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MASTER

STUDY OF REQUIREMENTS AND SUITABILITY OF AVAILABLE REACTORS FOR FAST FUEL TESTS

AEC Contract No. AT(11-1) - 865, Project 7

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February 23, 1963

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.

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I. INTRODUCTION

One of the great needs in the development of economic fast reactor fuel is to be able to test the fuel under simulated ~~fast reactor~~ conditions. The fast reactor fuel tests for the Experimental Breeder Reactor-II (EBR-II), the Enrico Fermi Atomic Power Plant (Fermi) and the Fast Ceramic Reactor (FCR) have been irradiated in thermal neutron fluxes with resulting burnup depressions in the fuel sample, low fast neutron exposure on the cladding, and a stagnant liquid metal environment. As a consequence, the application of the results are uncertain because they have not duplicated the conditions expected in the application. Further, the tests have been limited to single pin tests.

Similar deficiencies in irradiating thermal reactor fuels have been overcome with the use of loops in thermal test reactors. Not only can environments be more closely duplicated, but, in addition, these loops include the ability to test defective elements and to test fuel elements to failure to determine the highest reliable operating conditions.

Tests in a test reactor to determine the ultimate capability or potential of fuel by determining the conditions that cause failure would permit operation of this fuel in a fast power reactor up to the maximum burnup of the fuel. Without this knowledge, it is necessary to remove fuel prematurely from the reactor and to operate the fuel at low specific power. For example, in Fermi it has been decided to remove the fuel at approximately one-half the burnup and operate at two-thirds of the power that the fuel is believed to be capable of, because of uncertainties in the irradiation conditions that were experienced in the irradiation program for the fuel. Part of this potential can be realized by periodic examination of the fuel, thereby gathering sufficient data under reactor conditions to allow increasing the burnup and power. However, this is a rather time-consuming process and will involve considerable delays in the operation of the reactor. By irradiating fuel under duplicate conditions, postirradiation inspection of the fuel probably could be eliminated.

The purpose of this study is to establish the fast fuel testing requirements for the civilian fast reactor program and to determine the capability of existing test and fast reactors to meet these requirements. The potential fast reactor fuel experimenters were canvassed to determine their estimates of the projected future fast testing requirements, and these estimates were then combined into a master list as a part of this report. The potential fast fuel experimenters are Argonne National Laboratory (ANL), Atomic Power Development Associates, Inc. (APDA), Atomics International (AI), General Electric Company (GE) and United Nuclear Corporation (UNC).

Based on a study⁽¹⁾ of the use of a boron filter to produce an acceptable fast neutron testing environment in the Advanced Test Reactor (ATR), similar filter experiments for fast fuels were studied in the Engineering Test Reactor (ETR), the Experimental Organic Cooled Reactor (EOCR), and the PEP modification of the Sodium Reactor Experiment (SRE). Studies and inquiries were also made to determine if liquid metal loops were feasible and could be operated in the EOCR, Fermi, and EBR-II. It was concluded that since Pratt & Whitney Aircraft operated a package liquid metal loop in the ETR, no further layout and design work was required to establish that it was feasible to install loops in this type of reactor.

As a part of this study, the possibility of converting the EOCR to a fast test reactor (FTR) was considered. The EOCR presently is scheduled to be "mothballed" as a result of a de-emphasis on the organic cooled reactor program and various possible other uses for the reactor are being considered. The conversion to an FTR is considered one possible use. Two conversion possibilities were considered: (1) to use sodium as a coolant in the present facility and install a fast reactor core into the present EOCR reactor vessel, or (2) to utilize the organic coolant as a secondary cooling system for a sodium cooled fast reactor which could be built alongside the present facility. In this case, the sodium to organic intermediate heat exchanger would be installed in the space now occupied by the EOCR reactor vessel, or it is conceivable that the intermediate heat exchanger tube bundle could be inserted directly in the EOCR reactor vessel.

II. SUMMARY AND CONCLUSIONS

The purpose of this study is to determine the requirements for radiation space for fast reactor fuel testing and to determine the capability of existing Commission and private test and power reactors to meet these requirements. In the study, particular attention is given to the possible modification of the EOCR to a sodium cooled fast reactor fuel testing facility.

The main conclusions of the study are:

1. The future requirements for fast fuel testing as compiled, based on the estimates given by potential fast fuel experimenters, are extensive and cannot be completely met with existing test or fast reactors.
2. Only a part of the total program can be accomplished by using liquid metal loops in test reactors and these tests would be very costly.
3. Most of the proof testing of subassemblies can be accomplished in either the EBR-II or the Fermi fast reactor.
4. A fast test reactor with considerable testing space in the form of open and closed loops will be required for the fast reactor program. A total of eight closed loops and eight open loops of from 3 to 6 inches in diameter would be required, in addition to ten 1-1/2-inch holes and several 5-inch holes for capsule experiments for the total program. Part of these requirements can be considered met by the subassembly positions in EBR-II and the Fermi reactor. The closed loops must be designed to accommodate fuel tests taken to failure, and to operate at a very high fast neutron flux. These large closed loops are the principal deficiency in the existing fast reactors.
5. In the ETR and ATR, it is theoretically possible to install a loop or a capsule with a boron filter to obtain sufficiently high heat generation rates without serious flux depressions to obtain a worthwhile test. In these cases the total test reactor power that is required is very high and would result in very high operating costs⁽³⁾ and low reactor utilization, i. e., a large portion of the test reactor would be tied up with the fast loop test or capsule. The basic reason for this is that the filter removes the thermal neutrons whose production has been optimized in the reactor design, so that a satisfactory fast flux in the test region automatically involves a high power density in the thermal core.

6. The fast region in the proposed two-region SRE with the PEP thermal core appears to be mechanically satisfactory and feasible from a material compatibility standpoint. A fast converter in this reactor design is somewhat limited, since the total heat generation in the fast sample is about one-third of the value in the ETR for the same acceptable or superior flux depression. By substituting enriched for depleted uranium in the mixed oxide test sample, the minimum acceptable heat generation could be achieved.
7. A liquid metal fast loop in the EOCR is easier to accommodate mechanically and physically than in other test reactors which are not cooled by liquid metal, because of the better compatibility of sodium and organic. Using a boron filter, the heat generation in the fast test sample is about one-third of the value in the ETR and is, therefore, not particularly attractive.
8. It does not appear to be practical to directly convert the EOCR to an FTR because of the need for containment and the major neutron shielding changes required.
9. However, it does appear to be feasible and attractive to consider building a small FTR immediately adjacent to the EOCR, using the EOCR reactor vessel as a heat exchanger vessel between sodium and organic and the existing organic as a secondary coolant system.
10. Although no detailed physics analyses were made, it appears that the MTR, the General Electric Test Reactor (GETR) and the Westinghouse Test Reactor (WTR) are not suitable for filtered fast loop experiments. In these reactors the power density is considerably below that required for a satisfactory heat generation in the fast fuel specimen. In addition, no provisions were made in the MTR to install loops in the core. It may be possible to increase the power density or harden the spectrum, however, to make these reactors suitable for this work.

III. RECOMMENDATIONS

The following recommendations are based on the results and conclusions of this study:

1. A study should be made to determine the cost of a small 40 to 50 Mw FTR completely separated from the EOCR and one that utilizes the EOCR buildings and the present organic primary system as the secondary coolant for the FTR. This study should include a conceptual design of the FTR, an evaluation of the preferred test reactor neutron spectrum (i. e. , whether it should have a fast or intermediate neutron spectrum), the preferred core configuration, the fuel element concept and type of fuel, method of fuel and loop handling, the amount of experimental space and flexibility that could be provided, etc. Cost estimates of development, engineering, construction, and operating costs should be included and schedules determined. The purpose of the study is to determine the capability, feasibility, and cost of a small FTR, and to determine the cost savings if portions of the EOCR facility could be used.
2. Following the study of the small FTR, a conceptual design of a single fast test reactor, having the capability of meeting all the requirements listed in Table I with minimum reactor size, should be made. This study should also include an evaluation of the preferred neutron spectrum, the preferred core and fuel element configuration, the type of fuel selected for good operating economics, method of fuel and loop handling, etc. The study should also include estimated development, engineering, construction and operating costs. A schedule should be provided. The purpose of this study is to determine the comparative economics of a large FTR as opposed to several smaller test reactors. The first and second studies would form the basis for a decision as to what type of reactor to develop.
3. Additional studies of a more optimum method of utilizing the EOCR with liquid metal loops should be made. By hardening the neutron spectrum in EOCR through a redesign of the EOCR fuel element and core and reducing the boron in the neutron filter, it may be possible to increase the power generation per unit length of fuel pin to the required minimum value of 1 kw/inch with an acceptable flux depression. In addition, by using a converter region instead of a neutron filter, an improved power generation may be obtained. This converter would consist of a ring of U-235 cooled with a nonmoderator, such as sodium. Instead of the boron steel ring neutron filter, a plate of fully enriched UO₂ dispersed in stainless steel and clad with stainless steel would be used.

