtain our working equation for S_k :

$$S_k = e^{-\Delta u} S_{k-1} + \frac{(1 - e^{-\Delta u})}{1-\alpha} [F_{s_{k-1}} - \alpha F_{s_{k-L-1}}] - \frac{(1 - e^{-\Delta u - \Delta u})}{(1-\alpha) \Delta u} [F_{s_{k-1}} - F_{s_{k-2}} - \alpha \{ F_{s_{k-L-1}} - F_{s_{k-L-2}} \}]. \quad (12)$$

We see that Eq. (12) is a suitable expression for the evaluation of S_k since the right-hand side involves quantities for groups at higher energies which have already been evaluated when group k is reached.

2. Nuclear Data--Research and Development

a. Cross Section Measurements (N. D. Dudey)

(i) Radiative Capture Cross Sections

Last Reported: ANL-7581, p. 83 (June 1969).

A final report on measurements of the radiative capture cross sections for ^{89}Y , ^{85}Rb , and ^{87}Rb has been completed and submitted for publication. Because of budget restrictions, no further work will be done on this task during FY 1970.

(ii) Differential (n, x) Cross-section Measurements

Not previously reported.

As a result of budget restrictions and a reduction of manpower no work on this task will be initiated in FY 1970.

(iii) Spectrum-averaged Measurements

Last Reported: ANL-7577, p. 135 (April-May 1969).

Preliminary values have been determined for the number of captures, the number of fissions, and the capture-to-fission ratios (α) for samples of ^{233}U irradiated in various positions in EBR-II. In this experiment, weighed amounts of high-purity ^{233}U , which had been analyzed mass-spectrometrically, were introduced into individual capsules and inserted into pins of driver subassemblies. The samples were irradiated for periods ranging from 0.5 to 1.6 yr. The irradiated samples were dissolved, together with their stainless steel capsules, and aliquots taken for individual analyses. After radiochemical separation of the uranium, the number of capture products (^{234}U atoms) was determined by mass-spectrometric isotopic-dilution analysis.* The number of fissions was determined by direct counting of ^{137}Cs with a high-resolution Ge(Li) detector, assuming a fission yield for ^{137}Cs of 6.51%. The cesium values were corrected for in-pile decay and for fluctuations in the power level of the reactor during irradiation. The preliminary results are presented in Table III.A.1 for nine positions in EBR-II. The values given in columns 4, 5, and 6 are averages of duplicate measurements. No corrections have yet been made for burnup of the target or product nuclei, and assignments of errors to the data have not been completed.

*The mass spectrometry was performed by Matthew Laug at the EBR-II site.

TABLE III.A.1. Capture-to-Fission Ratio (Alpha) as a Function of Position in EBR-II

Distance from Core Center		Accumulated Exposure (MWd)	Captures per 100 ^{233}U Atoms	Fissions per 100 ^{233}U Atoms	Alpha
Radial (cm)	Axial (cm)				
2.86	-17.0	3069	2.433	0.177	0.0727
2.86	-8.5	3069	3.130	0.216	0.0690
2.86	+0.0	3069	3.386	0.230	0.0679
2.86	+8.5	3069	3.103	0.214	0.0690
2.86	+17.0	3069	2.316	0.168	0.0726
2.86	+51.5	3069	0.492	0.0464	0.0943
21.25	-3.49	4293	3.176	0.237	0.0746
40.83	-2.30	7640	1.699	0.148	0.0871
61.24	-2.30	9985	0.3711	0.0357	0.0962

The radiochemical and mass-spectrometric analyses have been completed for the number of captures and fissions in samples of ^{239}Pu irradiated in EBR-II; the computation of results is nearing completion. Radiochemical analysis of irradiated ^{235}U is beginning.

In other work, analysis of data from the EBR-II Dosimetry Test Experiment (Run 31F) is continuing. Spectrum-averaged cross sections for 14 reactions in Rows 2, 4, 6, and 7 of EBR-II have been calculated from measured reaction rates and experimentally determined flux levels at the sample positions. These values, which have been corrected to core mid-plane are given in Table III.A.2. The measured flux levels, which were determined from monitor reactions discussed in ANL-7577, were 2.24×10^{15} , 2.26×10^{15} , 2.25×10^{15} , and 1.82×10^{15} for Rows 2, 4, 6, and 7, respectively. These flux values are preliminary evaluations and may involve uncertainties in excess of $\pm 20\%$.

TABLE III.A.2. Calculated Spectrum-averaged Cross Sections in EBR-II

Reaction	Spectrum-averaged Cross Section (mb)			
	Row 2	Row 4	Row 6	Row 7
$^{59}\text{Co}(n,\gamma)^{60}\text{Co}$	14.35	14.2	18.0	23
$^{54}\text{Fe}(n,\gamma)^{55}\text{Fe}$	8.22	8.4	8.0	11
$^{62}\text{Ni}(n,\gamma)^{63}\text{Ni}$	7.54	8.1	9.7	13
$^{58}\text{Fe}(n,\gamma)^{59}\text{Fe}$	3.82	4.0	4.7	6
$^{63}\text{Cu}(n,p)^{63}\text{Ni}$	4.87	4.7	4.3	2
$^{47}\text{Ti}(n,p)^{47}\text{Sc}$	4.34	4.9	3.7	2
$^{32}\text{S}(n,p)^{32}\text{P}$	3.47	3.4	3.0	2
$^{59}\text{Co}(n,p)^{59}\text{Fe}$	2.68×10^{-1}	2.8×10^{-1}	2.4×10^{-1}	1×10^{-1}
$^{48}\text{Ti}(n,p)^{48}\text{Sc}$	5.16×10^{-2}	5.0×10^{-2}	4.1×10^{-2}	2×10^{-2}
$^{58}\text{Ni}(n,\alpha)^{55}\text{Fe}$	8.90×10^{-1}	8.0×10^{-1}	7.8×10^{-1}	4×10^{-1}
$^{50}\text{V}(n,\alpha)^{47}\text{Sc}$	2.11×10^{-1}	9.5×10^{-2}	9.9×10^{-2}	7×10^{-2}
$^{54}\text{Fe}(n,\alpha)^{51}\text{Cr}$	1.68×10^{-1}	1.8×10^{-1}	1.5×10^{-1}	7×10^{-2}
$^{63}\text{Cu}(n,\alpha)^{60}\text{Co}$	1.01×10^{-1}	9.2×10^{-2}	8.5×10^{-2}	5×10^{-2}
$^{51}\text{V}(n,\alpha)^{48}\text{Sc}$	4.07×10^{-3}	4.2×10^{-3}	3.4×10^{-3}	2×10^{-3}

A similar experiment, involving the irradiation of foils in a soft-spectrum core of ZPR-3 has been completed, and radiochemical analyses of the reaction products are in progress.

(iv) Yields of Low-mass Fission Products

Last Reported: ANL-7581, p. 85 (June 1969).

A preliminary experiment to measure the fission yield of tritium (see ANL-7581) disclosed two principal problems: (1) radiation damage to the fission detectors and (2) strict requirements for setting the timing of the coincidence system. It has been concluded that, in spite of the deterioration of the silicon detectors during irradiation, these detectors are still the best choice; when the operating characteristics become sufficiently degraded, they will be replaced. The timing problem can be solved by using a ^{252}Cf source in a special scattering chamber to set the timing before experimental measurements begin. The available facilities at the Dynamitron do not include a suitable scattering chamber. Drawings for the special chamber have been completed, but budget restrictions may not permit its construction. In addition, the recent loss of manpower due to budget restrictions makes the execution of the particle-identification experiment according to the original schedule appear unlikely.

As a result, an alternative experiment, which is less expensive but also less definitive, is being considered for determination of fast-neutron tritium yields. This experiment involves irradiation of gram quantities of fissile material in the Dynamitron, followed by radiochemical separation and counting of the tritium. This approach appears feasible with the fast-neutron intensities expected from the Dynamitron, provided sufficient irradiation times can be made available. If the radiochemical method proves to be feasible upon further evaluation, the principal effort during FY 1970 will be directed toward this method. The radiochemical and particle-identification methods, which employ widely different approaches, should ultimately supplement each other and add considerable confidence to the final values for tritium fission yields.

b. Burnup Analysis and Fission Yields for Fast Reactors
(R. P. Larsen)

(i) Development of Analytical Procedures for Fission-product-burnup Monitors

Last Reported: ANL-7595, p. 89 (July 1969).

An X-ray spectrometric method is being developed for the determination of the four principal rare earth fission products: lanthanum, cerium, praseodymium, and neodymium. In the procedure, terbium (a rare earth not produced in fission) is added to the sample as an internal standard;

the rare earths are then separated from interferences, electroplated on aluminum, and assayed X-ray spectrometrically. The measured line intensities are corrected for interelemental effects by means of a computer program; and the amount of each rare earth is calculated from the rare earth-to-terbium line-intensity ratios.

Recent effort has been directed toward eliminating a small negative bias (see ANL-7595) by means of an empirical correction factor. The negative bias has been attributed to a mass-absorption effect, since the amount of solids on plates containing a mixture of rare earths is much greater than that on plates of the individual rare earth standards. The use of standards consisting of mixed rare earths, in amounts proportional to their fission yields, was, therefore, considered as a means of eliminating the bias.

The feasibility of this approach was tested on a series of plates containing 40 μg of terbium and 30 to 180 μg of rare earths. Some of these plates were used as "standards" to prepare a calibration curve; others, containing $\sim 100 \mu\text{g}$ of rare earths, were treated as "unknowns." The results of the analysis of eight "unknowns" and the relative standard deviations are as follows:

Rare Earth	% Found
La	100.3 ± 3.9
Ce	98.0 ± 6.6
Pr	99.2 ± 5.2
Nd	101.1 ± 0.9
La + Ce + Pr + Nd	99.6 ± 3.2

The results indicate that this method of calibration eliminates the negative bias. The decrease in the precision of these results, as compared with previous results, may have been caused, at least partially, by instrumental difficulties.

In further work, the entire procedure was tested on a solution that simulated an irradiated fuel sample. Aliquots of this solution containing 100 mg uranium, $\sim 100 \mu\text{g}$ of rare earths, and appropriate amounts of the other major fission products (Rb, Sr, Y, Zr, Mo, Ru, Rh, Pd, Sn, Sb, Te, Cs, and Ba) were taken through the separation procedure, after which the rare earths were electroplated and assayed. The amount of each rare earth was then calculated from the previously prepared calibration curve. The results of four analyses and the relative standard deviations are summarized below:

Rare Earth	% Found
La	93.3 $\pm$ 3.4
Ce	94.5 $\pm$ 7.0
Pr	93.5 $\pm$ 2.7
Nd	99.4 $\pm$ 1.6
La + Ce + Pr + Nd	95.9 $\pm$ 2.5

The reason for the low results has not yet been determined. Various steps in the procedure are being examined as possible sources of difficulty.

The entire procedure was also tested on a solution of UO_2 fuel (13% enriched) that had been irradiated in MTR to ~ 6 at. % burnup. The UO_2 sample had previously been analyzed for total rare earths by EDTA titration (see Progress Report for January 1968, ANL-7419, p. 96). The test with irradiated UO_2 was made to determine whether any major problems, not previously considered, would be encountered in the analysis of actual fuel samples. The concentrations of the four rare earths, as determined by X-ray spectrometric analysis, were compared with the concentrations calculated (on the basis of fission yields of the individual rare earths) from the results of the total rare earth (TRE) titration analysis. This comparison is given below:

Rare Earth	Concentration ($\mu\text{mol/ml}$)	
	Determined by X-ray Analysis	Calculated from TRE Analysis
La	0.0842	0.0857
Ce	0.1752	0.1622
Pr	0.0762	0.0805
Nd	0.2706	0.2722
La + Ce + Pr + Nd	0.6062	0.6005

The agreement of these results indicates that the analysis of irradiated samples presents no unforeseen problems and lends confidence to the potential applicability of the method to burnup analysis; however, as discussed earlier, several problems must be solved before the precision and accuracy of the method can be fully evaluated.

(ii) Development of a New Method for the Determination of the Absolute Fission Yields of Burnup Monitors for Fast Reactor Fuels

Last Reported: ANL-7581, pp. 85-86 (June 1969).

(a) Determination of Alpha for ^{239}Pu , ^{233}U , and ^{235}U . An experiment is being conducted to measure the effective capture-to-fission ratio (alpha) for ^{239}Pu , ^{233}U , and ^{235}U in a neutron spectrum that emphasizes

the 10-keV region, where the greatest uncertainties in alpha exist. The planned irradiation of samples of the three isotopes in ZPR-3 has been completed. The number of captures will be determined by mass spectrometry from the change in ^{240}Pu , ^{234}U , and ^{236}U contents; the number of fissions will be determined by gamma spectrometry from the activity of the fission product ^{140}Ba . Prior to the long irradiation (82.3 hr), a short irradiation (80 min) was performed to establish the relationship between the ^{140}Ba activity and the number of fissions. Two types of samples were included in the short irradiation: (1) heavy samples (~15 mg) of the three fissile isotopes, which will be used to determine the ^{140}Ba activity, and (2) light samples (~2 ng) of the three isotopes in contact with mica fission-track counters, which will be used to determine the number of fissions.

Analytical measurements of the irradiated samples have begun; fission-track counting of the samples from the short irradiation has been completed and gamma-spectrometric analysis of the samples from both irradiations is in progress.

c. Reactor Code Center (M. Butler)

Last Reported: ANL-7606, pp. 90-91 (Aug 1969).

