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**IRRADIATION TESTING NEEDS AND FACILITIES
FOR FAST REACTOR FUEL AND MATERIAL DEVELOPMENT**

MARCH 1, 1966

RICHLAND, WASHINGTON

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IRRADIATION TESTING NEEDS AND FACILITIES
FOR FAST REACTOR FUEL AND MATERIAL DEVELOPMENT

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NOTICE: PRELIMINARY REPORT

This report contains information of a preliminary nature prepared in the course of work under Atomic Energy Commission Contract AT(45-1)-1830. This information is subject to correction or modification upon the collection and evaluation of additional data.

PACIFIC NORTHWEST LABORATORY
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This study was conducted to assist in continuing reviews of design criteria for the Fast Test Reactor (FTR). It was attempted to envision the problems associated with fast reactor fuel and materials development, and estimate how existing and proposed facilities might be utilized in solving these problems. The role to be played by the FTR was examined in terms of its currently proposed design. It is expected that these studies will continue as FTR design criteria are refined.

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IRRADIATION TESTING NEEDS AND FACILITIES FOR FAST REACTOR FUEL AND MATERIALS DEVELOPMENT

INTRODUCTION

The economic success of fast breeder reactors depends on developing reactor fuels and materials capable of operating to long exposures at high specific power in high, hard neutron fluxes. In fast reactor systems, environmental conditions are considerably more severe than those encountered in thermal reactor systems. However, the incentives for developing fast breeder reactors are well accepted. Fast reactors will conserve nuclear resources by converting fertile materials (U^{238} and Th^{232}) to fissionable Pu^{239} and U^{233} . Additional economic incentives arise from such features as the high thermodynamic potential of liquid metal cooling, the wide choice of nuclearly suitable materials for core structure materials, and the wide choice of permissible fuel materials.

A fast reactor is not simply an advanced type of thermal reactor. The operating conditions imposed on the fuel require, if not new technologies, considerable advances in existing technologies. These conditions are beyond the range which permit prudent extrapolation from existing experience. Additionally, the combination of high flux, hard spectrum, and long exposure introduces new or secondary and tertiary reactions which are not encountered in thermal systems. As a consequence, fuel for fast reactors will not become available as an evolutionary advance stemming solely from improvements in thermal reactor fuels. A deliberate and bold effort involving intensive irradiation testing in the environments encountered in fast reactors will be required to provide satisfactory fast reactor fuel and materials.

Several estimates have been made of the amount of irradiation testing required to develop fast reactor fuels and materials. Although these estimates vary in the amount of testing deemed necessary, all of the estimates agree that existing facilities are inadequate. To a great extent, the FTR, now in design, will provide much of the testing space required, but extensive use of existing facilities will also be necessary to meet testing needs.

This report broadly describes the irradiation performance information needed to develop fast reactor fuels and materials. To provide an appreciation of some of the factors which may influence fast reactor performance, a discussion of the effects of fast and thermal neutron environments on fuel and structural materials is included. This is followed by a general description of the types of irradiation tests required in a fast reactor fuel and materials development program. The last section of the report includes: (1) performance and testing criteria as specified by the Atomic Energy Commission⁽¹⁾ and by the four 1000 MW_e Design Studies;⁽²⁾ (2) a description of the testing capabilities of fast reactor facilities in place and proposed; (3) a comparison of these capabilities with the criteria; and (4) a description of thermal reactor facilities which may be used in the fast reactor fuels and materials testing program.

SUMMARY

In fast reactors, fuel and structural materials are exposed to environmental conditions considerably more severe than encountered in thermal reactors. However, practically all existing data on irradiation damage to fuel and structural materials have been obtained in thermal reactor environments. Thus, the extent of damage produced in these materials in the high, hard spectrum fluxes, typical of fast reactors is not known.

In both fast and thermal reactors, damage to fuel materials is derived principally from fission recoils and fission product accumulation. The effect of these fission-related events on fuel performance can be considerably different in a fast reactor and in a thermal reactor. In a thermal neutron flux, the flux is depressed or attenuated by the fuel material and as a consequence the thermal neutron flux in the outer region of the fuel material may be many times greater than at the center of the fuel. Thus, the greatest number of fissions (highest heat generation and burnup rates and greatest concentration of damage) occurs in this outer region. In a fast neutron flux, no significant flux depression occurs; consequently, heat generation, burnup rates and frequency of damage sites are uniform throughout the fuel. Additionally, equivalent fuel pins irradiated at the same heat generation rate in a fast flux and in a thermal flux will have different centerline temperatures and different temperature gradients. Thus, fission product migration, fuel relocation, fission gas release, and swelling in the fuel will be significantly different.