4. Provisions should be made to utilize to the maximum possible extent EBR-II and the Fermi reactor for proof testing of fuel sub-assemblies.
5. The SRE should not be converted to a fast fuel test facility unless a power generation of at least 1 kw/inch of pin length in a typical 19-pin fuel experiment can be obtained. The use of a converter region could barely produce this heat generation. However, by hardening the neutron spectrum (i. e., increasing the enrichment and removing some graphite moderator), it should be possible to improve the heat generation in the typical experiment studied.
6. No further work should be done on the direct conversion of the EOGR to an FTR.
7. No further work should be done now on the design, development, and construction of fast liquid metal loops in water cooled test reactors. If and when it is decided not to provide other more desirable fast fuel testing facilities, this work should be reactivated. In any reactivated effort, consideration should be given to hardening the spectrum of the water cooled test reactor by increasing the fuel to moderator ratio. This would permit a smaller amount of boron to be used in the filter and increase the flux in the fast fuel sample. The result would be a lower operating cost for the experiment.

IV. FAST REACTOR FUEL IRRADIATION REQUIREMENTS

The fast reactor fuel testing needs are summarized in Table I. These needs represent a compilation of the answers to an inquiry sent to potential fast reactor fuel experimenters.* This inquiry requested each organization to provide their estimate of the national fast fuel testing requirements. The representatives of these organizations agreed⁽³⁾ that a FTR was necessary to meet these needs, and their stated requirements are based to some extent on the assumption of the existence of such a facility. This approach at least allows one to draw a comprehensive picture of the irradiation requirements. The alternate approach, that of restricting the requirements to the experimental possibilities of the existing thermal test reactors, would result in a distorted view of the requirements. Now that these comprehensive requirements have been tabulated, the existing thermal test reactors can be examined to see if they can meet the requirements, as is done in Section VI of this report.

The three broad needs of the fast reactor fuel irradiation program are as follows:

1. To determine the effect of the fast neutron exposure on the physical properties of structural and clad materials to exposures ranging from 10^{24} to 10^{25} nvt at energies greater than 0.1 Mev.
2. To study the effects of high burnups in fast reactor fuels subjected to high exposures and to failure. For example it is conceivable that an oxide fuel could reach a burnup of 250,000 MWD/Tonne of the oxide (approximately 6.8×10^{21} fissions/cc of the oxide) although failure may occur first.
3. To determine the combined effects of heat transfer, fluid flow and radiation damage on full scale fast reactor fuel subassemblies operating under simulated design conditions and to failure.

Most, if not all, fast fuel irradiations conducted to date have been capsule tests in thermal reactors. Further insight into the needs of fast fuel testing can be obtained by listing the disadvantages of irradiated fast fuel specimens in capsules in thermal reactors:

* Argonne National Laboratory	- J. H. Kittel, E. Sowa
Atomic Power Development Associates, Inc.	- W. H. Jens, A. A. Shoudy
Atomics International	- D. T. Eggen
General Electric Company	- K. Cohen
United Nuclear Corporation	- G. A. Sofer

1. Clusters of fuel elements containing required amounts of plutonium or uranium cannot be irradiated without excessive variation in neutron heating within the cross section of fuel elements.
2. Fuel elements cannot be irradiated to failure in flowing liquid metal or other coolants to determine the extent of fuel redistribution or other dynamic effects resulting from clad or support structure failure.
3. The effect of fuel element movement due to coolant flow or temperature differences induced by flow cannot be evaluated. Distortion due to thermal differences cannot be determined.

Further disadvantages can be listed for the next best approach - fast neutron loops for liquid metal coolants in thermal reactors, e.g., the proposed ATR fast loop:

4. The large test section diameter necessary to accommodate full scale fast reactor subassemblies is difficult to achieve.
5. Fuel failure tests are of principal interest; yet instruments for detecting failures cannot be readily incorporated.
6. Since the ratio of fast to total fissions is not as high as in existing and contemplated fast reactors, the integrated fast neutron dose on the clad for a given burnup is not as great and therefore this effect cannot be evaluated. A filter or converter design to achieve the correct ratio is difficult, if not impossible, to accomplish. Furthermore, use of a filter or converter results in an inherent high neutron cost due to the wasteful absorption of the thermal neutrons. For example, in the proposed ATR with fast loop, 40 Mw of heat is removed from the thermal test elements in a lobe to drive a 100 kw fast fuel test element. The burnup of U-235 in the thermal test elements and fabrication charges for thermal reactor fuel result in very high operating costs of several million dollars per experiment per year.⁽¹³⁾

An alternate approach to accomplish the fast fuel testing is to do the required testing in loops that are incorporated in either fast or intermediate spectrum test reactors. Based on the requirements tabulated this would require either a very large test reactor or several smaller reactors.

A large FTR could provide at least ten times the available test space of two ATR loops under more typical fast reactor conditions. In addition, it would provide space needed to conduct clad and structural material damage studies vitally needed in the fast reactor development program. It is anticipated that the reactor would have a power range of 100 to 500 Mwt, the level depending partly on the total size of the experimental regions. The design

should emphasize experimental accessibility, ease of experimental handling and instrumentation, and reliability of reactor operation.

A fast reactor sufficiently large to contain all the required test facilities, may be very expensive and difficult to design and build, on the basis of our present day knowledge. For this reason, the building of several smaller test reactors has been considered. The advantages of this approach might be earlier availability of several fast neutron testing facilities, and overall lower cost of the fast reactor fuel testing program.

Several smaller fast or intermediate test reactors may be more desirable than one large reactor to simplify the operation of the test facility. In the case of a small test reactor the plant would be operated to meet the conditions of one large central loop to avoid the conflict in a large reactor where some of the loop experiments must be satisfied with conditions imposed by the principal experimenter. This approach is discussed in more detail in Section VIII, titled "Conversion of EOCR to FTR."

The requirements in Table I seek to strike a balance between the stated needs of the individual experimenters. These needs were obtained for the purposes of this report and represent the most up-to-date statement of fast reactor fuel irradiation requirements (Ref. 2). Some divergence was naturally present between the individual requirements, and it is difficult to make a satisfactory convergence to a definite value for each specific requirement. However, the table seeks to establish a suitable balance by giving the nominal value. The table also indicates the range of the stated requirements. Where this range extends far past the nominal value, it is noted in the text to follow. Also noted are remarks that were difficult to be included in the table.

Most of the values given are obvious and require no further explanation. In most cases the nominal value given is a single figure or otherwise obvious. In some cases, however, a range is given for the nominal value, e. g., for specimen lengths, power generation, etc. This indicates the range in which most experiments would take place.

The following points refer to those requirements which have a considerable deviation from the nominal value, and are numbered similarly to the numbers in parentheses in the "range" column of Table I.

- (1) The maximum requirement of 0.5 Mev for the neutron energy peak was stated (UNC-G. Sofer) in order to obtain equivalent damage in a shorter time.
- (2) For those experimenters who required shorter specimen lengths, it is assumed their requirements on total number of holes could be met in a reduced number of holes by stacking.

(3) The maximum requested value of 2000 F outlet temperature (ANL-E. Sowa) refers to lithium and boiling sodium coolants.

(4) The U-235-U-238 system was requested (UNC-G. Sofer) for mobile applications.

(5) This high power density is believed (UNC-G. Sofer) to represent a limit for the PuC-UC system.

(6) It may not be necessary to test the largest subassemblies (ANL-E. Sowa), in which case it is sufficient to test a 3.5" diameter subassembly and extrapolate the results. Furthermore, loops greater than the size needed to accommodate a 3.5" subassembly would be difficult to design and build.

The positions of the holes and loops in the reactor have not been specified. Again, assuming that a FTR is constructed for this testing, it would seem that the loops and perhaps half the holes should be in the core. The numbers given for the holes and loops in the nominal list of each column are additive. Thus the core of the FTR would contain 8 closed loops, 8 open loops, about 10 x 1.5" holes and one 5" hole. The rest of the holes would be in the reflector (the FTR would probably contain no blanket). Although not mentioned in the summary requirements, various test positions could, of course, be included in the shielding to test principally shielding materials.

It is APDA's opinion that the core should be long enough to test fuel specimens with a minimum length of 4 feet. The core may have a high length-to-diameter ratio in which case the large amount of leakage into the reflector would make it suitable for many test positions.