Seven program packages were incorporated into the Center library this month.

Four deal with the processing of ENDF/B tapes, and the preparation of multigroup cross sections from them. They include the set of Fortran IV routines developed at the National Neutron Cross Section Center--CHECKER, CRECT, DAMMET, PLOTFB, and SLAVE3--designed to permit the user to:

- 1) check and correct the data on an ENDF/B BCD card image tape,
- 2) delete data from, alter the mode of, and merge ENDF/B tapes,
- 3) edit and list and/or plot data from ENDF/B tapes, and
- 4) retrieve and process data for later calculations.

This package is described in the document ENDF-110 and is written for the CDC-6600. In addition, both IBM-7094 and CDC-6600 revised packages are available for the BNL AVERAGE program (ACC Abstract No. 376) which calculates the scattering, capture, and fission cross sections in the unresolved region from ENDF/B data and a UNIVAC-1108 package for the ETOX group-constant preparation program (ACC Abstract No. 388) developed at Battelle Northwest Laboratory has been added to the library.

A revised version of the TWIGL two-dimensional space-time diffusion program replaced the earlier package (ACC Abstract No. 338). This version incorporates two corrections affecting problems in cylindrical geometry. In the previous version the radius calculation used in the region-power calculations was in error for multiple thermal-hydraulic region problems, and the centerline fluxes ($r=0$) were computed incorrectly. These flux-calculation errors were noticeable only when small time steps were used. The current TWIGL package written for the CDC-6600 corrects these errors.

The GGA UNIVAC-1108 program COHBE which calculates the coherent inelastic-scattering law for polycrystalline beryllium was added this month, along with the ORNL IBM-7090 program LYNNE, which performs the multipole expansion of the Woods-Saxon potential. These two programs are ACC Abstracts 385 and 381, respectively.

B. Reactor Fuels and Materials Development

1. Fuels and Claddings--Research and Development

a. Behavior of Reactor Materials

(i) Metallic Fuels (H. V. Rhude)

Last Reported: ANL-7606, p. 99 (Aug 1969).

Burnup determinations by neodymium analyses have been completed from midlength portions of three U-Pu-Zr fuel pins irradiated in EBR-II. The results are about 20% lower than the calculated maximum burnups previously reported (see Progress Report for April-May 1969, ANL-7577, p. 152). One burnup determination was also made by technetium analysis. This analysis agrees well with the neodymium analysis. Results are given in Table III.B.1. Calculations of fuel-pin swelling, based on measurements of neutron radiographs, also are given for six U-Pu-Zr fuel elements irradiated in EBR-II. The average swelling of the four fuel elements from Subassembly XO28 (0.77% burnup) is 37.4%. Assuming that

TABLE III.B.1. Burnup and Swelling of U-Pu-Zr Fuel Elements Irradiated in EBR-II

Fuel Element	Subassembly	Burnup (at. %)			Swelling, $\Delta V/V \times 100$ (%)
		Calculated	Nd Analysis	Tc Analysis	
BA01	XO28	1.0	$0.77 \pm 5\%$	-	38.8
BA02	XO28	1.0	-	-	35.5
BA03	XO28	1.0	-	-	39.1
BA04	XO28	1.0	-	-	36.4
ND23	XG06	5.2	$4.22 \pm 7\%$	-	59.5
ND24	XG05	6.9	$5.61 \pm 6\%$	$5.57 \pm 5\%$	47.2

fuel elements ND23 and 24 (4.2 and 5.6% burnup, respectively) followed the same swelling pattern, these data indicate that from 60 to 80% of the swelling occurred before 1% burnup was reached. In all cases, the fuel had swelled to the cladding (as determined by neutron radiography and supplemented by metallography). Apparently, the restraining effect of the cladding and the release of fission gas from the fuel decreased the swelling rate after approximately 1 at. % burnup.

b. Oxide Fuel Studies (L. A. Neimark, F. L. Brown, W. F. Murphy, and E. J. Petkus)

(i) Examination of Group O-2

Last Reported: ANL-7606, pp. 1-3 (Aug 1969).

Metallographic examination of irradiated cladding from Group O-2 Vipac mixed-oxide elements continued on Elements SOV-3 and SOV-7 (Type 304 stainless steel cladding), HOV-15 (Hastelloy-X cladding), and TVOV-1 (V-20 w/o Ti cladding). Transverse samples from near the top end, the midlength, and the bottom end of SOV-3, SOV-7, and TVOV-1 were examined, and samples from near the top end and the midlength of HOV-15 were examined.

Grain boundaries in the stainless steel and Hastelloy-X were delineated by carbide precipitation during the long irradiation exposures at temperatures between 460 and 610°C. Further changes in the microstructures were not apparent, even though the chemical-etching characteristics were considerably altered by the irradiation.

Comparison of microhardness values of the irradiated versus unirradiated stainless steel disclosed significant increases in the irradiated material that could be correlated with axial location in the elements; the hardness increased from 26 to 57% from the preirradiation values of 174 (DPH 400-g load) with the largest increases occurring in the cooler operating regions near the bottom ends of both elements. A similar trend was observed in the Hastelloy-X cladding. Unlike the Fe-Cr-Ni alloys, the hardness of the irradiated V-20 w/o Ti cladding in Element TVOV-1 did not vary significantly along the length of the fuel column.

(ii) Preparation of Group O-3

Last Reported: ANL-7595, p. 2 (July 1969).

Eighteen pellet and vibratorily compacted fuel elements of Group O-3 have been inspected and accepted for irradiation. Preirradiation measurements of the fuel elements have been completed. Preparation of

the capsule tubes is nearly completed, and trial sodium-bonding runs have been made. Writing of the "Experiment Description and Hazards Evaluation" is complete.

2. Techniques of Fabrication and Testing--Research and Development

a. Nondestructive Testing Research and Development (H. Berger)

(i) Neutron Radiography (J. P. Barton)

Last Reported: ANL-7581, pp. 91-93 (June 1969).

A program is under way to study the possibilities of radiography using neutrons, and an important part of this program is to compare the potentialities of other neutron sources--accelerators and isotopic sources. Californium-252 is one promising neutron source, and the AEC has undertaken a study* of its application potentialities using the small amounts presently available. For these purposes a 141- μg ^{252}Cf source has been received, on loan from the Savannah River Laboratories. The neutron-radiography experiments are being conducted in a kilocurie hot-cell facility, using remote handling to manipulate the source.

The fast-neutron yield is 3.38×10^8 n/sec and Table III.B.2 presents measurements of the neutron flux as moderated in a light water environment of effectively large dimensions. The optimum ratio for fast neutron yield to thermal neutron flux of 76 is particularly good, and represents an advantage of 270% over ^{241}Am - ^{242}Cm -Be. The result may be due partly to the neutron-energy spectrum, and partly to the small self-absorption of the californium. Experiments are under way to study the advantage of using moderators of high hydrogen density with the ^{252}Cf neutron source.

TABLE III.B.2. Thermal-neutron Flux Distribution for ^{252}Cf Source in Water

Distance from Source (in.)	Cadmium Ratio (Indium) $\left(\frac{\text{Thermal Flux}}{\text{Cadmium Covered Flux}} \right)$	Thermal- neutron Flux (10^6 n/cm ² -sec)	Ratio $\left(\frac{\text{Fast Neutron Yield}}{\text{Thermal Neutron Flux}} \right)$
1	10.8	4.44	76.1
1.5	12.1	3.88	87.1
2	12.8	3.15	107
2.5	14.5	2.53	133
3	15.6	1.98	170
3.5	17.2	1.48	227

*USAEC, Major Activities in the Atomic Energy Programs, January-December 1969, pp. 39-43.

Activation-transfer neutron radiography has been performed with this small source, but only low collimation systems can be used. A collimator of length-to-diameter ratio 6:1 gave a density of 0.62 on Kodak Royal Blue film using 125- μ dysprosium converter screen under saturation conditions. High-quality neutron radiography of thick objects, such as radioactive fuel elements, would be possible using higher collimation and correspondingly larger californium sources.

The neutron-to-gamma radiation ratio for the source is found to be sufficiently high to permit good-quality direct-exposure neutron radiography, using either ^6LiF -ZnS scintillators or gadolinium metal foil as converter. The possibility of using gadolinium enables obtaining radiographs of very fine grain; of course, with a low-yield source and high collimation, exposure times are long. Useful film densities can be achieved with neutron exposures ranging from $5 \times 10^5 \text{ n/cm}^2$ (^6LiF -ZnS plus Royal Blue film) to 10^8 n/cm^2 (50 μ gadolinium plus AA film).

Fig. III.B.1
Photograph of Double-exposure Hologram with Uneven Heating due to Small Heater

Quantitative results have been obtained for the radiographic quality observed using the ^{252}Cf source under various conditions. These data include measurements of radiographic density, contrast, geometric resolution, and inherent resolution. One significant advantage for radiographic quality has been noted on separating the collimator input aperture from the source by a 5-cm air channel. This can double the contrast of 0.5-mm cadmium from 0.3 to 0.6 density unit; likewise, the contrast of 3-mm plastic is increased from 0.15 to 0.3 density unit; on the other hand, the contrast of lead, which is sensitive to gamma rather than neutron radiation, is actually decreased.

(ii) Laser Studies and Holography (N. Fernelius)

Last Reported: ANL-7581, pp. 93-96
(June 1969).

Some additional applications of double-exposure holography will be discussed. In this method the object is displaced slightly, due to thermal expansion, stress, etc., between the two exposures.

In the first example a small local heater, consisting of 10 cm of 15-mil platinum wire wound in a coil about 1 cm long, was inserted in a fuel-rod jacket. The heater was located at the level of the bar on a ruler (see Fig. III.B.1). One exposure was taken at room temperature, then the heater was turned on. After 5 min a second

exposure was taken. As can readily be seen, the fringe system was non-uniform, indicating that the rod did not expand uniformly.

Some studies were made on tubes of two dissimilar metals soldered together. In one case we used a copper-stainless steel specimen. We had difficulty in making good-quality holograms since the reflectivity of the stainless steel section in the red laser light was very poor. Figure III.B.2a shows a double exposure of the specimen placed on a hot plate before a mirror. Room temperature was 69°F for the first exposure of 6 sec; the hot plate was set at 106°F for the second exposure of 8 sec. A change in the fringe system at the interface of the two metals is more noticeable on the mirror image. A brass-Type 304 stainless steel specimen was made and vapor blasted to increase surface luminosity. A double-exposure hologram made by varying the hot plate from 107 to 72°F is shown in Fig. III.B.2b. A change at the interface is more readily visible. Good-quality holograms were also obtained by spraying the specimen with flat white paint. However, one has some doubts as to whether the fringe system seen represents effects of the specimen only or includes properties of the paint coating.

Fig. III.B.2. Photographs of Double-exposure Holograms of (a) a Copper-Stainless Steel Specimen and (b) a Brass-Type 304 Stainless Steel Specimen on a Hot Plate

Attempts were made to record effects of etching on an aluminum block dipped in Tucker's etch, but no fringe systems were seen. Apparently, the etching action is very nonuniform and one does not end up with a surface similar enough to the initial surface to cause interference fringes.

b. Nondestructive Testing Measurement of Effective Cold Work in Cladding Tubes (N. J. Carson)

Not previously reported.

A program is under way to study the feasibility of measuring the effective cold work in cladding tubes by nondestructive testing (NDT) techniques. The work is aimed specifically at AISI Type 316 stainless steel tubing of the size and shape to be used in FFTF. This program is divided into two elements: (1) the characterizing of cold-worked tubes to be used in standardizing the nondestructive testing techniques, and (2) the screening of a series of potentially suitable nondestructive testing techniques.

The objective of element one is (1) to verify what takes place when a piece of AISI Type 316 stainless steel is deformed at room temperature, and (2) to develop a simple technique to measure the degree of deformation induced. The primary technique for the assessment of plastic deformation will be transmission electron microscopy (TEM). With it, the substructure of worked steel will be observed and a plot will be made of degree of reduction in area versus substructure characteristics. A secondary technique will be sought that correlates well with the substructure characteristics, but requires less time and equipment than TEM. This secondary technique which should require only equipment normally available in fabricating plants, will then be used to characterize the tubes used as NDT standards.

In the first phase of this work, a survey was made of the literature on the effects of cold work in stainless steel. This indicated the most profitable experimental approach, discussed below, and also the important fact that cold work will influence the nature of the precipitation of carbides and sigma phases during service at reactor temperature and that this will, in turn, have an effect on mechanical properties and possibly on dimensional stability.

(i) Experimental Approach. The experimental approach consists of the following steps: (1) the preparation of a series of uniformly worked Type 316 stainless steel rods, (2) the examination of samples from each rod by TEM, and (3) the submission of samples from each rod to a series of destructive and nondestructive tests. The object is to relate substructure characteristics, as observed by TEM, to the findings of more conventional destructive tests and nondestructive tests.

Cold work is generally defined in terms of the percent reduction in cross-sectional area. This terminology implies a state function. But the amount and distribution of strain over the cross section is very much dependent on the process and the operating conditions. Light reductions by rolling or drawing generally strain the outside surfaces of the work more than the center portions. Grinding produces highly strained surface layers, but has little effect of the interior. A given reduction in area accomplished in a series of light reductions will result in a strain distribution quite different from the same reduction accomplished in a single pass. Percent reduction in area cannot define deformation for a process that does not strain the entire cross section uniformly.