Extensive irradiation testing of fast reactor fuel materials and fuel elements in fast reactor environments is necessary to establish fuel performance characteristics. Performance factors which require investigation are:

- Fuel pin swelling
- Fission gas release characteristics
- Fuel slumping
- Plutonium and fission product migration characteristics
- Fuel structural changes
- Fuel-cladding compatibility
- Failure behavior

- Transient behavior
- Fabrication process and design variables
- Mechanical stability of fuel pins and subassemblies.

Although the nuclear events which produce radiation displacement damage in structural and cladding materials are essentially the same in both thermal and fast reactors, the frequency with which these events occur and the total number of events which occur in a given time can be orders of magnitude greater in a fast reactor. For example, cladding doses in fast reactors will be typically between 10^{23} and 10^{24} n/cm² and will be obtained in fluxes of about 1×10^{16} n/cm²-sec. In thermal reactors, typical cladding doses are 10^{20} to 10^{21} n/cm² and are obtained in fluxes of about 1×10^{14} n/cm²-sec. It is this difference in degree which leads to uncertainties regarding the irradiation performance of these materials.

Radiation damage to structural and cladding materials may be produced by (1) neutron bombardment, in which atoms are displaced in the lattice structure, and (2) neutron reactions which create solid impurities and gas atoms through transmutations. Generally, such damage manifests itself as embrittlement and loss of ductility.

Radiation damage resulting from neutron bombardment can be a function of neutron flux. In the high flux encountered in fast reactors, displacement rates could be sufficiently high to cause dynamic mixing of atoms in a crystal lattice. Also, diffusion processes may be enhanced by the large number of excess vacancies created by irradiation in high fluxes.

Transmutations which involve (n, α) and (n, p) reactions promote damage to the mechanical properties of structural materials through the formation of helium and hydrogen atoms in the material. Helium atoms are formed as a result of (n, α) reactions by fast neutrons and also by (n, α) reactions in B^{10} by thermal neutrons. The extent of damage caused by the helium depends upon the distribution of the gas in the material. In a thermal reactor, the distribution of (n, α) produced helium in structural and cladding materials can be markedly different than in a fast reactor.

The neutron (n, p) reactions which are the principal sources of hydrogen have effective energies greater than 4 MeV. Therefore, the thermal component of the flux has no significant effect on hydrogen production in structural and cladding materials. The fast fluxes in thermal reactors are not sufficient to cause enough (n, p) reactions to significantly affect the mechanical or physical properties of these materials during their normal in-reactor lifetimes.

Structural and cladding materials must be irradiated in typical fast reactor environments to assess the radiation damage caused by exposure to high, hard spectrum, neutron fluxes. Some tentative information on irradiation performance can be obtained from simple screening tests in which pre- and postirradiation measurements of alloy properties are made. More elaborate tests are required to measure alloy property changes during irradiation.

In spite of their deficiencies, thermal reactors will continue to be used extensively for fast reactor fuel and materials testing simply because adequate fast reactor test facilities are not available. Thermal reactor facilities can be used for screening tests and to provide data for correlating thermal reactor experience to fast reactor service conditions. Although these facilities have fluxes, coolants, and temperatures which are not representative of fast reactors, they do provide (1) large volumes of test space capable of receiving instrumented tests, (2) relatively high burnup rates, and (3) loops for failure behavior tests.

Reactors which may be used for the bulk of the thermal irradiations of fast reactor fuels and materials are the MTR, ETR, GETR, SRE, EOCR, and ATR. (Irradiation facilities in these reactors are described in detail in Section IV of this report.) In some cases modifications have been proposed to provide a more desirable neutron energy spectrum or a quasi-fast reactor environment.

Existing U.S. fast reactor facilities (EBR-II and Fermi) are inadequate to meet the testing requirements for developing fast reactor fuels and materials. Although these facilities have a potential for irradiating a large number of subassemblies (10 to 20) in flowing sodium and in a fast neutron flux they

are seriously limited in providing for fast reactor fuels and materials testing needs in one or more of the following ways:

- High flux
- Test specimen size
- Restricted capability for accommodating instrumented tests
- Lack of closed loop facilities
- Necessity for encapsulating test fuel pins
- Restricted control of sodium coolant purity, temperature, and flow
- Lack of facilities for examining and recharging irradiated fuel assemblies
- Inability to significantly raise temperature of the coolant to an individual subassembly above that of the reactor coolant
- Inability to test for failure behavior
- Inability to test vented fuels in flowing sodium environment.