TABLE I

SUMMARY - FAST REACTOR FUEL IRRADIATION REQUIREMENTS

	Column 1		Column 2		Column 3		Column 4	
	Structural Material Requirements		Fuel Specimen or Single Fuel Element		Small Group of Fuel Elements		Fuel Subassembly	
Spectrum & Flux	Nominal	Range or Comment	Nominal	Range or Comment	Nominal	Range or Comment	Nominal	Range or Comment
ϕ_{max} , n/cm^2 sec	1×10^{16}	5×10^{16}	1×10^{16}	$5 - 2 \times 10^{16}$	Same as Column 2 (Single Fuel Element)	Same as Column 2 (Single Fuel Element)	Same as Column 2 (Single Fuel Element)	Same as Column 2 (Single Fuel Element)
ϕ_{min} , n/cm^2 sec	1×10^{15}	5×10^{15}	1×10^{15}	$.1 - .5 \times 10^{15}$				
$d\phi/dL$, %/ft	<50%	<30%	<50%	<30%				
$d\phi/dR$, %/in.	<10%		<15%	<10 - 20%				
$d\phi/dt$, %/cycle	$\pm 10\%$	$\pm 25\%$	$\pm 10\%$					
Neutron Energy Peak, Mev	0.2	0.1 - 0.5(1)*	0.1	.01 - .5				
Size & Number of Holes								
Minimum Diameter, inch	1.5	1.25 - 4	1.5	.5 - 4	1.5	1.1 - 2.0	2.5	2.0 - 3.0
Maximum Diameter, inch	5	3	2.5	1.5 - 8	2.5	2 - 3	5	3.5 - 6.0(6)
Length, inch	36	12-18	36	3 - 72	36	18 - 48	42 core; 96 total	36 - 48 core
Number Required	$5 \times 1.5"$ $1 \times 5"$	+8 loops(2)	16	+4 loops(2)	4 closed loops 2 open loops	6 closed loops 1 rabbit	4 closed loops 6 open loops	1 - 8 closed loops 2 - 12 open loops
Access & Instrumentation								
Thermocouples	Yes		Yes		Yes		Yes	
Heater Leads	Yes		Yes		Yes		Yes	
Gas Lines	Yes	No	Yes		Yes	No	Yes	No
Cooling Lines	Yes		No	Yes	Yes	No	Yes	No
Others	Yes	No	Yes	No	Yes	No	Yes	No
Environment								
Coolant	Liq Met/Gas	+Steam	Liq Met/Gas/Steam		Liq Met/Gas/Steam		Liq Met/Gas/Steam	
Flow - 1 hole, gpm	TBD		5/TBD/TBD	0 - 300 No	100/TBD/TBD		600/TBD/TBD	
Temperature								
T_{in} min, F	600	300 - 900	500/500/500	900	500/500/500	900	500/500/500	900
T_{out} max, F	1400	800 - 2000(3)	1400/1400/1100	2000 Li(3)	1400/1400/1100	2000 Li(3)	1400/1400/1100	2000 Li(3)
ΔP , psi	100		100/100/100	<25 Liq Met	100/<100/<100	<25 Liq Met	100/<200/<200	<25 Liq Met
P_{inlet} , psia	<150		<200/<1000/<1500		<200/<1000/<1500		<250/<1000/<1500	
Typical Experiment								
Number of Specimens & Geometry	1, or more if stacked		1, or more if stacked		7 - 19	9	100	7 - 500
Diameter, in.	TBD	0.1 - 2.0	.2 - .5	.1 - .5	.2 - .5	.1	0.2 - 0.5	.1
Length, in.	TBD	2 - 36	15 - 36	2 - 72	12 - 36	18 - 48	36 core	12 - 48 core
Fuel Composition	Structural/Refractory Materials		Metal, Oxide, Carbide, Nitride, Cermets, Pu-U/ U-Th	+U-235-238(4)	Same as Column 2	Same as Column 2	Same as Column 2	Same as Column 2
Maximum Power/Element, kw/in.	Low		1 - 5					
Allowable Flux Depression, max/avg	Low		1.1	1.5	1 - 5	6.25(5)	1 - 5	6.25(5)
Data Required	Physical changes as function of time, temperature; integrated fluxes; gas generation.		Operating history; integrated fluxes; metallurgical changes; physical changes; fission gas pressure; fission product dis- tribution; time of failure.		1.2 in bundle	1.1 - 1.3	1.2	1.1 in each rod
					Same as Column 2		Same as Column 2	

* The numbers in parentheses are referred to in the text.

V. SUMMARY OF TEST REACTORS

Thermal test reactors have been designed and built to provide radiations of interest such as neutrons and gamma rays in the various experimental spaces provided in the reactor. In general also, all the thermal test reactors were built to provide a high thermal flux. The fast flux which is of the most interest in this study was not a major consideration.

The exact flux obtained in any experiment will depend on the reactor loading, the other experiments in the reactor, the specific composition of the experiment being run, and the core and reflector geometry of the particular reactor. Therefore, the tabulated values shown in Table II for neutron flux must be considered merely as representative values. The thermal flux is particularly sensitive to the amount and kind of moderator that is present. For example, the thermal flux in the MTR beryllium reflector is higher than the core due to the added thermal flux created by the slowing down of fast neutrons in this region. Since in this study the interest is fast flux for fuel testing, in general the core fast flux values are given. These values will be considerably less in a reflector position or in the presence of a large amount of moderator.

Because of the complex nature of the energy distribution for the fast loop experiments studied, it was felt that the only comparison that could be made and justified would be to calculate, using multigroup diffusion methods, the power generation and flux depression in typical experiments for the various thermal test reactors studied. This has been done, and the results are given in Section VI.

TABLE 11

A. REIC No. 16, B. Raytheon-Nuclear Reactor Data - 2, 1956, C. IDO-16820, D. 1958 Geneva Conf. Vol. 10, Pg. 107, E. Calculated from critical mass, reactor power, and core volume, F. IDO-16667.

VI. PHYSICS ANALYSIS OF FAST FUEL TESTS IN THERMAL REACTORS

A. INTRODUCTION

It is the purpose of this analysis to evaluate the performance of fast fuel samples placed in existing thermal test reactors. Fast fuels have a very high fissionable material concentration which makes them very black to thermal neutrons. To produce power profiles in those fast test samples that are sufficiently flat to meet the test criteria (see Table I), it is necessary to prevent the low energy neutron flux from entering the sample. This can be accomplished by surrounding the test sample with a sufficient amount of a strong neutron absorber (neutron filter). The sample power will then be dependent on the epithermal and fast fluxes that are allowed to filter through the sample. Multigroup diffusion theory analysis has been used to evaluate the power depression and power levels occurring in the sample placed inside such a filter. This technique is described in more detail in references 1 and 14.

It is possible, however, to predict on a rather simple basis what the relative sample performance will be among the various test reactors when simple filters are used. The approach used is as follows. Assume all the thermal flux is easily restricted from entering the sample. The power generation in the sample will thus be due to the available fast flux in the thermal reactor. For thermal reactors of similar leakage and fast removal cross section, it can be shown that the fast flux is proportional to the power density.⁽⁴⁾ Thus, an approximate figure of merit for testing fast fuel in a thermal test reactor will be the nominal thermal core power density. These are listed in Table II as obtained from the literature. Values consistent with the calculational models used in this study are given in Table III.

It is interesting to compare this figure of merit for the present application to that which is normally considered in a thermal test reactor, the thermal flux. It can be shown⁽⁴⁾ that the thermal flux is proportional to the reactor power divided by the total mass of fuel (U-235). Thus, the thermal flux can be maximized by going to designs of large cores with very low fuel concentrations. However, the fast flux is restricted by the maximum power density which in turn is usually limited by the maximum allowable heat flux in order to prevent burnout.

The sample power could of course be enhanced by the use of a converter instead of or in addition to a neutron filter. However, the design of such a converter is beyond the scope of this study because such a design would require a careful optimization study between physics performance and converter heat removal. Thus, the nominal filter design (described in Reference 1)

TABLE III

COMPARISON OF THERMAL TEST REACTORS

Reactor	Active Core Volume, Liters	Fission Power, Mw	Maximum Allowable Power Density, Mw/Liter	Critical Mass, kg	Average Power Density, Mw/Liter	Radial Max/Average Power
ETR	287	163	1.1	15.9*	0.568	1.6
EOCR	510	37	0.26	60*	0.0725	1.86
SRE-PEP	3840	30	-	36.8	0.0073	1.21

* Includes thermal experiments.

consisting of a 1/8-inch-thick ring of 1.2 w/o B-10 steel is used in all the cases considered with one exception. Design work has already been performed on the SRE core where a filter of natural uranium has been chosen. Further, the design also essentially incorporates a well cooled converter inside of the filter. (See Reference 5). This design has been used in this study for evaluating the performance of the reference test sample in the SRE.