The number of samples required to evaluate the structure of the worked rods by TEM will be less if the rods can be strained uniformly. Present plans are to measure the dislocation densities, the diameter of subcells if present, and the density of deformation twins where applicable.

The major portion of each worked rod will be used in screening potential NDT techniques, chiefly those based on changes in magnetic susceptibility or electrical resistivity. Samples of each rod will be submitted to conventional destructive tests. Optical microscopy will be used to measure either density of deformation twins in the as-worked material or the distribution of precipitates in worked and heat-treated samples. Both phenomena have a direct casual relationship to cold work. Other conventional measurements, such as microhardness measurements, will be tried as well as some of the more elaborate measurements, such as X-ray asterism.

(ii) Progress. The initial work was directed toward the selection of a suitable annealing temperature. Pieces of Type 316 stainless steel rod, approximately 3/8-in. in diameter, were heat treated at the combinations of time and temperature shown in Fig. III.B.3. Heat treatments were carried out in an atmosphere of flowing argon, and all pieces were water quenched from the heat-treating temperature.

	Times (min)			
	5	20	60	180
Temperature (°F)		X		
1550		X		
1750		X	X	X
1850	X	X	X	
1950		X		
2150		X		

Fig. III.B.3. Time-Temperature Combinations Used in Annealing Study.

Those heat treatments carried out below 1850°F failed to accomplish carbide dissolution in reasonable time. Most of the heat treatments at higher temperature produced a nearly single-phase alloy but also produced grain growth. The optimum heat treatment

appeared to be between 1850 and 1900°F for approximately 20 min. This should produce a single-phase steel with an ASTM grain size in the 7-9 range.

The technique chosen for introducing cold work was tensile elongation. Tube drawing was not considered because of the large amount of redundant work in the process that cannot be included in a simple term such as "percent reduction in area." Every tube drawn includes a sinking operation in which the tube is drawn down onto the mandrel, a reduction in the cross section of the tube, and a reeling operation in which the tube is expanded for mandrel removal. Neither the sinking nor the expansion steps are represented in the conventional term "reduction in area."

The only two processes considered were rod drawing and tensile elongation, rod drawing because of its similarity to tube drawing, and tensile elongation because it is known to strain the cross section uniformly. The choice between these two techniques depended on the degree to which the strain varied over the cross section of a drawn rod.

Two specimens from the same Type 316 stainless steel rod were cold reduced 14%, one by rod drawing and one by tensile elongation. Samples of each were prepared metallographically, and micro-hardness measurements were made. There was a significant hardness difference between center and edge of the drawn specimen. There was also a significant difference in structure between the drawn and elongated samples. The structure of the drawn rod (see Fig. III.B.4) showed a higher density of deformation twins. This difference can be made more dramatic by a short 1200°F heat treatment. It appears that a far larger reduction is required to produce a given structure by tensile elongation.

Tensile elongation has, therefore, been chosen as the process for working the specimens, which initially will be rods. The center section of each rod will be reduced in diameter to minimize its tendency to "hour glass"; however, the elongation cannot be expected to be completely uniform over the entire length. The rods may have to be centerless ground and the ground surface removed chemically. The deformation produced by tensile elongation will differ from that produced by tube drawing only in degree, not in nature.

3. Engineering Properties of Reactor Materials--Research and Development

a. High Temperature Mechanical Properties of Ceramic Fuels

(i) High-temperature Stress Relaxation of UO_2 (J. T. A. Roberts)

Last Reported: ANL-7553, pp. 92-93 (Feb 1969).

The stress-relaxation measurements reported previously were characterized by large amounts of relaxation occurring in very short times. For example, at temperatures $\geq 1500^\circ\text{C}$, 75% of the stress relaxed

(a)

(b)

(c)

(d)

Fig. III.B.4. The Structure of AISI Type 316 Stainless Steel Rod: (a) Elongated 14%; (b) Drawn 14%; (c) Elongated 14% and Heat Treated 6 hr at 1200°F; (d) Drawn 14% and Heat Treated 6 hr at 1200°F

in less than 10 min. It was decided that relaxation in the machine and tungsten tooling could make a significant contribution to the total measured relaxation. There was no direct method of determining this effect and, therefore, an indirect method was used. Specimens having two different grain sizes were tested under identical conditions. The applied load was allowed to relax over a period of 5 min before reloading to a higher load. The effect of increasing stress on the amount of relaxation, as a function of grain size, is presented in Fig. III.B.5. For both grain sizes there

Fig. III.B.5. Stress Relaxation as a Function of Initial Stress and Grain Size

occurred an initial linear region followed by a second region of larger slope. The initial region was determined to be due to relaxation of the machine and tooling, whereas the second region represented the relaxation of the specimen. A computer analysis was used to subtract relaxation effects in the tooling and load train, and to fit the resulting data to a stress-relaxation equation.

The stress σ_R at the intersection of the first two regions (see Fig. III.B.5) defines the stress at which stress relaxation of the specimen is first detected. With a decrease

in grain size and temperature, σ_R increases. Figure III.B.6 shows that the temperature dependences of σ_R and σ_y , the yield stress from conventional stress-strain tests (see ANL-7553), are approximately parallel, with slopes equal to 0.93 and 0.94, respectively. For the 5- μ grain-size material, the slopes were 2.75 and 2.3, respectively. The observation of stress relaxation at stress levels lower than those that would cause macroscopic yielding of the specimen agrees with results from low-temperature stress relaxation of titanium.*

Fig. III.B.6
Plot of σ_R and σ_y
vs Temperature

*Sargent, G., and Conrad, H., Scripta Metall. 3, 43 (1969).

One equation derived to describe stress relaxation* gives

$$\sigma_0 - \sigma = \frac{kT}{V} \log_{10} [1 + (t/c)], \quad (1)$$

Fig. III.B.7. ΔW vs Time

σ_{app} curves, one for $T \geq 1500^\circ\text{C}$, and the other for $T \leq 1300^\circ\text{C}$. The 1320 and 1360°C curves fell in between.

The strain-rate sensitivity m' [defined as $m' \equiv \left(\frac{\partial \ln \dot{\epsilon}}{\partial \ln \sigma} \right)_T = \sigma \left(\frac{\partial \ln \dot{\epsilon}}{\partial \sigma} \right)_T$] can be obtained[†] by rearranging Eq. (1):

$$m' = \frac{2.3 \sigma_0 V}{kT} + 1. \quad (2)$$

For the $17\text{-}\mu$ grain-size material, m' was approximately stress-independent at stresses approaching the ultimate tensile strength. Values of 20-33 were obtained at $T \leq 1360^\circ\text{C}$, while at $T \geq 1500^\circ\text{C}$, $m' = 11$ -14. The limited data for the $5\text{-}\mu$ material gave $m' = 25$ -30 in the low temperature range, and $m' \simeq 3$ at 1600°C .

These preliminary data support the idea of two different deformation processes in the temperature ranges 1100 - 1300°C and 1500 - 1800°C . In conventional four-point bend stress-strain tests semibrittle behavior was observed up to some transition temperature T_t ($\sim 1400^\circ\text{C}$ for $\dot{\epsilon} = 0.097/\text{hr}$, and independent of grain size). Beyond 1400°C a steady-state flowstress develops for large amounts of strain, and the strength decreases (see ANL-7553).

*Guin, F., and Pratt, P. L., *physica status solidi* **6**, 111 (1964).

**The word apparent is used to emphasize that the applied stresses are used in its evaluation.

[†]Hull, D., and Noble, F. W., *Acta Met.* **12**, 1089 (1964).

where σ_0 = initial stress at $t = 0$, σ = stress after time t , k = Boltzmann's constant, T = temperature, V = apparent activation volume,** and c = time constant. An example of the fit of this equation to corrected data is shown in Fig. III.B.7.

Activation volumes were strongly stress-dependent, decreasing with increasing applied stress from approximately $1000 b^3$ to approximately $100 b^3$ (values multiplied by two to convert to shear; b = magnitude of Bergers vector). Generally, the points fell on two V versus

In a material having few slip systems, extensive plastic deformation can occur by a grain-boundary mechanism, such as diffusional creep and/or grain-boundary sliding. The former requires complete Newtonian viscosity at the boundaries* ($\epsilon \propto \sigma$). It is possible, however, for grain-boundary sliding to be a deformation process that is not Newtonian viscous, i.e., $\dot{\epsilon} = k\sigma^{m'}$, $m' > 1$.** Such grain-boundary sliding must involve grain deformation, grain-boundary dislocations, or grain-boundary migration. The present results show a significant grain-size dependence of m' ; m' decreases in a similar proportion to the decrease in grain size. As m' approaches 1, diffusional grain-boundary creep must become more favorable. One can, therefore, envisage a grain-size dependence such that large-grained material deforms by dislocation-controlled grain-boundary sliding, whereas fine-grained material deforms by a diffusion-controlled grain-boundary mechanism. This possibility is being further investigated.

C. Engineering Development--Research and Development

1. Instrumentation and Control

a. Boiling Detector (T. T. Anderson)

(i) Acoustic Method

Last Reported: ANL-7595, pp. 103-105 (July 1969).

(a) Environmental Sound in EBR-II and CCTL (T. T. Anderson and F. H. Just)

Acoustic monitoring with externally mounted accelerometers on the EBR-II reactor (see Progress Report for January 1969, ANL-7548, pp. 99-100) and CCTL loop (see Progress Report for February 1969, ANL-7553, pp. 96-97) demonstrated the existence of impulse-type flow-induced sounds. An analysis of this type of sound, combining acoustic decay time with random occurrence of signals, gave insight into the relationship between frequency spectra and real-time signals. Analytical results were obtained for four types of expected pulses: (1) an exponentially decaying impulse, (2) an exponentially decaying sine-wave signal, (3) a single haversine pulse, and (4) a square pulse.

The pulse observed most often on the CCTL vessel was an exponentially decaying sine wave, corresponding to a reverberant decay of sound energy with a time constant of 10-20 ms. For sound decay according to the Sabine theory, this relatively long time constant corresponds to an energy-absorption coefficient for tank walls of only 2-4%. These results

*Often termed Newtonian viscous grain-boundary sliding.

**Heuer, A. H., Cannon, R. M., Jr., and Tighe, N. J., paper presented at 15th Sagamore Army Materials Research Conf. on Ultrafine Ceramics, Sagamore Lake, New York, August 23, 1968, to be published.

were verified by measurements of reverberation decay and sound levels within an 1.68-m-dia by 3.35-m-high water-filled tank, the construction of which is similar to that of the CCTL vessel.

(b) Sodium-flow Noise

Last Reported: ANL-7581, p. 102 (June 1969).

Cavitation-like sounds (see Progress Report for March 1969, ANL-7561, pp. 75-76) have been detected in two large sodium-coolant systems, the EBR-II reactor and the CCTL loop. Because this flow-related background noise could mask noises caused by thermally induced boiling of sodium, a review was made of feasible acoustic-detection methods for flowing-coolant systems. From the review it was concluded that, if acoustic boiling detection is to be applied to any coolant system, it might be necessary to design the flow system for minimum hydrodynamically induced noise, and to use an acoustic-detection system capable of determining the location of core-induced noise and of rejecting extraneous noises.

(c) Irradiation and Resistance Tests of Piezoelectric and Insulator Materials (T. T. Anderson, S. L. Halverson, and W. Primak)

Last Reported: ANL-7595, pp. 103-105 (July 1969).

Thermal-annealing tests have been performed on disks of lithium niobate piezoelectric crystals that had been irradiated with electrons. In the annealing tests, the piezoelectric crystals were immersed in a thermostatically controlled oil bath for 20-min intervals. Electrical measurements before and after each annealing showed no changes. Effects of annealing could be observed optically.

A new impedance-measuring system for determining transducer characteristics was out of calibration when received, so it was returned to the vendor.

b. Vibration Sensor (T. P. Mulcahey)

Not previously reported.

LMFBR instrumentation is being developed to measure fuel motion, vibration of the core structure, control-element flutter, and acoustic vibrations due to boiling that might disturb overall reactor stability. Emphasis is on a vibration transducer (accelerometer or low-frequency sound transducer) for in-core use. Out-of-pile tests determine material compatibility in sodium and temperature effects; in-pile tests will determine the additional influence of radiation on the detectors.

In a test facility that has been assembled and is being checked out, the input is a vibration table that can excite a transducer at frequencies of 5 Hz to 20 kHz at up to 80 g, and the output is processed by a spectrum analyzer.

After the literature about commercially available high-temperature vibration sensors was surveyed, specifications were prepared for procuring 1200°F vibration transducers for both out-of-sodium and in-sodium environments. Three 1150°F nonimmersible transducers have been procured and tested; three additional units with special cable connectors have been purchased for further tests. Three immersible 1150°F transducers have been purchased for testing in sodium.

(i) Out-of-pile Tests of Nonsodium-immersible Commercial Transducers. Two nonimmersible 1150°F accelerometers with soft cables were procured from Gulton Industries. After calibration tests on a shaker table determined the frequency response of the transducers for a spectrum range of 100-1000 Hz at temperatures up to 700°F, these transducers were installed on the outside of the Core Components Test Loop (CCTL), where they were tested in a high-temperature environment while operationally monitoring the loop. Some of the information from these CCTL tests has been reported in the boiling-detector program (see Progress Reports for February 1969, ANL-7553, p. 96, and March 1969, ANL-7561, pp. 75-76) and in the CCTL program (see Progress Reports for February 1969, ANL-7553, p. 28, and April-May 1969, ANL-7577, pp. 53-54).