The FTR is being designed to meet criteria for fast reactor fuels and materials testing as specified by the AEC-LM-FBR Criteria⁽¹⁾ and by the four 1000 MW_e Design Studies.⁽²⁾ Specified powers, coolant temperatures, and fluxes will be representative of fast breeder reactors. Closed loops will provide for failure behavior studies and other high risk tests, special environments, and instrumented tests. Test sections in open and closed loops will be sufficiently large to accommodate prototypic fuel subassemblies. Open and closed loops will be equipped with instrumentation for routinely monitoring flow, temperature, and pressure. The FTR design provides for sufficient reactivity control and safety features to compensate for wide variations in core nuclear characteristics. Capabilities for postirradiation viewing, measuring, and manipulation of test assemblies in both wet and dry cells are included as part of the reactor facility. In addition, a critical facility will be available for assessing the effect of each core loading on reactor safety and performance.

IRRADIATION TESTING REQUIREMENTS
FOR FAST REACTOR FUEL AND MATERIAL DEVELOPMENT

A major effort in the development of fast breeder reactors will have to be applied to irradiation testing of fuel and structural materials, for it is in the area of irradiation performance in fast neutron fluxes where major uncertainties exist. Specifically, concern regarding fuel and structural material performance stems from the high, fast fluxes to which the materials are subjected. Additionally, these materials will be typically exposed to higher temperatures and to potentially more reactive coolants than encountered in thermal reactors. Operating conditions representative of fast and thermal power reactors are summarized in Table I.

TABLE I
TYPICAL OPERATING CONDITIONS
FOR FAST REACTORS AND THERMAL REACTORS
 (For Oxide Fuel)

	<u>Fast Reactors</u>	<u>Thermal Reactors</u>
Flux, Total, n/v	$0.3 - 1.0 \times 10^{16}$	$\sim 10^{14}$
Fraction with $E > 0.1$ MeV	0.6 - 0.8	0.3 - 0.4
Burnup, MWd/ton	100,000-200,000	10,000-20,000
Cladding Dose, n/cm ²	10^{23} - 10^{24}	10^{20} - 10^{21}
Specific Power, W/g fiss. atoms	500-1200	~ 1000
Median Fission Energy	100-500 keV	~ 0.1 eV
Coolant Outlet Temperature, °F	800-1000	500-600

Practically all existing performance data have been obtained from thermal reactor experience. Irradiation effects in fast reactors can be so different from those in thermal reactors that extrapolation of thermal reactor experience to fast reactor service conditions is at best provisional. As a consequence, intensive study under fast reactor conditions of all aspects of fuel and materials technology is considered essential to the development of economical and safe fast breeder reactors.

FUEL MATERIALS AND FUEL ELEMENTS

Irradiation Effects in Fast and Thermal Neutron Fluxes

Irradiation damage in fuel material is due principally to fission recoil and fission product buildup. This damage can manifest itself by any of several different ways: (1) deterioration of structural properties, (2) swelling of fuel, (3) distortion of fuel pins, (4) stressing of the cladding, and (5) cladding failure. The damaging mechanisms are the same in both thermal and fast neutron fluxes. However, a significant difference exists in the distribution of irradiation damage in fuel materials irradiated in the two types of fluxes. This difference stems from the depression or attenuation of the thermal neutron flux by the fuel material itself. (Such flux depression does not occur to any significant extent in a fast flux.) As a result of this flux depression, the thermal neutron flux in the outer region of the fuel material may be several times greater than at the center of the fuel. Thus, the greatest number of fissions (also the highest heat generation rate and the most rapid burnup of the fissile atoms) occurs in this outer region (see Figures 1 and 2). As a consequence, the site of greatest damage (i.e., fission recoils and fission product accumulation) is in the outer region. As irradiation proceeds the fissile atoms in the outer region of the fuel are consumed and the location of greatest heat generation rate and fuel damage moves to the center. In addition, as fissile atoms are consumed, the number of fissions (and the heat generation rate) decrease. Thus, in a thermal neutron flux, heat generation rate and burnup rates are not uniform across the fuel; nor are they constant throughout the lifetime of the fuel.

In a fast neutron flux, no significant flux depression occurs and fission rate and associated fuel damage is essentially uniform throughout the fuel material. Also, the heat generation rate may remain constant or even increase (due to bred-in fissile atoms) throughout the lifetime of the fuel.

In mixed oxide fuel, microstructure, stoichiometry, fission product migration, and fission gas release are all sensitive to temperature and temperature gradients, and all have an effect on fuel performance. For a given heat generation rate in the fuel, the flux depression associated with fuel testing in thermal reactors produces a different temperature distribution than is obtained

FIGURE 1

Power Density Profile in a Fast Reactor Fuel Irradiated
in a Thermal Flux

Fuel	UO ₂ -20 wt% PuO ₂
Fuel Diameter	0.250 in.
Fuel Density	87% TD
Clad	304-SS
Clad Thickness	0.012 in.