Due to the brief scope of the work, a number of assumptions were necessarily incorporated that lead to some uncertainty in the final performance results. Most of these assumptions are identical to those incorporated in the similar study performed for the Advanced Test Reactor (ATR), and are discussed in Appendix A of Reference 1. However, the results are sufficiently accurate to provide meaningful comparisons among the various reactors.

The power generation rates, power depressions, etc., in various fast reactor test specimens inserted in the various thermal test reactors available was determined in the analysis for each reactor type. The reactor types considered are listed, as follows:

1. ETR-type, with fully enriched U-Al alloy fuel element, moderated and cooled by light water. This type includes: MTR, ETR, ATR, GETR, WTR.
2. EOCR with fully-enriched UO_2 -SS cermet fuel elements, moderated and cooled by organic coolant.
3. SRE with low-enriched UC fuel elements (PEP core) moderated by graphite and cooled by sodium.

At least two definite philosophies of operation can be envisioned. One is to operate the test reactor for the sole benefit of the fast loop assuming no other experiments are present. Thus, high fuel concentration, small core, and the minimum total power are desirable. The other alternative is to use the nominal size core with the possibility of allowing other experiments and thus to operate at the full allotted reactor heat removal capacity, here to be taken as the nominal power rating of the core. The two basic types of reactors in which filtered loops are placed (ETR and EOCR) were treated with these two philosophies in mind. That is, operation with both a small core and a nominal core was assumed. In some cases, intermediate sizes were also considered. However, for the case of SRE, only the proposed two-region reactor with the PEP thermal core as given in references 6 and 7 was considered. In this two-region system, with a large fraction of the core power being generated in the fast region, the option of running additional thermal experiments is not considered.

It is obvious that both systems involve a detailed investigation of the complete flux energy spectrum from fission neutron energies down to thermal

energies. Although most fast reactor core calculations can ignore the epithermal and thermal energies, these energies exert a direct effect on the present calculations for two reasons:

1. The thermal core must be calculated with sufficient accuracy to give the approximate critical size or critical fuel concentration and power distributions, because these characteristics affect the performance of the fast fuel being tested.
2. The fast fuel specimen should be calculated with accurate values for the epithermal cross-sections, particularly for Pu-239, as a considerable fraction of the specimen power comes from neutrons in this energy range.

Most of the present calculations were made with the AIM-VI code (Reference 8). The cross-section set used was basically an 18-group set developed by Los Alamos, with corrections applied to the thermal and epithermal regions. These cross-sections are further discussed in Part C of this section.

B. NEUTRON FILTER AND TEST SPECIMEN

It was decided to investigate a fast reactor fuel specimen which would be representative of future requirements. A PuO₂-UO₂ specimen with 25 w/o Pu-239 oxide and 75 w/o natural uranium oxide was chosen. The compositions chosen for a typical subassembly were 52.6 v/o fuel, 22 v/o sodium and 25.4 v/o stainless steel. This composition corresponds to that used for the test region in the design of the fast loop in the ATR. See Tables A-5 and A-6 (Case 14) of Ref. 1.

In keeping with the section of this report covering the requirements of a fast fuel irradiation facility, it was decided to investigate a single pin, a bundle of 19 pins and a complete subassembly with 91 pins in each of the thermal reactors. No attempt was made in the present study to produce an engineering design of these specimens, since only their physics aspects were of immediate interest.

In choosing a reference layout for the filter of FTR and EOCR, the fact that the filter material and test specimen should be separately removable from the reactor to allow for different burnup was considered. The specimen was enclosed in a stainless steel tube. The boron filter, consisting of stainless steel alloyed with 1.2 w/o B-10, was encased between this tube and an outer stainless steel tube. The filter was placed on the outside of the loop for the calculations on the 91-pin subassembly. The sodium coolant supply for these tubes and the filter was included with the stainless steel composition of the tubes for the calculations. A more detailed design of the filter assembly would take containment requirements into consideration. Any such engineer-

ing alterations would not significantly affect the basic physics performance of the filter.

The layout of the filter assembly and test specimen for the single-pin, 19-pin and 91-pin cases in ETR and EOCR are shown in Figs. 1, 2, and 3, respectively. The compositions of the separate regions are given on the same figures. The hexagonal can shown for the test specimen is tentative only.

The converter design being considered by AI for providing a fast-thermal coupled reactor in the SRE is shown schematically in Fig. 4. The fast test section in the center is surrounded by a buffer section of enriched fuel to adjust the energy spectrum and a decoupler section of natural uranium to absorb thermal neutrons. The thermal region, surrounding the decoupler section, would be of typical SRE-PEP configuration and composition. The modification of the SRE to accomplish this would consist in removing the central seven moderator cans of the PEP core and replacing them with the regions described above.

C. CROSS SECTIONS

The cross sections used in the multigroup AIM-VI code for the physics analysis were essentially the 18-group set compiled by C. B. Mills of Los Alamos Scientific Laboratories (Ref. 9). No attempt was made to accurately reproduce all the detailed physics characteristics of the thermal reactors in question. However, in those cases where the fast test sample performance would be grossly altered due to the use of improper cross sections, the necessary cross sections revisions were made. A discussion of these cross section revisions follows:

1. Fast Test Sample

For the cases where the fast test sample is placed in a loop with a boron filter, a large portion of the sample power is generated in the energy region of the Pu-239 resolved resonances. Thus, it is important that the resonance fission and capture cross sections of the test fuel pin materials (Pu-239 and U-238) be fairly accurate, taking into account energy and spatial self-shielding effects. The test sample composition used in this study is quite like that treated in a similar study involving the Advanced Test Reactor (ATR). Thus, the same resonance cross sections for Pu-239 and U-238 as used for the ATR study were used here. See Appendix A of Ref. 1.

2. Sodium Reactor Experiment (SRE)

The fuel subassembly design of the SRE-PEP core is such that there exists large thermal and resonance self-shielding effects. (The fuel pin is 0.5 inches in diameter). Further, the high operating temperature causes a signi-

REGION	COMPOSITION ATOMS/ $\text{CM}^3 \times 10^{-24}$					
	LI-238	PU-239	O	FE	Na	B-10
FUEL PIN	.01731	.00577	.04615			
SODIUM ANNULUS					.02252	
INNER WALL				.0483	.00973	
FILTER				.08212		.00523
OUTER WALL				.0483	.00973	

FIG. 1
CALCULATIONAL MODEL OF SINGLE PIN
ELEMENT AND TEST LOOP IN ETR AND EOCR

REGION	COMPOSITION ATOM/ $\text{CM}^3 \times 10^{-24}$					
	U-238	Pu-239	O	Fe	Na	B-10
TEST REGION	.00909	.00303	.02424	.0203	.00484	
INNER WALL				.0483	.00973	
FILTER				.08212		.00523
OUTER WALL				.0483	.00473	

FIG. 2
CALCULATIONAL MODEL OF 19-PIN
BUNDLE AND TEST LOOP IN ETR AND EOCR

REGION	COMPOSITION ATOMS/ $\text{CM}^2 \times 10^{-24}$					
	U-238	Pu-239	O	Fe	Na	B-10
TEST REGION	.00909	.00303	.02424	.0203	.00484	
WALL				.0483	.00473	
FILTER				.08212		.00523

FIG. 3
CALCULATIONAL MODEL OF 91 PIN
SUBASSEMBLY AND TEST LOOP IN ETR AND EOCR

REGION	COMPOSITION ATOMS/CM ³ X 10 ⁻²⁴					
	U-235	U-238	Pu-239	O	FE	Na
TEST REGION		.00909	.00303	.02424	.0203	.00484
BUFFER	.00303	.01169		.02944	.01360	.00386
DECOUPLER	.0001303	.01772		.0357	.008497	.002145

FIG. 4
CALCULATIONAL MODEL USED FOR 19
PIN BUNDLE AND CONVERTER IN SRE-PEP

ficant shift in the base of the Maxwellian distribution of the thermal flux. Neglect of these effects on the fission and absorption cross sections would entirely disrupt the analysis so that the resulting test sample performance values would have no real meaning. Thus, the thermal and resonance region cross sections were revised.

The thermal cross sections for all materials assume a Maxwellian distribution of the thermal flux with the most probable neutron speed corresponding to 0.60 ev (425 C). Self-shielding factors are applied based on the relative average fluxes obtained for the various regions of a typical cell. See Chapter III of Reference 10. Further, group 17 is made to be the thermal group (scattering into group 18 is made to be zero) because its energy nearly corresponds to the operating temperature of the reactor.