The satisfactory experience with these two transducers led to the purchase of three additional 1150°F accelerometers, whose cables are attached by special connectors, from the same vendor for further tests.

(ii) Out-of-pile Tests of Sodium-immersible Commercial Transducers. After procurement of 1200°F in-sodium transducers from one manufacturer was terminated because he could not meet the service-temperature requirements, three 1150°F immersible accelerometers were purchased from Gulton Industries. These units have integral cables that have mineral insulation and stainless steel sheaths. They will be tested immersed in sodium.

2. Heat Transfer and Fluid Flow

a. Liquid-Metal Heat Transfer in Pin Bundles (T. Ginsberg)

Not previously reported.

The literature is being reviewed to determine the state of the art in calculational techniques for evaluating heat-transfer conditions in rod

bundles. A recent survey[†] covered much of the pertinent information. However, the intent in this work is to determine whether analytical techniques are developed sufficiently to give local subchannel heat-transfer information pertinent to the evaluation of: (1) rod bowing, swelling, and displacement, (2) subchannel flow blockage, and (3) flow maldistribution. The problem of resulting thermal stresses in a fuel pin will be considered.

3. Engineering Mechanics

a. Structural Dynamics--Structure-Fluid Dynamics (M. W. Wambsganss, Jr.)

(i) Mathematical Modeling of Parallel-flow-induced Vibration (M. W. Wambsganss, Jr.)

Last Reported: ANL-7595, pp. 106-107 (July 1969).

Solution of the mathematical model describing vibration amplitude requires, among other things, knowledge of system damping and a mathematical characterization of the random forcing function.^{††} Neither of these features can be described theoretically with state-of-the-art technology. To interpret measured rms-displacement data from single-element flowtests, results from an experimental study of the flow-velocity-dependence of damping (see Progress Reports for April-May 1969, ANL-7577, p. 166, and June 1969, ANL-7581, pp. 105-107) and a phenomenological model for the random pressure field[‡] are used.

An approach taken by Reavis,^{‡‡} noting that the first bending-mode response is dominant, and with the assumption that the mean-square-value spectral density of the pressure is "flat" over the frequency range of interest and that the rms pressure is proportional to the mean flow velocity squared, leads to an expression of the form

$$y^*(\bar{u}, l/2) = FC \frac{(\bar{u})^2}{[\bar{\zeta}(\bar{u})]^{1/2}} \quad (1)$$

for the nondimensional rms-displacement of a cylindrical element in parallel flow, where F is a turbulence flow factor, C is a function of system parameters, $\bar{u}$ is mean axial flow velocity, and $\bar{\zeta}(\bar{u})$ is the functional dependence of damping on mean flow velocity. The experimental results from the study of damping have been correlated with the expression

$$\bar{\zeta}(\bar{u}) = 1 + 0.044 \bar{u}. \quad (2)$$

[†]Heat Transfer in Rod Bundles, Symposium at Winter Annual Meeting, ASME, New York (December 3, 1968).

^{††}Wambsganss, M. W., Jr., and Boers, B. L., Parallel-Flow-Induced Vibration of a Cylindrical Rod, ASME Paper No. 68-WA/NE-15, presented at Winter Annual Meeting of ASME, New York, N.Y., December 1968.

[‡]Corcos, G. M., Resolution of Pressure in Turbulence, J. Acoust. Soc. Am. 35, 192-199 (1963).

^{‡‡}Reavis, J. R., WVI-Westinghouse Vibration Correlation for Maximum Fuel Element Displacement in Parallel Turbulent Flow, Trans. Am. Nucl. Soc. 10(1), 369-370 (June 1967).

Substitution of Eq. (2) into Eq. (1) yields

$$\frac{y^*(\bar{u}, l/2)}{FC} = f(\bar{u}) = \frac{(\bar{u})^2}{(1 + 0.044 \bar{u})^{1/2}} \quad (3)$$

Results from several flowtests in which the displacement-time histories of the test element were recorded showed that, over the range of flow velocities tested, the rms displacement increased with mean axial flow velocity raised to a power (for example, see Progress Report for November 1968, ANL-7518, pp. 101-102); for the different tests the exponent varied from 1.65 to 2.0.

Figure III.C.1 shows the effect of the mean axial flow velocity, in four different flowtests, on the rms displacement normalized by the constant FC that best correlates the data with the functional relationship given by Eq. (3). The experimental points agree well with the solid curve representing $f(\bar{u})$ in Eq. (3) over the entire range of test flow velocities, which implies that the dependence of damping on mean axial flow velocity might explain the variance of the exponent in the power-function relationship between rms displacement and mean flow velocity.

Fig. III.C.1
Rms Displacement as a Function
of Mean Axial Flow Velocity

(ii) Preparation of Two Structural-Dynamic Test Loops
(B. L. Boers and P. L. Zaleski)

Last Reported: ANL-7606, pp. 110-111 (Aug 1969).

Construction of the large test loop has started and is proceeding on schedule.

Spectrum analyses of pressure-time histories measured at several piping-wall locations on the small test loop show acoustic disturbances at frequencies corresponding to the running speed of the pump, vane-passing of the impeller, and their higher harmonics. An effort is being made to attenuate these disturbances by using various filtering systems.

(iii) Damping (P. L. Zaleski and M. W. Wambsganss, Jr.)

Last Reported: ANL-7606, p. 111 (Aug 1969).

In the study of the effect of fluid viscosity on effective damping, viscosity has been varied from 1.4 to 2 lb/ft-hr. The experiments show that damping increases with viscosity in a manner that might be linear. To verify this, additional data are being obtained.

(iv) Effect of Initial Curvature on Stability of Tube in Parallel Flow (S. S. Chen)

Last Reported: ANL-7577, pp. 166-169 (April-May 1969).

If a tube submerged in a stationary fluid is subjected to the action of lateral force, a small initial curvature of the tube has no effect on the bending stress. However, if the fluid is moving, the deflection and bending stress will be influenced substantially by initial curvature. For a simply supported tube, the initial shape of the axis of the tube is given by a series

$$y_0 = \sum_n a_n \sin \frac{n\pi x}{l}.$$

The tube in parallel flow is subjected to the action of a longitudinal compressive force μu^2 , where u is the fluid transport velocity and m is added mass. The total deflection is

$$y = \sum_n \frac{a_n}{1 - r_n^2} \sin \frac{n\pi x}{l},$$

where r_n is the ratio of fluid transport velocity to its critical value corresponding to the n^{th} mode of buckling. It is seen that the initial deflection is magnified in the ratio of $1/(1 - r_n^2)$.

Fig. III.C.2. Response Curves for Displacements of Tube with Small Initial Curvature as Affected by Unsteady Fluid Transport Velocities

If the fluid transport velocity is not steady, $u = u_0(1 + \mu \cos \omega t)$. The small initial curvature of the tube does not affect the instability regions (see ANL-7577), but the initial curvature is much magnified. As a numerical example, for a simply supported tube with $y_0 = a_0 \sin(\pi x/\ell)$ and having natural frequency of ω_1 for $u = 0$, Fig. III.C.2 shows the steady-state response curves for the case where the damping factor ζ_1 is 0.01, the excitation parameter μ is 0.2, and the magnification factor is defined as the maximum deflection of the tube to its initial deflection.

These response curves are valid near the second instability region, which demonstrates the effect of initial curvature under pulsating flow. Although nonlinear theory shows

the response in the unstable regions more realistically, it is important from the practical viewpoint that the linear theory as discussed here be useful in designing to avoid the onset of instability.

(v) Preparation of Tube-bank Test Section for Use in Crossflow Studies (H. Halle)

Design of a tube-bank test section for use in a study of crossflow-induced vibrations has begun. Initial emphasis is on an instrumented spring-mounted test element.

D. Chemistry and Chemical Separations

1. Fuel Cycle Technology--Research and Development

a. Fluoride Volatility Processes

(i) Engineering-scale Development for FBR Fuels (N. M. Levitz)

Last Reported: ANL-7595, p. 112 (July 1969).

(a) Alpha Facility Maintenance. Removal of contaminated equipment from the large glovebox in the Engineering-Scale Alpha Facility

(which had been used in fluoride volatility development work) is nearly complete. Decontamination of the interior of the large glovebox and equipment removal from the small glovebox have begun. A newly developed packaging technique (direct packaging in strong fiber cartons rather than in plastic bags) is being used that is both safer and faster than ordinary bagging methods for plutonium-contaminated materials. As a result of advance planning and careful execution of the work, complete containment of the radioactivity (no release to the work area) has been achieved.

b. Molten Metal Decladding

(i) Engineering Design, Analysis, and Evaluation (N. M. Levitz)

Not previously reported.

A conceptual design study of a liquid-metal-decladding plant capable of processing 5 tonnes/day of stainless steel-clad LMFBR oxide fuel is in progress. The decladding procedure involves direct dissolution of the cladding in a liquid metal and separation of the resulting solution from the fuel oxides. This decladding method is an alternative to sodium removal, disassembly, and shearing operations required for mechanical-decladding schemes.

The principal process objectives are separation of stainless steel cladding from the fuel, containment of released fission product volatiles, and the preparation of a compatible feed for further processing in an aqueous plant. The reference fuel is an Atomics International design.* Preliminary layouts of cell equipment have been prepared, material and heat balances have been made, a tentative operational scheme has been developed, and a conceptual design of the graphite decladder vessel and furnace is being prepared. Zinc, which is known to dissolve stainless steel at practical rates (see Progress Report for April-May 1969, ANL-7577, pp. 175-176), has been selected as the solvent metal.** Zinc is also a suitable solvent for zirconium alloy; previous experiments (see Progress Report for September 1968, ANL-7500, pp. 117-118) have shown that Zircaloy-2 dissolves in zinc at a rate about one order of magnitude higher than the dissolution rate of Type 304 stainless steel. Therefore, this decladding technique might also be applicable to thermal reactor fuels.

In the conceptual design study, a decladding vessel, located in a process cell with an inert gas atmosphere, will contain enough zinc to declad 5 tonnes of fuel subassemblies. In the proposed process, fuel assemblies are removed from a sodium-filled shipping cask with a gas-cooled grapple-turret (container) device, and the sodium is allowed to drain from

*Dieckamp, H., and Roddis, L. H., Jr., The GPU/AI Demonstration Plant, Nucl. News. 11(7), 41 (July 1968).

**An alternative solvent for decladding stainless steel is antimony-copper (see Progress Report for June 1969, ANL-7581, pp. 109-110).

the assemblies into the shipping cask. As the next step, either the turret may be moved to a cutting station where the bottom hardware of the sub-assemblies is sawed off for disposal as waste, or sawing may be omitted and all steel components of the assembly subsequently dissolved in the liquid zinc. The turret is next positioned on the decladder vessel, and the fuel assembly is lowered into a prepositioned basket and into the mildly agitated molten zinc. When stainless steel dissolution is completed, the basket containing the fuel (which remains virtually unattacked by the zinc) is lifted out of the zinc-stainless steel solution, the zinc is allowed to drain, and the fuel is fed to the aqueous process. This cycle, estimated to require 6 hr, is performed four times in a given decladding vessel, after which that vessel is emptied and cleaned while a second decladding vessel is on-stream.

Provision is made for recovery of oxide fines that escape through the basket holes into the zinc solution. After four decladding cycles, vigorous mixing of the zinc solution with molten salt (e.g., CaF_2) overlying the zinc transfers the oxide fines to the salt; the salt is further processed to recover the contained fuel.

The decladder is closed to the cell atmosphere during decladding, and the volatile fission products (Kr, Xe, I, and ^3H), released when the fuel subassemblies are first broached, are removed in a stream of purge gas. The final zinc solution, which contains any sodium originally adhering to the subassemblies and ≤ 15 wt % stainless steel, may be cast into rods for storage as waste after this solution has been monitored to verify that it does not contain actinides. The salt, which is expected to contain little fission products, is recycled to the decladding operation.

Criticality was also considered in this preliminary evaluation. A decladder design utilizing mass control rather than geometry control was selected. A major factor preventing criticality is the limitation of the batch size to three assemblies per basket, since double batching of this batch size would not result in a critical mass. Other events that could affect criticality were considered: basket rupture, large-scale migration of graphite fines into the basket or of fuel fines out of the basket, and undetected accumulation of fuel fines and graphite particles in the salt. The safety criteria include the double-contingency principle, which states that two independent and very unlikely events that might cause criticality do not occur simultaneously.

Since the degree of subdivision of irradiated fuel will strongly influence the design of the decladder, existing data will be surveyed to determine size distribution of high-burnup UO_2 -20 wt % PuO_2 . Techniques for achieving accountability control of plutonium are being investigated.

(ii) Engineering Development (R. D. Pierce)

Last Reported: ANL-7606, pp. 113-115 (Aug 1969).

(a) Materials Evaluation. Corrosion studies are being made to identify materials of construction suitable for vessels containing molten metals and salts. In one test, samples were prepared by brazing two 0.1-in.-thick tungsten plates (sandwich fashion) with Coast Metals 60 alloy (Ni-20% Cr-10% Si-3% Fe) and then diffusion treating at 1370°C for 8.5 hr. The samples were mounted as agitator blades, and tested in molten zinc and molten CaCl_2 -20 mol % CaF_2 at 130 rpm for 144 hr at 800°C. The blades appeared to be in excellent condition after the test, indicating that the brazing procedure has a potential for tungsten vessel fabrication.