The major contribution to resonance capture in SRE is that due to U-238. Self-shielded cross sections over the resonance region of U-238 were obtained from an unpublished listing by C. B. Mills and G. Hansen, that gives self-shielded group cross sections for a range of values of effective scattering per atom. The effective scattering per atom was obtained by applying the method of Bell⁽¹¹⁾ to the cell geometry of the SRE. The temperature and self-shielding effects of the remainder of the reactors are not sufficiently large to warrant cross section revisions. The major error that is expected to result in test sample performance in these reactors due to the use of the unrevised 18-group cross section set mentioned earlier is expected to result from an incorrect U-235 fission cross section. The value actually used is in all cases too high since it is based on unshielded room temperature cross sections. The result would be an under-estimation of sample power.

3. Boron Filter

The unrevised 18-group cross section was also used for the boron filter so that some error exists in the thermal cross section due to the temperature shift of the Maxwellian flux distribution. However, the filter is in any case essentially black to thermal neutrons so that such an error will have little effect on overall reactivity or on the test sample power (essentially no power is generated in the test sample from thermal neutron fission).

D. POWER NORMALIZATION

The fast test sample power will be directly proportional to the thermal core power. It will be to the fast reactor experimenter's benefit to maximize this power. The thermal core power will be fixed either by the heat removal capacity of the reactor proper or by the maximum allowable power density in the core. These limits for the various cores of interest are listed in Table III. It should be noted that since the nominal thermal reactor designs are optimized, the total heat removal capacity for these basic designs is consistent

ith (i. e., almost equal to) the power obtained by assuming operation at the maximum power density (or heat flux) as given in Table III. However, insertion of the fast test sample loops in the various ways described in Part A of this Section results either in a change in the thermal core volume, the ratio of average to maximum power in the core, or both. If insertion of the fast test loop results in a reduction of the product of these two quantities, the resulting thermal core power will be lower than the reactor heat removal capacity since it would then be limited by the maximum allowable power density. If the product of the thermal core volume and the ratio of average to maximum power density remains the same or increases upon insertion of the fast test loop, the thermal core power will be identical to the heat removal capacity of the reactor proper.

CALCULATIONAL PROCEDURE

1. Introduction

All calculations were made on a one-dimensional cylindrical model of the reactor and experiments, using the multigroup diffusion theory code AIM-VI.

Because of the major cross-section changes in the thermal and epithermal regions required in the cross-section set normally used with the AIM code, the confidence in calculating thermal systems with it was not very great. It was necessary to establish a base case, consisting of a known critical configuration and composition, for each reactor and determine the k_{eff} . This value was then used to determine the fuel concentration or core size necessary for criticality.

The different calculations on the three types of reactors described earlier are presented below.

2. ETR

The reactivity base case was set up on the basis of a loading corresponding to the first set of experiments in ETR as given in Ref. 12. All subsequent ETR models were adjusted so as to give approximately the same reactivity value as obtained in the base case. (See Part B of this Section). To obtain this value with various fast test samples at the core center involved a change in core size at constant concentration of the materials, or a change in the uranium loading at constant core size. Although the preferred method of operation of ETR appears to be at a constant active core size of 49 elements, freedom to change the core size was assumed. The two methods of operation (i. e., constant core size or constant fuel loading) can result in a different ratio of power in the test sample to power in the active core as well as different core powers. Table IV lists the calculations made on the test reactors. In the first two cases for ETR the experimental regions present in the ETR

TABLE IV

LIST OF CALCULATIONS MADE FOR THERMAL TEST REACTORS

Case	Reactor	Test Region and Filter	Thermal Core			Fission Power, Mw
			Outer Rad, cm	Volume, liters	U-235 Conc, atoms/cc x 10 ⁻²⁴	
1	ETR	Single Pin Fig. 1	41.45	287	0.0000836	163
2	ETR	19 Pins Fig. 2	41.45	287	0.0000832	163
3	ETR	91 Pins Fig. 3	31.4	274	0.00011	163
4	ETR	19 Pins Fig. 2	22.45	142	0.00011	82
5	EOCR	Single Pin Fig. 1	38.9	430	0.00028	37
6	EOCR	19 Pins Fig. 2	36.9	382	0.00028	29
7	EOCR	19 Pins Fig. 2	47.9	511	0.00028	37
8	2 Region SRE-PEP	19 Pins Fig. 4	41.9	2970	0.0000245	13

were included in the geometry in order to obtain the outer radius. The active core volume is that of a homogenized mixture of 49 fuel elements and 6 control rods loaded with 58% of a fuel element loading. The uranium loading of the core was adjusted to give the required value of k_{eff} . For the last two cases the experimental regions and the control rods were assumed to be of the same material as the active core. The outer core boundary was then adjusted to give the required value of k_{eff} .

3. EOCR

The reactivity base case was calculated on the basis of information supplied by Phillips Petroleum Co. (Ref. 13). The three EOCR calculations given in Table IV were made by adjusting the thermal core boundary to give the same reactivity value as the base case. In cases 5 and 6 the only experimental volume included in the core was that of the fast test loop. In case 7 a separate experimental region was added within the core boundary to allow for typical thermal experiments in the remaining EOCR experimental regions.

4. SRE-PEP

The base case for this reactor was calculated from preliminary information from Atomics International (Ref. 6). One calculation was made with the reference 19-pin fast test sample placed in the center of the fast test region of the proposed two region SRE-PEP core (Case 8 of Table IV). Details of this two region figure concept was obtained from Reference 5 and a brief description is given in Fig. 4 of this report.

The enrichment of the buffer region was adjusted to give the same reactivity value as the base case, keeping the outer core boundary and thermal core fuel concentrations fixed. Since the spectrum of the fast region approximates that of a fast reactor, the power profile in the test bundle is essentially flat; thus, it was not necessary to perform an additional multigroup analysis for a single test pin.

F. SAMPLE PERFORMANCE

1. General

Once the reactor power is normalized as described in Part D, the fast test sample power at the central plane and the radial power distribution follow immediately from the multigroup analysis. These results for the various reactors and for the operating philosophies of interest are given in Table V. Also included in the table are some of the pertinent operating characteristics of the thermal core that result from operation with the fast test sample.

A discussion of these results for each of the various reactor types follows.

TABLE V

FAST FUEL SAMPLE PERFORMANCE IN THERMAL TEST REACTORS

Case	Reactor	Central Plane Sample Power				Core Fission Power, Mw	Central Plane Core Power Density		Number of Sample Pins
		Average Power Density, Mw/liter	Linear Power, kw/in.	Linear Power for Average Pin			Radial Power Density, Mw/liter		
				Average Pin kw/in.	Radial Max/Avg				
1	ETR	1.70	1.37	1.37	1.1	163	1.29	0.87	1
2	ETR	0.75	22	1.14	1.16	163	1.31	0.891	19
3	ETR	0.57	92	0.92	1.12	163	1.08	0.886	91
4	ETR	0.52	15	0.79	1.15	82	1.6	1.1	19
5	EOCR	0.38	0.31	0.31	1.07	37	1.73	0.210	1
6	EOCR	0.14	4.2	0.22	1.11	29	1.79	0.190	19
7	EOCR	0.15	4.4	0.23	1.11	37	1.78	0.178	19
8	SRE-PEP	0.071	2.1	0.11	~ 1.0	30*	1.2+	~ 0.01	19
a**	SRE-PEP	0.142	0.11	0.11	~ 1.0	30*	1.2+	~ 0.01	1
b++	SRE-PEP	0.16	4.5	0.24	~ 1.0	30*	-	-	19
c++	SRE-PEP	0.32	0.24	0.24	~ 1.0	30*	-	-	1

* Total reactor power

+ Refers to thermal core only

** Sample performance is obtained by proportioning the 19-pin results of Case 8

++ Based on design values given in References 5 and 7 (maximum performance core with 25 Mw in fast region)

Specific multigroup analysis was not performed for Cases a, b and c

2. ETR

As shown in Table V, the choice of the larger cores (Cases 1-3) results in core operation at the full fission power capacity of 163 Mw. This is due to the fact that even though the core volume has been slightly reduced, the presence of the loop has considerably flattened the thermal power distribution. The average homogeneous fuel concentrations necessary to maintain criticality are consistent with the presence of several low fuel density thermal experiments. Thus, the 163 Mw may be thought of as driving additional thermal experiments as well as the reference fast test loop.

The sample power depressions appear quite acceptable although the power levels are marginal for those three cases. Perhaps a more optimum filter choice would have been one of slightly lower boron concentrations, thus increasing the sample power at the expense of the low power depressions.

The small core (Case 4) utilizes a rather high U-235 concentration and hence must be thought of as omitting other thermal experiments or filling the thermal loops with high concentrations of fissile material in order to maintain criticality. The reference loop placed in such a small core results in a greater maximum to average power density. This fact together with the reduced core volume results in a much lower thermal core power (82 Mw), the power now being limited by the maximum allowable power density. The filter effect is about the same as for the large cores since the power shape in the sample is approximately the same. However, the sample power is lower for the case of the smaller core due to the lower average thermal core power density and the greater neutron leakage.