(iii) Process Demonstration Experiments (R. D. Pierce)

Last Reported: ANL-7595, pp. 112-113 (July 1969).

A recent change in the objectives of the program has resulted in the redirection of effort from the development of a flowsheet for the salt-transport process to the development of a head-end process for LMFBF fuels. Until the experimental and equipment needs of the head-end development work are defined, efforts on the construction of helium-atmosphere alpha gloveboxes are being concentrated on completion of glovebox 3 and its auxiliary equipment.

Leak checking of box 3 and performance testing of the purification system were completed. Both the dryer and the oxygen absorber of the purification system performed well. The automatic pressure-control system for box 3 was tested and operated satisfactorily. The piping manifolds for the furnaces for box 3 are essentially complete and are to be installed under the box soon. The side panels for the area under box 3 have been installed.

The three Mo-30 wt % W transfer lines have been fabricated and are being wound with resistance heaters. Installation of control panels and refrigeration controls is being completed. In the cupolas on the roof of box 3 two storage baskets were installed.

A test was performed to determine whether failures of the primary containment crucible and the furnace will that would cause molten Mg-Cu-Pu alloy and molten salt to leak into the area under box 3 would result in a fire hazard. Molten salt (MgCl_2 -30 mol % NaCl-20 mol % KCl) and Mg-40 at. % Cu at 800°C were poured, in turn, on a stainless steel tray, onto a painted concrete floor, and on Fiberfrax. No oxidation or flames were observed upon pouring into the tray; pouring onto the concrete floor resulted in scorching and the evolution of fumes;

contact of molten metal with Fiberfrax resulted in blue and white flames, probably due to reaction of the metal with water in the Fiberfrax. On the basis of these results, it is believed that the proposed damper on the air inlet to the underside of the glovebox, which is designed to close automatically in the event of a metal spill, will be effective in preventing oxidation and the spread of contamination.

(iv) Behavior of Volatile Fission Products (M. J. Steindler)

Last Reported: ANL-7606, pp. 115-116 (Aug 1969).

(a) Laboratory Process Development. The relatively large mass and radioactivity of fission product iodine present in 30-day-cooled LMFBR oxide fuel necessitates that the distribution of iodine in the fuel pins be determined so that the design of fuel-processing steps will take into consideration the quantity of iodine in gaseous and other process streams. The fraction of iodine in the fuel-pin plenum was estimated to be 50 to 70% of the fission product iodine formed, on the basis of an evaluation of data in the literature. Therefore, decladding steps must be designed to handle this quantity of gaseous iodine.

2. General Chemistry and Chemical Engineering--Research and Development

a. Thermophysical Properties

(i) Partial Pressures of Vapor Species in the U-Pu-O System and in the U-Pu-O System Containing Fission Products
(J. E. Battles and P. E. Blackburn)

Last Reported: ANL-7581, pp. 112-114 (June 1969).

Mass-spectrometric studies of the volatilization behavior of the uranium-plutonium-oxygen system are being made to determine (1) the composition of the vapor phase in equilibrium with the condensed phase(s), (2) the partial pressures of the vapor species as a function of temperature, and (3) the thermodynamic properties of the vapor species.

Additional mass-spectrometric measurements were made to determine the enthalpies of reaction for the reactions

and

primarily to supplement the limited data previously obtained for Reaction 3 (see Progress Report for October 1968, ANL-7513, pp. 127-130).

The results from the latest experiments (Runs PR-8B, -8C, and -9A), as well as results from previous experiments are summarized in Table III.D.1. The heats of reaction given in the last three columns relate to condensed materials with O/Pu atom ratios of 1.54 to 1.56. The temperature dependency of the vapor species in these latest experiments exhibited linear behavior and was consistent with the data reported in ANL-7513, whereas anomalous behavior was observed in experiments reported in the Progress Report for March 1969, ANL-7561 (pp. 89-91). The anomalous behavior observed in those experiments may be attributed to a transition from a two-phase to a single-phase condensed material.

TABLE III.D.1. Heats of Reaction to Produce Gaseous Pu, PuO, and PuO₂ from Two-phase (Pu₂O₃ + PuO_{1.61})

Run No.	Cell Material	Temperature Range ^a (°K)	Composition of Solid Material (O/Pu)	Heat of Reaction ^b (kcal/mol)		
				Pu	PuO	PuO ₂
PW-5A	W	1704-2145	1.55	108.3 ± 1.5	116.4 ± 0.9	-
PR-5A	Re	1797-2134	1.60 ^c	111.9 ± 1.6	114.4 ± 0.8	-
PR-6A	Re	1685-2147	1.56	111.2 ± 1.3	120.9 ± 1.1	-
PR-6C	Re	1657-2148	1.54	111.5 ± 2.8	119.1 ± 0.7	-
PR-6D	Re	1735-2147	1.56	109.9 ± 1.8	118.8 ± 0.6	126.2 ± 4.9
PR-8B	Re	1648-2224	1.55 ₀	113.1 ± 1.0	120.6 ± 0.5	136.3 ± 2.1
PR-8C	Re	1803-2212	1.54 ₇	109.1 ± 1.5	120.7 ± 1.0	131.7 ± 6.3
PR-9A	Re	1812-2216	1.56 ₁	105.2 ± 1.0	116.6 ± 0.7	133.8 ± 2.0
Average				110.0 ± 1.7	118.4 ± 1.0	132.0 ± 4.4

^aThe measurements for PuO₂⁺ covered the temperature range 1950°K to the maximum listed.

^bThe errors listed are standard deviations; an ionizing electron energy of 11 eV was used for measurements of Pu⁺ and PuO⁺, and eV for measurements of PuO₂⁺.

^cAnalysis is believed to be incorrect.

As previously reported (see ANL-7581), two preliminary experiments on the volatilization of 20 wt % PuO₂-80 wt % UO₂ material at ~2260°K have been conducted, one using a tungsten effusion cell and the other a rhenium cell. Two problems were encountered in these experiments: (1) the starting material, which had an oxygen-to-metal (O/M) atom ratio of 2.01₃, reacted with the tungsten and rhenium effusion cells to form vapor species of W-O and Re-O, respectively. (2) The concentration of PuO₂(g) in the vapor phase could not be evaluated because the mass spectrometer (Bendix T-O-F) does not completely differentiate between ²³⁹PuO₂⁺ and ²³⁸UO₂⁺. Reaction with the cell material will be eliminated by using iridium effusion cells, which have been shown to be nonreactive with plutonia (see Progress Report for April 1968, ANL-7445, pp. 124-127). The second problem can be eliminated by substituting ²³⁵U for ²³⁸U. The Metallurgy Division has prepared batches (~25 g each) of 5 wt % PuO₂-95 wt % UO₂, 10 wt % PuO₂-90 wt % UO₂, and 20 wt % PuO₂-80 wt % UO₂ material using enriched uranium. The isotopic composition of the uranium is 93.2 wt % ²³⁵U, 5.9 wt % ²³⁸U; the balance is ²³⁴U and ²³⁶U. The isotopic composition of the plutonium is 86.1 wt % ²³⁹Pu, 11.3 wt % ²⁴⁰Pu, and 2.4 wt % ²⁴¹Pu.

(ii) Phase Diagram of the U-Pu-O System Containing Fission Products (Nonradioactive) (H. Chen and P. E. Blackburn).

Not previously reported.

Among the many factors possibly related to fuel failure in reactors are chemical interactions among the fuel, its fission products, and the cladding material. The purpose of this study is to obtain for these chemical processes thermodynamic data which might be useful in designing a fuel system having a suitably long lifetime.

Analyses of irradiated fuel suggest that cesium and its compounds, which can penetrate the grain boundary of stainless steel, have a close relation with the corrosion of the cladding. The Cs-U-O ternary system, therefore, will be investigated with an emphasis on the variation of oxygen activities with composition and temperature. The experimental scheme calls for the use of the Knudsen effusion technique and a micro-balance in conjunction with a mass spectrometer. The oxygen potential will be determined either from the vapor pressure of cesium, using the Gibbs-Duhem equation, or from the equilibrium constant of the dissociation of cesium oxide in the vapor phase. The work chamber, which houses the balance and the mass spectrometer, will open to a dry box so that samples can be handled under a controlled atmosphere. All parts of the equipment except the mass spectrometer are ready for assembly and testing.

In connection with the interaction of stainless steel and cesium compounds, the possibility of a chemical transport of iron by iodine (Van Arkel process) is being examined by transport experiments and thermodynamic analyses. The results indicate that in the presence of Cs_2O , which is much less stable than CsI , there is no free iodine available for the transport of iron. However, it was observed experimentally that at about 900°C free iodine was released from $\text{CsI} + \text{UO}_{2+x}$ when $x \geq 0.18$ (the lowest value of x yet studied). The released iodine attacked stainless steel extensively, but did not transport iron efficiently unless the remaining unreacted CsI and UO_{2+x} was removed. Without transport, an unidentified white deposit was observed.

A plausible explanation for these observations is as follows. Formation of a ternary Cs-U-O compound, which is more stable than Cs_2O , could cause the release of iodine from CsI . Under a relatively high oxygen potential, the iodine will attack stainless steel and form an Fe-O-I compound instead of FeI_2 , which is essential in the transport of iron. These results point out the necessity to understand further the Cs-U-O system and, in particular, the thermodynamic properties of its ternary compounds. To prevent corrosion of the cladding by fission product iodine, an oxygen content in the fuel must be found that will make cesium available to iodine for the formation of CsI , which is not corrosive chemically to stainless steel.

More experiments are being planned to identify the new compounds formed and to determine their thermodynamic properties.

b. Preparation of Nuclear Materials (A. A. Jonke)

Last Reported: ANL-7581, p. 114 (June 1969).

A 25-kW, inductively heated plasma torch is being used to convert mixtures of uranium oxide and carbon to UC. Analysis of products from eight earlier runs (see ANL-7581) indicated a high oxygen content of the product, suggesting that the hot particles emanating from the torch might have reacted with atmospheric oxygen in the reaction chamber. The reaction chamber has been modified so that oxygen may be excluded. Prior to each run, the chamber is evacuated, then filled with argon gas. Three runs have been made with the modified apparatus, and samples of the products are being analyzed.

c. Liquid Metal-Salt Technology (I. Johnson)

(i) Liquid Metal-Salt Processes

Last Reported: ANL-7606, p. 116 (Aug 1969).

(a) Solubility of Zirconium in Liquid Cu-Mg-Ca Alloy.

The solubility of zirconium in the liquid Cu-Mg-Ca alloy that would be used in the reduction of fuel oxide to the metallic form in the pyrochemical head-end step was measured at 550-852°C. Analytical data indicate that below 770°C, a zirconium-copper intermetallic compound with the approximate composition ZrCu_2 is in equilibrium with liquid alloy containing ~0.6 atom fraction magnesium and ~0.4 atom fraction copper. It was found that when calcium metal is added to the system at constant temperature, the solubility of zirconium decreases and the copper-to-magnesium ratio increases slightly. The logarithm of the zirconium solubility is a linear function of the calcium content of the liquid phase.

(b) Behavior of Tellurium. Since radioactive tellurium undergoes beta decay to radioactive iodine, for which aqueous solvent-extraction processes have a limited capacity, it is necessary to know the behavior of tellurium in proposed pyrochemical head-end steps. In the present study, metallic tellurium was added to a tantalum crucible containing 278 g of CaCl_2 -20 mol % CaF_2 and 758 g of Mg-49 at. % Cu at 854°C; the system was stirred for 12 hr and allowed to settle. Filtered samples of each phase were analyzed for tellurium. The distribution coefficient (wt % Te in salt/wt % Te in metal) was about 260. This result, if confirmed, would indicate that by use of a pyrochemical reduction step as a head-end step for an aqueous solvent extraction plant, tellurium would be separated from the uranium- and plutonium-bearing metal stream early in the process.

As a result, the quantity of radioactive iodine (see ANL-7606, pp. 115-116) that would have to be handled in the separation stages of the process would be reduced significantly.

3. Sodium Chemistry--Research and Development

a. Fundamental Properties and Radioactive Contaminant Behavior

Items under this category are now reported under Sect. I.C.

PUBLICATIONS

The Use of the Conducting Mask in Pulsed Electromagnetic Testing

C. W. Cox* and C. J. Renken

ANL-7172 (April 1969)

A Demountable, Stainless Steel-to-Aluminum Transition Seal for Ultrahigh Vacuum

P. M. Danielson

Vacuum 19, 365 (Aug 1969) Letter

Transient System for Measurement of Thermal Properties of Nuclear Fuel Powders of Varying Densities

N. H. Schilmoeller and D. E. White

Proc. 8th Conf. on Thermal Conductivity, Purdue Univ., October 7-10, 1968. Plenum Press, New York, pp. 857-870

Fast-Neutron Cross Sections of Hafnium, Gadolinium, and Samarium

G. L. Sherwood, A. B. Smith, and J. F. Whalen

ANL-7567 (May 1969)

Rapid and Specific Extraction of Aluminum from Complex Mixtures with N-Benzoyl-N-Phenylhydroxylamine Spectrophotometric Determination with 8-Quinolinol

R. Villarreal, J. R. Krsul, and S. A. Barker

Anal. Chem. 41, 1420-1422 (Sept 1969)

*South Dakota School of Mines.

IV. NUCLEAR SAFETY RESEARCH AND DEVELOPMENT

A. LMFBR Safety--Research and Development

1. Accident Analysis and Safety Evaluation

a. Parametric and Sensitivity Analysis and Code Intercomparison Studies (G. J. Fischer)

Not previously reported.

The Fuel Deformation module of the fast reactor accident study code SAS1A evaluates the fuel-pin behavior during an accident situation. Since this evaluation requires the knowledge of thermophysical and mechanical properties of both fuel and cladding which are rarely known for excursion conditions, a sensitivity study for this module has been undertaken. Results to be discussed below are based on a recent experiment* (S-3) on fuel failure behavior during transient meltdown in a sodium-filled piston autoclave (see Progress Report for August 1969, ANL-7606, pp. 123-124).

Two sets of calculations using different values for change in volume due to fuel melting have been performed. In the calculations performed for the assumed standard case, the molten-fuel density of 93% theoretical density (TD) pellets was taken to be the same as the density of 100% TD molten fuel. The resulting increase in fuel volume due to phase change, upon melting and filling the void spaces, therefore, amounts to 5.8% instead of 13.8% for 100% TD fuel. This large difference in volume change due to fuel melting is due to the neglect of volume occupied by pores and voids in the solid fuel. The calculated plastic deformation of the clad for S-3 experiment is shown by curve 1 in Fig. IV.A.1. In this figure, the mass fraction of molten uranium dioxide is also shown. These calculations have been repeated by using a larger change in volume upon melting for uranium dioxide. If only half of the pores and voids which existed in the solid fuel are assumed to be filled due to fuel melting, the change in fuel volume due to melting is calculated to be 9.7%. For this case, the calculated plastic strain is shown by curve 2 in Fig. IV.A.1. It is seen that the mass fraction of molten fuel is unchanged since the rate of heat production and loss is not changed. As expected, both curves 1 and 2 are identical until uranium dioxide begins to melt (around 1030 J/g- UO_2). It is evident that a larger change in fuel volume due to melting results in an increased plastic strain of cladding.

*Ivins, R. O., Pressure Generation due to Violent Meltdown, ANL-7553, p. 113 (February 1969).

Fig. IV.A.1

Calculated Plastic Deformation of the Clad and the Fraction of UO₂ Melted As a Function of the Energy Produced in the Fuel for S-3 Experiment. Curve 1 corresponds to the base case (5.8% $\Delta V/V$ due to fuel melting for 93% TD UO₂) and curve 2 corresponds to the assumed 9.7% $\Delta V/V$.

To illustrate further the consequences of volume change due to melting, we find that 5% plastic deformation of the clad is achieved with an energy release of 1570 J/g-UO₂ for the second case, instead of 1740 J/g-UO₂ for the standard case. Alternatively, if it is assumed that molten fuel density is unaffected by the initial porosity of the fuel and that it is equal to the molten fuel density of 100% TD uranium dioxide, the results shown in Fig. IV.A.1 can be interpreted as due to different initial porosity of the fuel. In other words, curve 1 is the calculated plastic strain for 93% TD fuel while curve 2 corresponds to 96.5% TD fuel provided all other parameters are unchanged.

Retaining the explicit temperature dependence of clad yield strength, strength values are increased by a factor of two from the base values as a second sensitivity test. There appears to be no significant change in the plastic deformation except for a small delay in the initiation of plastic flow. This observation is attributed to the proportionately small role of elastic deformation played in the rapid transient under consideration. It is anticipated that such a variation in yield strength values would be more important for slower transients. No efforts were made to study the sensitivity of clad deformation with respect to change in explicit temperature dependence of the strength values. In all of these calculations the values of high-temperature yield strength were obtained by a linear extrapolation.

Finally, the effect of varying bond conductance on cladding behavior is studied. In the heat-transfer module of the code the bond conductance is expressed as

$$h_b = c/\Delta r, \quad (1)$$

where c is a constant and Δr is the existing clearance between fuel and clad. The bond conductance h_b , as given by Eq. (1), is restricted to a maximum input value of $h_{b \max}$. Theoretically, $h_{b \max}$ is the contact conductance. Neither the constant c nor $h_{b \max}$ is precisely known under the experimental conditions. For this reason we have carried on a sensitivity study in which only $h_{b \max}$ value is varied. The value of the constant c was taken to be 4.2×10^{-3} W/cm-°C, while $h_{b \max}$ was varied

Fig. IV.A.2

A Plot of Maximum Cladding Temperature and the Plastic Strain of the Clad for Two Different Values of the Contact Conductance. Vertical lines indicate the corresponding energy produced in the pin. Curve 1 shows results obtained for the base case ($h_{b \max} = 2 \text{ W/cm}^2\text{-}^\circ\text{C}$) while curve 2 is for $h_{b \max} = 10 \text{ W/cm}^2\text{-}^\circ\text{C}$.

This observation is once again attributed to the model used for calculations of plastic strain.

from $2 \text{ W/cm}^2\text{-}^\circ\text{C}$ (corresponding to the standard case) to $10 \text{ W/cm}^2\text{-}^\circ\text{C}$. This change in bond conductance (or contact conductance) is directly reflected in the cladding temperatures as shown in Fig. IV.A.2. This figure is a plot of the maximum clad temperature as a function of the plastic deformation of the clad. It should be noted that the values given do not necessarily correspond to the same axial position; they are hot spot values. The corresponding energy release is shown by almost vertical lines. Curve 1 in this figure corresponds to the base case; curve 2 is the result obtained with a value for the maximum bond conductance that is five times larger. It is seen that the cladding temperatures are directly affected by the variation in $h_{b \max}$. Constant energy-release lines, which are almost vertical, once again ascertain practically no significant change in the plastic strain due to change in yield strengths that result from variation of clad temperature.

2. Reactor Control and Stability

a. Reactor Stability and Stability Design Criteria for Spatially Dependent Systems (C. Hsu)

Last Reported: ANL-7595, p. 116 (July 1969).

(i) Nonlinear Model. Preliminary investigations of the quasi-linearization iterative procedure for obtaining solutions to nonlinear, one-dimensional, steady-state reactor equations have been completed. In addition to the types of problems described previously, the computational procedure has been used successfully to obtain the solutions $\phi(x)$ and α in the problem where the parameter α simulates control-rod position according to

$$\frac{\partial}{\partial x} D(x) \frac{\partial \phi(x)}{\partial x} + \Sigma_1(x) \phi(x) + p(x - \alpha) \Sigma_2(x) \phi(x) + f(x, \phi, T_f, T_c) = 0,$$

where $p(x - \alpha) = 0$ when $x < \alpha$, or 1 when $x \geq \alpha$, and $f(x, \phi, T_f, T_c)$ represents some nonlinear function of the flux ϕ , fuel temperature T_f , and coolant temperature T_c . This computational procedure will be used as a tool in the investigation of control-rod and stability aspects of LMFBR nuclear safety analysis.

3. Coolant Dynamics

a. Sodium Superheat (R. E. Holtz and R. M. Singer)

Last Reported: ANL-7581, pp. 121-122 (June 1969).

(i) Boiling from Surface Cavities

(a) Data from Sodium-expulsion Experiment. Figure IV.A.3 shows sodium-superheat data from the sodium-expulsion experiment during high heat-flux operation. Despite the considerable scatter, these data agree excellently with the theoretical curve generated from the pressure-temperature model, as do the data obtained from the sodium-superheat experiments at much lower heat fluxes; therefore, the heat flux probably has an insignificant effect on incipient boiling superheats in pool boiling when little inert gas is in the surface cavities.

Fig. IV.A.3

Sodium-superheat Data during Operation of Sodium Expulsion Experiment at High Heat Fluxes. Solid data points were taken at heat flux q_w of 124.5 W/cm^2 and with initial temperature T_0 at start of test run of $569 \pm 21^\circ\text{C}$, open-symbols were for $q_w = 63.7 \text{ W/cm}^2$ and $T_0 = 588 \pm 19^\circ\text{C}$ (when saturation temperature T_s was approximately 850°C) and $T_0 = 581 \pm 9^\circ\text{C}$ when $T_s \approx 820^\circ\text{C}$ and 760°C .

For the experiments for which data are given in Fig. IV.A.3, the pressure P_L' was 1.37 atm, the temperature T_L' was 531°C , the pressure PA_0 of the argon cover gas (indicating the amount of argon in the surface cavities) during the establishment of the 1.37 atm and 531°C pressure and temperature history was 0.03 atm, the sodium height L_0 above the heated length during the testing was 94 cm, and the nominal surface roughness was 0.64 micron.

(b) Empirical Correlation of Contact Angles and Partial Pressure of Argon in Surface Cavities. The empirical correlation of the ratio of advancing and receding contact angles is continuing. Data from both the sodium-superheat and the sodium-expulsion experiments indicate that it is best to assume the ratio of the advancing and receding contact angles equal to unity.

A theoretical study of diffusion of inert gas into and from surface cavities has been completed,* however, the analysis has not yet been confirmed experimentally. Because all the Argonne experimental data were obtained with a very small amount of inert gas in the surface cavities, it does not seem possible that the scatter in these data can be attributed to an inert-gas effect.

(c) Verification or Revision of Superheat Model. The data from the sodium-superheat and sodium-expulsion experiments strongly support the pressure-temperature history model for predicting incipient-boiling superheats when boiling initiates from surface cavities. That prediction seems to be applicable to flowing-sodium systems when account is taken of the two effects: (1) the presence of inert-gas bubbles in the liquid stream provides additional nucleation sites, and boiling might initiate from these bubbles, thus significantly reducing the incipient-boiling superheats, and (2) a relatively steep temperature profile in the liquid might exist during flowing tests, resulting in relatively low bulk superheats even when the local wall temperature is considerably superheated.

b. Sodium Simulations (M. A. Grolmes and H. K. Fauske)

Last Reported: ANL-7577, p. 183 (April-May 1969).

(i) Expulsion of Nonmetallic Fluids from Single-tube Geometries (Static Tests). Two different modes of fluid ejection were observed in the expulsion of superheated nonmetallic fluids from single-tube geometries. A rapid slug-type ejection and a relatively slower two-phase flashing ejection phenomena were described previously (see Progress Report for January 1969, ANL-7548, pp. 117-119). The latter voiding process resulted in a dispersed two-phase mixture being expelled from a well-defined liquid, two-phase interface that propagates into the undisturbed superheated liquid. Additional observations of this phenomenon indicate a threshold in the liquid superheat required for initiating and sustaining this flashing mode of expulsion. This initial threshold superheat is diameter-dependent for a given fluid, as shown in Table IV.A.1 for Freon-11. For the data in Table IV.A.1, the initial temperature T_0 was approximately 30°C and T_s was the saturation temperature at the reservoir pressure after rapid depressurization (see ANL-7548). The data should be interpreted in the following manner: For initial liquid superheat less than indicated in Table IV.A.1 for a given tube diameter, only evaporation takes place at the free surface and there is no two-phase flashing. For initial liquid superheat greater than indicated for a given tube diameter, two-phase flashing occurs; vapor plus entrained liquid are observed to proceed from the free surface. Within the indicated uncertainty of the initial liquid superheat, the two-phase flashing is intermittent and unsustained.

*Singer, R. M., and Holtz, R. E., On the Role of Inert Gas in Incipient Boiling Liquid Metal Experiments (to be published in the International Journal of Heat and Mass Transfer).

TABLE IV.A.1. Liquid Superheat
Required for Initiation of Two-
phase Flashing Ejection of
Freon-11

Tube Diameter (mm)	Liquid Superheat, $T_0 - T_s$ (°C $\pm$ 3)
2	58
5	48
10	41
20	32
50	28

These data, along with some selected measurements of exit pressures during this mode of fluid expulsion, indicate that, while two-phase critical flow phenomena might be present in some instances, phenomena governing the mass and energy transport at the interface play an important role in determining the actual two-phase flowrate and must be better understood. Experiments of this nature will be extended to include water and methyl alcohol in single-tube and multipin geometries. Multipin tests will be made to determine whether an

equivalent-hydraulic-diameter relation to initial liquid superheat exists in the same manner as illustrated for single tubes in Table IV.A.1. Analytical efforts will be made to determine general criteria for the onset of this flashing mode of expulsion, and to extend these criteria to ongoing work in the preparation of mathematical models of coolant voiding and reentry.

4. Core Structural Safety (C. K. Youngdahl)

Last Reported: ANL-7606, p. 127 (Aug 1969).

a. Dynamic Plasticity Analysis of Circular Shells

Results obtained for the dynamic plastic deformation of a circular cylindrical shell loaded by a pressure that varies both axially and with time indicate that the effect of variations of the axial load shape can be characterized by an effective length of loaded regions. This is illustrated by Figs. IV.A.4 and IV.A.5, which show the final plastic deformation at the center cross section as a function of the load length. Figure IV.A.4 considers the actual length; the results are plotted as a function of effective load length in Fig. IV.A.5. (The axial load shapes considered are defined as in ANL-7606; the pressure-time functions are the same for each curve.) The efficacy of the effective load length in correlating the deformation results is indicated by the bunching of the second set of curves as compared to the first.

The effective load length is defined by

$$a_e = \frac{2 \int_0^{z_y} z \phi(z) dz}{\left[\int_0^{z_y} \phi(z) dz \right]^2},$$

where z_y is the static yield hinge location. Because a_e depends on integrals over the load shape, it is insensitive to local inaccuracies in the measurement of the shape, which is encouraging for experimental investigations.