3. EOCR

The two size cores considered for the EOCR (Cases 5-7 in Table V) are actually quite similar. The only difference is that the larger core (Cases 5 and 7) assume the presence of four thermal experimental loops with a volume consisting of approximately 20% of the total thermal core (core plus thermal experiments). The smaller core assumes the same core fuel concentrations as the larger core but omits the thermal experiments. The similarity in sample performance for the two cases are reflected in the results of Table V. As in the case of ETR, it would profit the experimenter to decrease the filter concentration so as to increase the sample power levels which, as given in Table V, are below the required values. It is interesting to note that the relative sample power performance between the nominal size ETR and EOCR cores (Cases 2 and 7) reflects the difference in the power densities of the nominal thermal cores (Table III), and the fact that the fast removal cross-section of the EOCR is approximately $3/4$ that of the ETR. It is also of interest to note that operation of the smaller core at the maximum allowable power density results in a thermal core power level almost equal to the heat removal capacity of the reactor. This is due to the fact that insertion of the loop flattened

the core power density just enough to offset the smaller core volume necessary for criticality with the thermal loops removed.

4. SRE

The sample performance in the fast region of the two region SRE-PEP reactor as given in Table V (Cases 8 and a-c) is typical of a small low power fast reactor. The power profile is essentially flat and the sample power densities roughly correspond to operation of a fast core of 347 liters at power levels of 17 Mw (Cases 8 and a) and 25 Mw (Cases b and c). The fast region power level is essentially limited by the heat removal capacity (30 Mw) of the reactor proper. However, if the UO_2 of the 19-pin test sample contained fully enriched uranium, the sample power would increase by very nearly a factor of four over the values given in Table V, and still remain essentially flat.

VII. USE OF EOCR WITH LIQUID METAL LOOPS

Since sodium is considerably more compatible with organics than with water, the advantages of using sodium test loops in the EOCR as compared to the water cooled test reactors was investigated. With the present reactor arrangement, it appears that loops can be easily installed and the operation of such loops is much less hazardous in EOCR than in a water cooled reactor.

A. DESCRIPTION OF A TEST LOOP

A study to determine a possible loop arrangement has been made and is illustrated in Fig. 5. This loop is designed to provide the maximum space possible for the test fuel element. For testing smaller elements such as single pins, smaller diameter loops could be used.

The loop would enter and leave through the bottom of the reactor, through one of the penetrations already provided. It would consist of a 3-1/2-inch-diameter pipe inside of a 6-inch-diameter pipe return. The combined lines would enter the sub-pile room under the reactor through the side wall, turn up and enter the penetration seal. A spring loaded support for carrying the loop weight down to the chamber floor would be fastened to the 6-inch elbow. Inside the reactor, the supply line would rise to a level just below the core and terminate in a nozzle seat into which the test fuel element would be placed. The fuel element would consist of a lower nozzle section, a fuel section enclosed in a structural shroud, to support the fuel and contain the coolant flow, and a spring loaded handling head on the top extending up to a level just above the No. 1 spider. The 6-inch return line would also extend to the same elevation where it would be capped with a gasketed cover. A neutron filter in the form of a tubular sleeve of boron steel alloy would surround the fuel element inside the 6-inch return pipe and be suspended from a flange, located directly under the gasketed cover, by a tube perforated at its lower end so that the return coolant could pass outside as well as inside the neutron filter. The loop outside the reactor would contain a heat exchanger, a surge tank in which the pump would be located, a flowmeter and other instrumentation, including a sodium clean up system all contained within a shielded cubicle.

A sodium make up and storage tank would also be required which could be located underground and serve as a dump tank for the loop.

The sodium pipes in the subpile room would be shielded to permit access only during shutdown after a suitable decay period.

FIG. 5
ARRANGEMENT FOR FAST FUEL TEST LOOP

B. ASSOCIATED EQUIPMENT AND FACILITIES

A general arrangement of the additional facilities and equipment for servicing the test loops has been worked out and is shown in Fig. 6 and Fig. 7.

For handling the test fuel elements and the renewable neutron filters, a cask or casks would be required which would clamp onto the top of the loop with a gas tight seal. The cask would contain a two position lazy Susan type of handling mechanism to bring the fresh filter and/or fuel element to the loop and to remove the spent components.

The cask would be filled with argon with a self contained system of cooling.

It is possible to keep the cask filled with sodium, if decay heat is a problem, which probably would eliminate the need for cooling. The added complication however of dealing with the sodium at the seal of the cask with the loop and when discharging the fuel to the hot cell tend to favor gas cooling.

A hot cell would be desirable to examine the tested fuel elements and should be located underground next to the reactor building so that a simple transfer car could be used to move the cask from the reactor to a position where it could discharge through the roof of the hot cell into the cleaning and examination equipment. Otherwise some sort of transfer station to unload the cask into a shipping container would be required.

The cask represents the most sophisticated part of the support facilities and would probably be the most expensive. It would require an inert gas atmosphere, probably argon, to protect the sodium filled loop and to prevent the formation of oxides, etc., on the test fuel element after withdrawal and perhaps to provide cooling for the fuel test element. A cleaning chamber in the hot cell into which the cask would discharge the irradiated fuel test element using steam for cleaning, would be necessary before examination.

C. COOLANT COMPATIBILITY

The reaction of sodium and santowax is believed to be similar to the reaction of sodium and tetralin. The reaction is endothermic with hydrogen, carbon, methane and higher aliphatic hydrocarbons as the reaction products. These products do cause a pressure rise and have been demonstrated to cause plugging of coolant passages in sodium cooled reactors. The organic processing system for EOGR probably could be modified to handle any products that form due to leakage of sodium into the organic. This however is doubtful if the sodium pressure in the loops would be permitted to be higher than the organic coolant pressure for safety reasons. It, therefore, seems that a loop designer would either have to accept or design for a leak of organic into the sodium loop. The introduction of organic coolant into the sodium could cause

SECTION A-A
SEE FIG. 6

FIG. 7
FTL LOOP OF EOCR REACTOR
SERVICE FACILITIES

an increase in reactivity due to the high fissionable material density in the test fuel specimen. Since plugging and possible fuel melting could result, double containment may be required. Therefore, even though the organic is considerably more compatible chemically with sodium than sodium is with water, the precautions taken in loop design may not be much different for the organic cooled as compared to the water cooled reactor.

VIII. CONVERSION OF EOCR TO FTR

Two possible methods of converting the EOCR to a fast or intermediate spectrum test reactor have been considered. The first is a direct conversion in which a sodium or NaK coolant is substituted for the Santowax organic coolant and a fast core is accommodated in the present reactor vessel or perhaps a new vessel. This approach has been called the direct conversion of EOCR to an FTR. The other possible approach is to build a small (approximately 50 Mw) FTR directly adjacent to the present EOCR reactor building and use the present organic coolant system as the secondary system for the FTR. This approach is called the indirect conversion of EOCR to FTR.

A. DIRECT CONVERSION OF EOCR TO FTR

The problems associated with the direct conversion of EOCR to an FTR are presented in this section. These conclusions are based on various discussions and considerations given to the problem by the APDA staff. The information used to arrive at the conclusions stated is based on the data given in IDO-16820, "Experimental Organic Cooled Reactor Safety Analysis Report," and on discussions that took place during the visit of three APDA personnel to the EOCR site. Consequently, they are qualitative but correspond to the intent of the scope of the study.

1. FTR With Containment

Containment of large fast high power density reactors is at present generally regarded as a necessity. A core meltdown could lead to a supercritical arrangement of fuel, and the resulting nuclear excursion would give a large energy release. To contain the fission products (and plutonium in the case of plutonium-fueled reactors), a complete containment structure around the entire reactor and primary system is generally considered to be necessary. The lower part of this structure is in place before erection of the reactor vessel and primary system begins.

It is difficult to see how such a containment structure could be erected around the EOCR reactor vessel and primary piping without first removing the reactor vessel and piping and demolishing the existing plant. All that would remain of the existing facilities is the Administration building and possibly some experimental and instrumentation equipment such as the control room.

2. FTR Without Complete Containment

It may be possible to consider that complete containment of the entire primary system is unnecessary in view of the site location at the National Reactor Testing Station (NRTS), Idaho. If such is the case, the feasibility of converting the existing EOCR to an FTR depends on considerations such as space requirements, suitability of reactor vessel and piping, radiation, shielding and hazards problems associated with the heat removal systems, the possible uses for the organic coolant processing systems, and the general shielding problems. These problem areas are inter-related; in particular, the shielding requirements affect all the others. However, an attempt will be made to separate them for ease of discussion.

a. Space Requirements - A typical diameter of a reactor vessel for a fast reactor of approximately 500 Mwt is about 9 feet; for a fast reactor of approximately 50 Mwt, the diameter is about 6 feet. Both of these figures include space for an extensive radial blanket around the core. The existing EOCR vessel is 8 feet in diameter. A typical height difference between coolant inlet and outlet for a fast reactor is about 18 feet (500 Mwt) and 7 feet (50 Mwt). For EOCR it is 21 feet.