Fig. IV.A.4. Final Plastic Deformation Divided by the Square of the Impulse as a Function of the Actual Length of the Loaded Region for Various Load Shapes and Fixed Pulse Shape

Fig. IV.A.5. Final Plastic Deformation Divided by the Square of the Impulse as a Function of the Effective Length of the Loaded Region for Various Load Shapes and Fixed Pulse Shape

5. Fuel Element Failure Propagation (J. F. Schumar)

a. Out-of-Pile Studies (J. F. Marchaterre, R. E. Wilson, and B. M. Hoglund)

Last Reported: ANL-7577, pp. 187-190 (April-May 1969).

(i) Propagation Phenomena

(a) Gas-jet Impingement on Single Channel in Water Loop.

One regime of fission-gas release associated with potential fuel-failure propagation is that of a persistent gas jet that impinges on an adjacent fuel pin long enough to cause failure. The second phase of an experimental study of the heat-transfer mechanism produced by such an impinging gas jet has been completed. Encouragingly, it shows that the surface is cooled by the jet at the point of impingement.

This phase of the experiment was carried out in an electrically heated test section with a flow channel (0.050 in. x 0.578 in. x 22 in. long) having a heated length of 18 in. The test section permitted inserting gas-injection tubes whose inner surface was flush with the inside wall. An orifice plate was brazed to the end of each gas-injection tube; the orifice was 4 in. downstream from the test-section inlet, and thus

2 in. within the heated region. To measure the temperature transients, induced by the gas release, a thermocouple was mounted on the test-section wall directly opposite the orifice; a second thermocouple was mounted on the wall $3/4$ in. upstream. Coolant thermocouples and differential-pressure transducers were located at the channel entrance and exit. A turbine-type flowmeter was upstream of the test-section inlet.

The range of parameters studied was:

Heat Flux (Btu/hr-ft ²)	Coolant Velocity (ft/sec)	Orifice Diameter (in.)	Gas-plenum Pressure (psi)
500,000	30	0.016	500, 750, 1000
		0.025	500, 750, 1000
		0.050	500, 750, 1000
	20	0.016	500, 750, 1000
		0.025	500, 750, 1000
		0.050	500, 750, 1000
	10	0.016	500
	30	0.016	500, 750, 1000
		0.025	500, 750
0.050		500, 750	

Although the test data have not been analyzed, preliminary inspection shows that the wall temperature directly opposite the orifice always decreased during the initial phase of the transient. In some cases (0.050-in. orifice), the coolant was blown completely out of the channel and the wall temperature, after the initial decrease, rose to a peak value of $\sim 130^{\circ}\text{F}$ (worst case) above the steady-state temperature. Figure IV.A.6 is an example of the type of data obtained. In all tests, the thermocouple $3/4$ in. below the point of gas-jet impingement experienced a more severe temperature excursion than the one at the point of impingement. To check the hypothesis that this is caused primarily by the reduced coolant flow during the transient, calculations will be performed with the flow data from the tests. It is encouraging that no temperature transient was severe enough to damage the test section, because the flow transients in this confined channel are believed to be more severe than would be experienced in open-latticed fuel subassemblies.

(b) Hydrodynamics of 19-pin Bundle in Water Loop. Modifications of the water loop and assembly of the 19-pin test assembly have been completed, and the system has been checked. Three minor difficulties occurred during initial operation: (1) vibration-induced excessive noise in the coolant-flow-orifice transducers (this has been corrected in the test-section flow transducer but is still giving some problems in the total-flow transducer), (2) some small debris that had accumulated in the test loop

was deposited throughout the 19-pin bundle (several trial gas releases did not dislodge it), and (3) the nylon piston used to initiate the gas releases absorbed sufficient moisture to cause it to bind (this problem was resolved by machining the piston to a smaller diameter). The debris was cleaned from the test section while it was disassembled for repair of the gas-release piston.

Fig. IV.A.6. Typical Data from Test of Gas-jet Impingement on Heated Wall of Single Channel in Water Loop

Instrumentation calibration has been completed and a number of trial runs have been made with various lighting arrangements to determine which combination would provide the best definition in the high-speed motion pictures.

6. Fuel Meltdown Studies with TREAT (C. E. Dickerman)

a. Transient In-pile Tests with Ceramic Fuels

Last Reported: ANL-7595, pp. 115-118 (July 1969).

(i) Checkout-1 Mark-II Loop Experiments. The first TREAT loop experiment on meltdown of an oxide pin was performed on July 2, 1969, and preliminary results reported (see ANL-7595). Following posttransient radiography at TREAT of the test section in the loop, the test section was removed from the loop and shipped back to the Illinois Site for posttransient inspection. Transient instrument data are shown in Fig. IV.A.7. The rapid drop in both inlet and outlet flow resulted from shutdown of the Annular Linear Induction Pump (ALIP). The trip of the pump-power

Fig. IV.A.7. Transient Data from Checkout-1 Oxide Meltdown Experiment in the Mark-II TREAT Sodium Loop. Sample fission energy of 1.73×10^3 J/g.

extreme motion of fuel did occur. Postmortem inspection confirmed the radiographic data and showed that molten fuel had come into contact with the

circuit was traced to gamma heating of a temperature-control point. Since this transient heating did not represent an operational hazard to the loop, provisions were made to bypass this trip point just prior to initiation of a subsequent TREAT transient. "Pressure" signals shown for the inlet and outlet pressure transducer have the characteristic shapes of response of these two sensors to the high transient radiation levels in TREAT and quantitatively were in satisfactory agreement with extrapolations of the responses recorded in the preliminary, low-power transient conducted on the preceding day. Figure IV.A.8 shows an X ray of the sample test pin and two of the six (inert gas-filled) dummy pins run in a ring around the fueled pin. By contrast, Fig. IV.A.9 shows an artist's rendering of the pin condition after the experiment, as recorded in neutron radiographs. There was no depleted oxide pellet at the top of the fuel stack (see Fig. IV.A.8), and thus axial motion of molten fuel could occur readily. Figure IV.A.9 shows that

Fig. IV.A.8

X Ray Showing the Sample Fuel Pin for Checkout-1 Loop Meltdown Experiment, and Two of the Six Dummy Pins Run in a Ring around the Fueled Sample

cladding, "wetting" it and solidifying. Melthrough of the cladding did not occur. Pre- and postexperiment measurement of the cladding diameter showed no appreciable cladding strain. An upper limit of 0.3% strain was assigned, conservatively, to the measurement.

(ii) Checkout-2 Mark-II Loop Experiment. The loop experiment described above gave guidance on modeling fuel failure, demonstrated the effectiveness of axial motion in relieving radial loading of cladding, and gave support to calculations indicating that the flowing sodium coolant can effectively cool cladding during some meltdown incidents. However, it did not provide guidance on the consequences of oxide meltdown into flowing sodium with transient vaporization of the coolant, impact-pressure pulses, expulsion, and reentry, like those found in earlier loop experiments on meltdown of metal fuel. Consequently, a backup pin was modified and loaded into a test section as the Checkout-2 experimental sample, for a run at higher energy. The modified pin contained a natural UO_2 pellet at the top of the 13%-enriched fuel stack, to prevent axial melthrough. Then, a 5-cm-long steel rod was placed between the natural oxide pellet and the steel retainer spring providing inertia to resist upward axial movement. These two changes were specified in collaboration with the group working on the Accident Analysis program, to reflect some typical features of the current fuel design, while avoiding rigid axial restraint.

Checkout-2 was performed during the current reporting period. The test pin, as before, was contained inside a ring of dummy pins, and the initial sodium flow was about 4 m/sec through the test section. TREAT supplied about 2.59×10^3 J/g of oxide, in a power pulse of about 0.24-sec half-width. Failure was marked by anomalies on the three (inlet, center, and outlet) test-section thermocouples, on the two (inlet and outlet) flowmeters, and on the two (inlet and outlet) pressure transducers. Analysis of the data is underway. The failure time corresponds to an energy input of about 1.8×10^3 J/g. Maximum recorded pressure pulse was about 800 psi (i.e., only approximately twice the magnitude in the most energetic previous metal meltdown loop experiment). Preliminary data indicate extensive dispersal of the test fuel. Markers for the automated reduction of the neutron hodoscope data, were not successfully recorded. Hodoscope data on fuel motion are thus being analyzed manually.

Fig. IV.A.9

Artist's Rendering of the Posttransient Neutron Radiographs Showing Extensive Movement of Molten Oxide inside the Unfoiled Cladding of the Fuel Pin during the Checkout-1 Loop Meltdown Experiment

b. Experimental Support

Last Reported: ANL 7595, p. 118 (July 1969).

(i) Shipping Cask. Most, if not all, future Mark-II loop shipments (fresh, previously irradiated, and irradiated only in TREAT) will require a new shipping cask to meet current AEC-Department of Transportation regulations on shipping fissionable material. The commercial vendor has supplied the first of two packages of documentation required for AEC approval. This first package consists of the vendor Quality Control and Inspection Systems Manual, GWP No. 27-Welding of P-Number 1 Group of Materials, Procedure #103, Part A - Rev. B - Gamma Probe Testing, Procedure #131, Rev. 2 - M.P. Inspection of Weld Metal, Procedure #153, Rev. O-L.P. Inspection of Metal Surfaces, and Procedure #225, Rev. O-Helium Leak Testing. The second package, due to be transmitted at the end of this report period, will consist of vendor shop drawings. After both packages are received, the vendor submission will be given a technical review to ensure that it meets the operational requirements and that it will provide a cask consistent with the detailed ANL analysis for normal and accident conditions. Following this review and resolution of any questions that may be raised during the review, the ANL design analysis will be finalized. The vendor quality-control and quality-assurance procedures will also be reviewed.

(ii) Special Form Container. In order to use the shipping cask, it will be necessary for the Mark-II loop (or other experimental apparatus) to first be sealed inside a "special form" type of container. A detailed review has been made analyzing the loop shipping container used for shipping the prototype Mark-II loop to TREAT for the checkout tests. This review has confirmed the general design of the loop shipping container as satisfactory for service as a "special form" container, but resulted in specification of a minor change in the closure flange design. This analysis is being reviewed for approval prior to shop rework of the closure flange.

c. Analytical Support

Last Reported: ANL-7577, pp. 191-192 (June 1969).

During the safety review of the Checkout-1 Mark-II loop test, questions were raised concerning potential damage to the loop arising from sodium impacts at the top of the loop expansion volume, and secondary impacts if the sodium were to rebound downward after the original upward acceleration. These concerns arise from hazards calculations for a limiting case in which conversion of the order of 10% of the thermal energy input to the sample appears in kinetic energy of a sodium "slug." Calculated velocities at impact for this case are typically of the order of 400 m/sec.

(The maximum sodium velocity actually measured by a flowmeter in Checkout-2 was less than 20 m/sec.) Because Checkout-2 represented an exploratory experiment into a relatively unknown regime, the test-section support member inside the loop expansion space was modified to provide a design margin in the event the actual experimental conditions were to approximate the conditions of the hazards calculations. Four reenforced orifice plates were placed on the support member to limit the sodium slug velocity at impact. The bottom orifice was placed high enough that sodium expulsion could proceed unimpeded, until the test section was voided past the upper flowmeter. Thus, there would be no compromise of experimental objectives, at least on expulsion. Detailed sizing and placement of the orifice plates was made on the basis of an analysis of sodium slug displacement, velocity, and acceleration, using a nonlinear resisting force, for sodium of known (and constant) mass, density, and viscosity moving through a duct of variable area. Computer calculations were performed for a driving pressure of 410 atm (sodium critical pressure). The transient pressures resulting from the impacts of the sodium slug with the orifice plates and the top of the loop were calculated using results from Streeter and Wylie* and from Wright.** It was concluded that addition of the orifice plates would accommodate the conditions of the assumed accident.

7. Materials Behavior and Energy Transfer

a. High-temperature Physical Properties (M. G. Chasanov)

Last Reported: ANL-7581, p. 127. (June 1969).

(i) Heat Capacity of Liquid UO_2 . Measurements of the heat content of UO_2 sealed in tungsten capsules (see Progress Report for March 1969, ANL-7561, p. 99) are continuing. Data obtained to date at temperatures above the melting point of UO_2 are shown in Fig. IV.A.10. The liquid enthalpies are in good agreement, in the region of overlap, with those reported by other workers.[†] These measurements (to 3475°K) extend previously available data by more than 200°K. A straight line fits the liquid enthalpy data within about 1%, and the heat capacity of 33.6 cal/(mol)(°K) can probably be used for moderate extrapolations. A preliminary value for the heat of fusion of ~18 kcal/mol is also in good agreement with the literature[†] value of 18.2 kcal/mol.

At present, the calorimeter system is being modified for use with $(\text{U,Pu})\text{O}_2$. When this conversion is completed, attempts will also be made to extend the liquid UO_2 data to still higher temperatures.

*Streeter, V. L., and Wylie, J. B., Hydraulic Transients, McGraw-Hill Book Co. Inc., New York (1967).

**Wright, R. W., Pressures and Jets from Water Impacts Upon Open Structures, Trans. Amer. Nucl. Soc. 12(1), 363 (1969).

[†]Hein, R. A., and Flagella, P. N., Enthalpy Measurements of UO_2 and Tungsten to 3260°K, GEMP-578 (1968).

Fig. IV.A.10
Enthalpy of UO_2

8. Violent Boiling

a. Simulations of Fuel Dispersal (J. J. Barghusen)

Last Reported: ANL-7595, p. 122 (July 1969).