It is concluded that the EOCR reactor vessel could physically accommodate a fast reactor core. However, the shielding requirements for an FTR may be a limiting factor (see paragraph e below).

b. Reactor Vessel and Piping - The EOCR reactor vessel and piping are fabricated from carbon steel and designed for 335 psig and 850 F. The organic coolant inlet and outlet temperatures would be 500 F and 523 F, respectively, at 40-Mw operation. With sodium coolant, the corresponding temperatures would be 500 F and approximately 540 F at the same flow rate and power generation. At these temperatures, the steel would suffer no appreciable corrosive attack from the sodium nor would the NDT shift be to a temperature greater than 500 F, the minimum operating temperature, for an nvt of approximately 10^{20} ($E > 0.1$ Mev). However, there would be a ductility decrease, and this effect would have to be studied in more detail. At the operating temperature, the vessel would appear to have sufficient strength.

The integrated flux limitation of 10^{20} nvt ($E > 0.1$ Mev) is based on the carbon steel operating at 500 F. In paragraph e below, it is pointed out that 10^{22} is a more typical figure for a fast reactor vessel. The use of the present vessel would consequently necessitate about 100 annealings at approximately 750 F. Each annealing would require holding it at approximately 750 F for at least 5 hours. Operating the reactor at this temperature to avoid periodic annealings is, of course, ruled out by the strength limitation of the carbon steel and possibly by corrosion of the steel.

The conclusion is that although it is possible to use the present vessel and piping with an FTR, it would raise many operational difficulties because of the necessity to anneal the vessel periodically. For this reason, the conversion of the complete primary system to stainless steel would have to be recommended. The necessity to anneal would not then arise.

c. Heat Removal System - In EOCR a single heat removal system is used, the heat being dissipated in an air cooled heat exchanger. A single primary pump is used. It is a vertical, centrifugal pump of 25,000-gpm capacity made of carbon steel and equipped with a water cooled mechanical seal.

If sodium coolant were used, and if it were necessary to limit the temperature rise due to thermal stress limitations, a flow rate of 42,800 gpm at the same reactor power would be required. With existing technology, four sodium pumps would be necessary. These pumps would probably be sump-type with an oil lubricated mechanical seal. To accommodate four of them, the existing piping would require extensive modification. Assuming, however, from paragraph b that the piping must, in any case, be changed to stainless steel, these modifications would create no special problems. If, however, thermal stresses are not limiting and the sodium flow rate could be reduced to approximately 25,000 gpm and the consequent temperature rise of approximately 40 F accepted, the question arises as to the suitability of the existing pump for handling sodium.

The main problem would be created by the water cooled mechanical seal. If it could not be adapted to oil or other suitable coolant, the pump probably would be unavailable for sodium service.

Furthermore, the shielding requirements of the existing system containing sodium would be greatly increased, as pointed out in paragraph e. Although the present heat exchanger could possibly be used, the difficulty of shielding it adequately, because of the radioactivity of the coolant coupled with the danger of radioactivity release in the event of a fire, would make its use inadvisable.

d. Coolant Processing Systems - It may be possible to use the superfluous piping of the organic processing system for sodium processing (oxide removal, purity control). The steam tracing on the existing lines would then present a hazard that would have to be evaluated carefully. Hot organic could perhaps be used in place of the steam, or electrical heating installed. If organic could be used, the process system would be needed to purify the organic since it will suffer some thermal and radiation decomposition.

e. Shield Analysis - The shielding problems in a fast reactor are greater than in a thermal reactor, mainly because of the higher neutron leakage, harder spectrum and high coolant activity with a liquid metal coolant. The first two considerations mean that a fast reactor shield must contain a

large amount of moderating material, with both heavy and light materials, followed by a neutron absorbing material and the usual concrete biological shielding. The high activity of sodium coolant (higher by a factor of 10^6 than organic coolant) adds to the shield problems, especially in the upper end of the reactor vessel and in the coolant piping. The main problems are discussed below:

(1) Adequacy of the Top Head Shield Inside the Reactor Vessel

- In EOCR neutron and gamma ray shielding is provided by the 18 feet of coolant and top head. However, in a sodium cooled fast reactor, the sodium would provide little neutron shielding and would furthermore create a high gamma-flux because of its induced activity (approximately 0.01 c/cc). Therefore, an axial vessel shield plug that is approximately 10 feet thick and made of steel, graphite, and boron carbide would be necessary to reduce the dose rate at floor level to tolerance limits.

(2) Adequacy of the Top Shield Outside the Reactor Vessel -

In EOCR a biological shielding of serpentine aggregate concrete blocks is placed in the annular space between the reactor vessel and the upper cavity liner. For the same reasons as in (1) above, this shielding is inadequate with a fast reactor core and would have to be replaced by approximately 10 feet of steel, graphite and boron carbide.

(3) Adequacy of the Sub-Pile Room Shielding -

In EOCR the sub-pile room is shielded by 9 feet of organic coolant and the vessel bottom head, augmented by a 16-inch-thick concrete ring and 8-inch-thick steel ring. This shielding allows access to this room after reactor shutdown, though not in operation. It would be highly inadequate even during shutdown if a fast reactor core and sodium coolant were used. As in (1) above, an axial shield of steel, graphite and boron carbide would somehow have to be inserted at the bottom of the reactor vessel. The feasibility of doing this is much more questionable in the lower reactor vessel than in the upper reactor vessel.

(4) Adequacy of the Radial Biological Shield -

In EOCR there is a radial thermal shield of approximately 6 inches of steel between the core and vessel and a 73-inch-thick magnetite concrete shield outside the vessel. If this thermal shielding were increased to about 18 inches of steel, and the present vessel replaced by one of stainless steel, fast neutron damage to the vessel from a fast reactor core could be ignored. It would take about 20 years at 100 per cent load factor to reach 1×10^{22} nvt ($E > 0.1$ Mev).

The 73-inch concrete biological shielding would still be capable of attenuating the neutron and gamma flux from a fast reactor core to somewhere near tolerance limits inside the test cells. However, it is highly doubtful that the concrete shield integrity could be maintained under the high flux. Even if optimum shielding materials (boron steel, boron carbide, etc.) were used inside the vessel in place of a blanket and thermal shield, the inci-

dent neutron flux on the inner concrete face at 40 Mw would be approximately 8×10^{11} n/cm²/sec. This would cause overheating, water loss, irradiation damage and cracking of the concrete.

(5) Shielding During Fuel Handling - The need for a large upper vessel shield plug with a fast reactor core complicates the fuel handling problem. Three possible methods could be used for fuel handling.

In the first method, a gas bag would be placed over the reactor and the plug removed. All handling on the core would then be done remotely. The method planned for EOCR (addition of nonradioactive coolant to the existing coolant in an extension tank, and portable radial shielding) is not feasible because of the excessive requirements for portable shielding and the mixing of highly active sodium with the fresh sodium.

A second method would be to install a fuel handling device in the plug. It would be a complicated device since it would have to provide for both rotational and reciprocating movements.

The third method would be to employ an inert gas enclosure above the reactor into which the plug could be removed. The fuel and experiments would be handled remotely, or a hot cell with windows could form the inert gas enclosure.

(6) Adequacy of Auxiliary Area Shielding - In general the shielding in auxiliary areas would be adequate with a fast reactor core, except for piping containing radioactive coolant. The biological shielding required for primary sodium coolant is approximately 6 feet of high-density concrete or its equivalent and about 4 inches of thermal shield plate as an inner liner on the concrete. Also, the piping for the sodium cover gas (usually argon) requires shielding, since argon becomes moderately radioactive. A further shielding difficulty with a fast reactor core is the increased streaming of neutrons along coolant and gas lines, cable ducts, etc.

(7) Summary and Conclusions - The existing shield design for EOCR is wholly inadequate for a fast reactor. For all practical purposes, the present biological shielding would have to be removed and a new one erected. Even then considerable shielding would be necessary inside the reactor vessel, and the axial shielding in the vessel would become complicated.

It is APDA's opinion that extensive modification of the present EOCR shield design would be required to convert the reactor to a fast test facility. In some instances it may not be possible to construct the necessary shielding within the present reactor building.

B. INDIRECT CONVERSION OF EOCR TO FTR

The major disadvantages of the direct conversion of EOCR to an FTR is the lack of a containment building, the major changes required in the reactor shielding, the changes required in fuel handling, and the high radiation level and hazard associated with the air heat exchanger. All of these objections can be overcome by using the present organic primary system as the secondary coolant for a sodium cooled intermediate or fast reactor that would be located in a containment building immediately adjacent to the present EOCR reactor building. This arrangement is shown in Figure 8.

Assuming that sodium or NaK is used as the coolant for the FTR, the sodium pipes would be placed in the shielded tunnel presently used as a water storage pool. They would connect to the sodium-to-organic heat exchanger which would occupy the space now occupied by the EOCR reactor vessel. A more detailed study might indicate that the present reactor vessel can be used for this purpose merely by replacing the EOCR core with internals of the sodium heat exchanger. It is expected that the shielding in this area is adequate to handle the sodium gamma activity. The organic system would remain intact and would be reprocessed to remove any corrosion products, thermal or gamma radiation decomposition products. It probably would be necessary to design the sodium-organic heat exchanger and sodium pipe tunnel as an integral part of the fast reactor containment. The reactor room could be converted to an experiment assembly area for testing loop experiments before insertion in the FTR. The basement room would be used for fast test loop experiments such as those noted in Table I, each occupying a separate shielded cubicle. The loop piping would be run down a portion of the shielded tunnel now called the water storage pool but separately shielded from the primary sodium piping.

It may be possible to utilize a large fraction of the control room for the FTR and associated loop experiments. Other auxiliary systems on the site, such as emergency power, probably could also be used.

It is anticipated now that the two cooling systems would be designed to prevent any leakage of organic into sodium to prevent an increase in reactivity of the reactor due to the introduction of hydrogen, and to prevent fouling that results from the reaction of sodium with organics. Although the reaction is endothermic, the reaction products (hydrogen, carbon, methane, and higher aliphatic hydrocarbons) do create a pressure rise and do cause plugging of fuel elements, cf. SRE fuel damage due to tetralin leak. It is anticipated that the present organic purification system can be modified to handle the decomposition products that result from a leak of sodium into the Santowax. A test to demonstrate that organic coolant cannot leak into the sodium and to investigate how purification of the organic coolant would be handled would be a necessary test in the next phase in designing this conversion of EOCR to FTR.

FIG. 8
TEST FACILITY FOR FAST FUEL USING
PRESENT EOCR REACTOR AS HEAT EXCHANGER

If the organic coolant can be operated at the 800 F design point without serious thermal decomposition, it is felt that no serious limitation would be imposed on the average maximum sodium outlet temperature from the fast reactor. On the other hand, if the limit is lower, a limitation might be imposed on the test reactor. This could easily be overcome in the design of the intermediate heat exchanger by increasing the thermal resistance of the heat exchanger in the region of the high temperature sodium. Consideration would be given to the use of a double-tube heat exchanger, probably with mercury between the tubes. Such an arrangement would make it possible, by adjustment of the mercury level, to vary the temperature difference between the primary and secondary coolants. The fact that the control of level is slow is not important since the decomposition of the organic is a function of time at high temperature. This design would also minimize the safety problem of organic coolant entering the sodium. This problem could be further reduced by appropriate layout of the heat exchanger. If, however, the problem remained in spite of these precautions, it could be eliminated by substitution of NaK for the organic coolant.

In designing the fast or intermediate test reactor, it may be desirable to completely isolate both physically and thermally the loops from the primary coolant system. In this case there would be little need to operate the sodium coolant at a high temperature.

To improve the test reactor fuel burnup and economics, it probably would be desirable to operate with a low primary coolant temperature.

The general approach of considering several small fast test reactors, and specifically to utilize the buildings and cooling system of EOCCR, may prove to be the most economical approach to obtain the needed fast fuel testing facilities. Further study is required to determine the relative economics of a completely separate small FTR as compared to a small FTR that utilizes the EOCCR. Without such a study, it is difficult to show that imposing the restriction of using the present organic cooling system of EOCCR may not result in a more expensive FTR. In addition, further study is required to show the relative schedule and economics of several small fast test reactors as compared to one large FTR.

IX. FAST FUEL TESTING IN THE FERMI REACTOR AND EBR-II

A large amount of the test space required in Table I is associated with proof testing subassemblies of a final design of fuel element and, to obtain by repeated tests, statistical information on the performance of the fuel element under proper environmental conditions. These requirements for proof testing could be met by conducting these tests in Fermi or EBR-II, if these facilities were available. The neutron flux and coolant conditions in these reactors are very desirable and many full size subassembly positions could be made available for fuel proof testing. In addition, uninstrumented capsule tests could be run if the reactivity associated with the fuel sample is small so as to limit the hazard of the test.

It must be kept in mind however that the Fermi reactor and EBR-II were not designed with the needs of the fast reactor fuel testing program in mind. They, therefore, contain few auxiliary design features or equipment with the flexibility needed for handling fuel specimens, loops, etc. The main difficulties of using these reactors as fast fuel test reactors for fuel failure testing are:

1. All tests would have to be terminated long before the point of failure or even excessive damage is reached. Excessive contamination of the existing coolant cannot be tolerated.
2. Inserting one or more loops into the existing reactors cannot be considered because they were built primarily for other purposes, and the needed flexibility was not built into either reactor.

These two difficulties preclude the use of these reactors for fuel failure testing, at least as long as they are required to serve their original purposes. However the possibility remains of using existing subassembly positions for fuel proof testing purposes, or even for uninstrumented capsule tests. In both cases the following difficulties must be faced:

3. The fuel specimens would have to conform to the fuel external geometry not only in specimen size, but also with appropriate end fixtures. This is necessary since the manipulation of the specimen would have to be done with the existing fuel handling mechanism.
4. The present construction precludes the installation of additional instrumentation so that only a limited number of parameters can be determined.

Some possibility exists of inserting specially instrumented capsules or subassemblies into the reactor, but this cannot be fully established until a

proper design effort is made. Until such designs are produced, and approved by the organizations responsible for the operation of these reactors and any government agency with regulatory authority, plans for instrumented subassemblies cannot be made.

If these conditions can be accepted, then proof testing of subassemblies can be carried out in both reactors. The information obtained would be time-averaged neutron flux, total power generated and burnup of the subassembly; the fuel and clad operating temperatures would have to be calculated, however. The information from the uninstrumented capsule experiments would be the same, with the addition of approximate time and mode of failure in the case of tests carried to failure.

END

REFERENCES

1. Blessing, W. G. et al "Conceptual Design of an In-Pile Package Loop for Fast Reactor Fuel Testing" Atomic Power Development Associates, Report No. APDA-144, July 1961.
2. Correspondence of Potential Liquid Metal Fuel Experimenters:
 - a. Letter of W. H. Jens to other experimenters, ATD-2205 etc. January 15, 1963.
 - b. Letter of K. Cohen to W. H. Jens, January 29, 1963.
 - c. Letter of D. T. Eggen to W. H. Jens, 63AT731, January 28, 1963.
 - d. Letter of J. H. Kittel to W. H. Jens, January 22, 1963.
 - e. Letter of G. A. Sofer to W. H. Jens, SEEO-52, January 30, 1963.
 - f. Letter of E. S. Sowa to W. H. Jens, January 24, 1963.
3. Minutes of meeting at COO on ATR Liquid Metal Loop Facilities, January 3, 1963.
4. Wolfe, B. "Fundamental Factors of Test Reactor Design" Test Reactors Meeting for Industry, IDO-16520, May 13-15, 1959.
5. Private communication from D. T. Eggen, Atomics International on converter design for the two-region SRE-PEP.
6. "Proposal to Increase the Power Level of the SRE" - NAA-SR-Memo-6837. November 1961.
7. Private communication from D. T. Eggen, A.I. on proposal to convert SRE to a fast-thermal coupled reactor.
8. Platt, H. P. and Baller, D. C. "AIM-VI; A Multigroup, One Dimensional Diffusion Equation Code" NAA-SR-4694, March 1, 1960.
9. Mills, C. B. "Neutron Cross-Sections for Fast and Intermediate Reactors" LAMS-2255, 1959.
10. Fillmore, F. L. "Two-Group Calculations of the Critical Core Size of the SRE Reactor" NAA-SR-1517 (Revised) January 15, 1959.
11. Zweifel, P. F. "Neutron Self Shielding" Nucleonics, 18, September 1960.
12. Bush, D. D. et al "Engineering Test Reactor: Engineering Design and Safeguards Report," IDO-24020, July 1956.
13. Private communication R. Goins, Phillips Petroleum Co., Idaho Falls.
14. Lazar, A. H. and Gordon, R. H., "Design Criteria for a Fast Flux Liquid Metal Loop in the Advanced Test Reactor," IDO-24041, July 1962.