(i) In-pile Experiments on Material Interactions during Nuclear Transients

(a) In-pile Experiments with UO_2 in Piston Autoclave. To measure the pressures generated and the momentum transfer during a nuclear transient of energy sufficient to rupture stainless steel-clad UO_2 fuel specimens, a fourth meltdown experiment (S-4) was performed in the sodium-filled piston autoclave at the TREAT reactor facility. The reactor energy of the destructive transient was 1050 MW-sec, resulting in a fission-energy release in the test fuel of 720 cal/g UO_2 . The value of peak reactor power was 6650 MW, and the reactor period was 34 ms.

The fuel loading and configuration for this experiment were identical to those used for Experiment S-3 (see Progress Report for February 1969, ANL-7553, pp. 113-116). The sodium (197 g) was heated to $\sim 200^\circ\text{C}$ before the transient.

Although detailed analysis of the data from the experiment has not been completed, a few results can be reported. Five large discrete pressure pulses were observed (see Table IV.A.2); several minor pulses of amplitudes less than 400 psi were also recorded. The first two pulses, which occurred about 30 ms after peak power, were followed by a large pulse composed of two peaks less than 0.5 ms apart. It is not clear that this second pulse represents two separate pressurization events. The second peak of this double pulse had an amplitude of ~ 1800 psi, which is the largest pressure pulse observed in this series of experiments. The fourth pulse, 1280 psi, occurred 42 ms after peak power. The final pulse was significantly delayed, occurring 2.4 sec after peak power. This pulse

probably is not due to pressurization effects resulting from failure of a fuel rod, but might represent other interaction effects, such as flashing of sodium vapor. The bandwidths of all pressure pulses were about $1/2$ to $1/3$ of the widths observed in Experiment S-3.

TABLE IV.A.2. Principal Pressure Pulses Resulting from Meltdown Experiment S-4 in Sodium-filled Piston Autoclave^a

Pulse Amplitude (psi)	Bandwidth (ms)	Time after Peak Power (ms)
870	1.0	30
1360	0.8	31
1800 ^b	0.9	35
1280	1.0	42
780	1.0	2420

^aThe autoclave contained five fuel specimens (each of which contained 41 g of UO_2 that was 10% enriched in U-235), four dummy rods filled with helium at 15 psia, and 197 g sodium initially at 200°C . With a period of 34 ms, the reactor power rose to a peak of 6650 MW; integrated power was 1050 MW-sec, or 720 cal/g UO_2 . Fission-energy input at the first pressure pulse was about 480 cal/g UO_2 . Maximum piston velocity was 807 cm/sec.

^bAppears to be a double pulse: one peak at 1060 psi, the second peak at 1800 psi.

The maximum piston velocity was ~ 807 cm/sec, which occurred coincidentally to the 1800-psi pressure pulse. Temperatures greater than 3000°C were recorded by two tantalum-sheathed beryllia-insulated W-5Re/W-26Re thermocouples located in the upper portion of the sodium cavity.

Neutron radiographs of the piston autoclave showed that all fuel-rod specimens had failed and that UO_2 was widely distributed throughout the inside of the autoclave.

9. Post-accident Heat Removal

a. Engineering Analysis (J. C. Hesson)

Not reported previously.

(i) Decay-heat Sources in Fuel Components. For analyses of postaccident heat removal from damaged, collapsed, and molten cores,

information on the distributions of fission-product-decay heat sources in the solid, molten, and volatile components of the core residue are needed. Information on the decay heat (both beta and gamma) from fission products as a function of time and fission elements is being assembled from several sources and compared. Information on the probable distribution of these elements in the vapor, liquid, and solid phases of cores at various degrees of heating and melting is being developed. From this information the decay-heat distribution will be estimated.

(ii) Thermally Stable Fuel-debris Configurations in Sodium.

In analyzing heat removal from collapsed or molten cores under liquid sodium, the sizes and geometries of "thermally stable" configurations are important. (Thermally stable refers to geometries that are in thermal equilibrium with surroundings and do not erupt because of internal boiling or undergo significant geometry changes.). Analyses are being made of the sizes of thermally stable spheres, cylinders, plates, and pools as functions of decay-heat ratio, thermal conductivity, and sodium temperature, as well as the presence of adjacent structural parts. Internal circulation of molten fuel within the configurations is being considered. Analyses are also being made of the sizes of piles of pieces, cooled by sodium natural convection, as a function of particle geometry, pile porosity, decay heat rate, and sodium conditions.

(iii) Models of Core Slumping. For evaluating the movement of molten and slumped core materials for heat-removal analysis, models of core slumping are needed. Work is proceeding on two preliminary concepts of core slumping models.

B. Engineering Safety Features--Safety Features Technology--
Research and Development--Containment

1. Hydrodynamic Response to High-Energy Excursion (Y. W. Chang)

Last Reported: ANL-7577, p. 201 (April-May 1969).

To extend the hydrodynamic analysis to the domain where the tensile strength of the vessel material might become important, the computer code has been revised to include the effects of the membrane strength of the vessel material in the analysis. The mechanical properties of a solid material at high pressures can be described by using a compressible-fluid model. Thus, during shock compression, the momentum equation developed for fluids also can be used for solids. However, as the material recovers from the shock loading; or the material is under tension, the tensile strength becomes important, particularly for a cylindrical shell where the radial movement of the shell produces membrane forces, and where the tensile strength of the material is the key mechanism for stopping the motion of the shell.

The reactor vessels considered in the analysis must be concentric, right circular cylinders. Like the other media, the vessel wall can be divided into zones by grid lines, and the membrane strength of the vessel is assumed to be concentrated at one of the grid lines. If the vessel wall is relatively thin compared to other dimensions, it can be represented by a single line.

In finite-difference form, the momentum equation for the cylindrical shell considers that the acceleration in z direction remains unchanged, so the acceleration in the r direction becomes

$$\ddot{r}_{I,J} = \frac{1}{(A\rho)_{I,J}} \left\{ [(P_{I-1,J-1} - P_{I,J})(z_{I,J+1} - z_{I,J-1} + z_{I-1,J} - z_{I+1,J}) - (P_{I,J-1} - P_{I-1,J})(z_{I,J+1} - z_{I,J-1} + z_{I+1,J} - z_{I-1,J})] - 4\ell_1 \left[\frac{h_1}{r_{I,J}^0} \sigma(\epsilon) \right] \right\},$$

where $A\rho$ is zone mass, P is zone pressure, ℓ_1 is length of segment of the cylindrical shell, h_1 is thickness of the cylindrical shell, $\sigma(\epsilon)$ is the stress, the subscripts I and J are zone numbers, and the superscript 0 denotes the initial time.

For the bottom plate, the momentum equation in finite-difference form considers that the acceleration in the r direction remains unchanged, so the acceleration in the z direction becomes

$$\ddot{z}_{I,J} = \frac{1}{(A\rho)_{I,J}} [(P_{I,J} - P_{I-1,J-1})(r_{I,J+1} - r_{I,J-1} + r_{I-1,J} - r_{I+1,J}) - (P_{I-1,J} - P_{I,J-1})(r_{I,J+1} - r_{I,J-1} + r_{I+1,J} - r_{I-1,J})] - \frac{4\ell_2 h_2}{(A\rho)_{I,J}} \left\{ \frac{1}{(\Delta\bar{R})^2} \left[(z_{I+1,J}^0 - z_{I+1,J}) - 2(z_{I,J}^0 - z_{I,J}) + (z_{I-1,J}^0 - z_{I-1,J}) \right] \sigma_r + \frac{1}{2(\Delta\bar{R})r_{I,J}} \left[(z_{I+1,J}^0 - z_{I+1,J}) - (z_{I-1,J}^0 - z_{I-1,J}) \right] \sigma_t \right\},$$

where ℓ_2 is the length of segment of the plate, h_2 is thickness of the plate, $\Delta\bar{R}$ is average zone length, σ_r is radial stress, and σ_t is the tangential stress.

The revised code has been completed and is being documented. It will be available through the Argonne Code Center under the name REXCO-H.

A PLOT-3D Fortran subroutine has been developed to draw three-dimensional surfaces. It improves the visualization of the complicated functions such as pressure, displacement, and velocity from tabulated data on the computer digital printout. The PLOT-3D output subroutine can be displayed either by the IBM-2280 Film Recorder or the Calcomp-780 Plotter. PLOT-3D draws a three-dimensional surface from an array of points (x, y, z) , where x and y are the spatial coordinates of a point, and z is the value of the function to be displayed at that point. The surface can be drawn after arbitrary rotations. Figure IV.B.1 shows the surface of the pressure at time $810.75 \mu s$ as drawn by PLOT-3D with a Calcomp-780 plotter for a sample problem reported previously (see Progress Report for February 1969, ANL-7553, Fig. V.D.1, p. 121). The surface has been rotated first 60° around the z axis and then 30° around the y axis.

Fig. IV.B.1

Pressures (z) at Points on x - y Plot as Drawn by PLOT-3D Fortran Subroutine. Reactor radial axis is x ; axial axis is y .

a. System Geometries and Excursion Magnitudes and Durations

Not previously reported.

The REXCO-H code developed for containment analysis can be used for studies of system geometry. The reactor designer can use the code to study the failure sequence of various components and its effects on loads imposed on other system components, or to vary a single parameter so that its independent effect can be studied.

Fig. IV.B.2. Reactor Configuration with Two Vessels is Typical of Arrangement for Which REXCO-H Code Can Be Used to Study Sequence of Failures and Effects of Loads Imposed on System Components. Inner vessel in this example has 390-cm diameter and 7.5-cm-thick wall.

For example, to study how rupture of a reactor vessel affects loads imposed on the shield plug, the reactor designer could consider using a reactor configuration with two vessels, as shown in Fig. IV.B.2. Figure IV.B.3 shows the force acting on the shield plug for systems with or without the inner vessel. The inner vessel directs the kinetic energy of the coolant upwards, lifting the coolant and compressing the inner gas blanket much harder than does the system without the inner vessel. Thus the force acting on the rotating shield plug could be reduced considerably if the inner vessel were designed to rupture at the beginning of the excursion. By varying the numerous parameters involved in the codes, the designer can study their effects on the overall safety of the design.

Fig. IV.B.3. How Presence or Absence of Inner Vessel That Ruptures during Excursion Affects Total Force Imposed on Shield Plug

b. Transformation of Code Coordinates from Lagrangian to Eulerian Systems

Not previously reported.

Experience shows that the Lagrangian method of simulating the fast-reactor excursion does not supply all the relevant parameters for a complete containment design. The Lagrangian solution must be terminated prematurely because of excessive distortion of the fluid mesh. At the point of termination, the effect of the shock wave on the containment can be evaluated, but the effect of the gas bubble can not.

To study the progress of the excursion further, use of the Eulerian approach, in which excessive deformation would not be a problem, has been considered. A search of available Eulerian methods showed that the PIC* (particle-in-cell) method, in particular, had been greatly successful in solving numerous practical problems of compressible fluids. After study indicated that the PIC method could be adapted to the containment problem, relevant variables from the Lagrangian output at a predetermined time were transformed by the TRANSF routine into input for the PIC routine. This transformation was written in Fortran for IBM-360 computers.

Basically, this computer program converts a volume of a given material into a chosen number of particles per unit of initial grid area. Then the program divides the individual (deformed) grid area into as many equal portions as there are particles and assigns them to their respective centroids. Concurrently, each particle is given the properties of the original Lagrangian mesh. A chosen Eulerian grid is superimposed on the various

Fig. IV.B.4. Example of Deformed Lagrangian Grid for Containment Study

particles established previously. Location of the particles within the Eulerian grid is noted and the particles are labeled accordingly. Then properties of the individual Eulerian grid are determined by averaging the particle properties within each grid. Figure IV.B.4 shows a deformed Lagrangian grid; Fig. IV.B.5 shows a corresponding Eulerian particle distribution.

Now that the input for the PIC procedure of the Eulerian method of solution is complete, it will be developed for the continuation of the containment problem.

*Particle-in-Cell Method for the Calculation of the Dynamics of Compressible Fluids, LA-3466 (June 1966).

Fig. IV.B.5

Eulerian Particle Distribution Corresponding
to Deformed Lagrangian Grid Shown in
Fig. IV.B.4

2. Inelastic Response to High-Energy Excursion (G. Cinelli, Jr.)

Last Reported: ANL-7577, p. 202 (April-May 1969).

a. Introduction of Material Equations of State into Codes

The introduction of various equations of state into the computer program has been accomplished. Sample problems can have any number of different media the user desires.

b. Introduction of Variable-mesh Approach and Sliding-surface Routine to Extend Code past Shock-wave Durations

The analysis on the sliding-surface routine is being developed.

When materials slide on one another, a decoupling of grid points on the interface must be provided; otherwise large grid displacements will result. For a large class of problems, a simple decoupling of the grid points gives satisfactory results. The method adopted in this program considers one surface to be a fixed boundary during a given time step Δt . The equations of motion for the sliding material are the same as those given for motion along a fixed boundary. Then the fixed boundary is advanced in time using the force field of the sliding material next to it. The new position of the fixed boundary provides a new boundary for the sliding material. In this type of calculation, it is essential that the parameters on the interface of the sliding material be associated only with the sliding material and that the material providing the fixed boundary be treated as though the boundary were an exterior surface with pressure forces acting on it.

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