FIGURE 2

Burnup Profile for Fast Reactor Fuel Irradiated
in a Thermal Flux

Fuel	UO ₂ -20 wt% PuO ₂
Fuel Diameter	0.250 in.
Fuel Density	87% TD
Clad	304-SS
Clad Thickness	0.012 in.

in a fast flux. Central temperatures initially would be much lower in a thermal flux than in a fast flux and the thermal gradients would not be as severe. Therefore, fast fuel pins must be irradiation tested in a fast (rather than thermal) flux to obtain reliable performance data. It is not possible with the data presently available to assess the relative severity on the fuel of fast flux testing conditions as compared to thermal flux conditions.

The aforementioned differences in temperature, temperature gradients, and damage sites obtained in fast reactors, as compared to thermal reactors, will also affect the stresses imposed on the cladding material. It is doubtful that the stresses imposed on the cladding by the fuel damage concentrated in the outer area of the fuel (as obtained in a thermal flux) are comparable to the stresses obtained in a fast reactor. Additionally, temperature gradients influence fission product mobility, which in turn, can affect fuel-cladding compatibility and even mechanical properties of cladding.

Fuel pins irradiated in a thermal reactor flux experience a fuel burnup-to-cladding dose ratio which is not representative of the ratio obtained in a fast reactor flux. For a given fuel burnup, the cladding dose in a fast reactor may be orders of magnitude greater than in a thermal reactor. For example, a stainless steel clad $\text{UO}_2\text{-PuO}_2$ fuel was irradiated in the GETR to 100,000 MWd/tonne in a flux of 4.5×10^{13} n/cm²-sec. The fast component ($E > 0.18$ MeV) of this flux was 1.5×10^{13} n/cm²-sec. The cladding dose was 1.1×10^{21} thermal and 0.36×10^{21} fast or about 1.5×10^{21} n/cm² total. Such a fuel pin irradiated in a representative fast breeder reactor would typically be exposed to a flux of 3×10^{15} n/cm²-sec ($E > 0.18$ MeV) - a fast flux intensity about 200 times greater than that in the GETR. In achieving the same fuel burnup of 100,000 MWd/tonne in this fast reactor, cladding dose would be 2×10^{23} n/cm². Further, some damage mechanisms in the cladding are energy sensitive, thus the effective increase in damage could be even larger than that indicated by the greater neutron dose. With such a difference in cladding doses, it would appear imprudent to extrapolate thermal reactor testing data to fast reactor conditions.

Fuel and Fuel Element Component Testing

Irradiation testing of fast reactor fuels logically starts with establishing basic behavior of fuel materials and culminates in irradiating prototype fuel assemblies. In addition, transient (over-power) tests to ascertain safety characteristics of the fuel are essential for completely assessing fuel performance.

Basic Fuel Behavior

Studies of the basic behavior of fuel in fast reactor environments are necessary to develop fast reactor fuel element designs. The type of information required includes:

- Fission gas release rates as a function of fuel burnup, fuel burnup rate, and operating temperature.
- Fission product migration as affected by neutron flux, temperature, and temperature gradient.
- Fuel swelling as a function of fuel burnup, operating temperature, and specific power. These studies must be carried out on both restrained and unrestrained fuel to determine the forces required to minimize the fuel swelling and to determine the consequences of restraining the swelling.
- Effect of burnup and neutron exposure on such basic physical and mechanical properties of the fuel as thermal conductivity, thermal expansion, mechanical strength, and ductility.
- Chemical behavior of fuel after high burnups.
- Migration of plutonium and/or uranium within the fuel element.
- Solid-state interaction of the fuel with the cladding.

This information, for the most part, can be obtained using relatively small, instrumented fuel specimens. Small, closed loops which are readily accessible and capable of accommodating complex instrumentation can provide the type of irradiation facility needed for most of these studies. Fuel swelling data can be obtained in small, open loops. To simulate fast reactor conditions, the irradiations should be performed in fluxes between 5×10^{15} and 1×10^{16} n/cm²-sec.

Fuel Pin Testing

Data on basic fuel behavior provide a basis for fuel pin design. Successful fuel pin designs are obtained when the appropriate material, design, and process variables are selected and combined in a manner to adequately meet service conditions. Thus, fuel pin testing must provide for detecting the effect on fuel performance of not only these variables, but also the interactions among these variables.

Core-Cladding Interactions. Interactions between the candidate fuel and cladding materials should be determined. Screening tests can be performed in thermal reactors. These tests might effectively demonstrate only prototypic interface temperature effects. However, because of the different temperatures, temperature gradients, fission product distribution, cladding damage rate, and fluxes obtained in fast reactors, fuel testing in a fast reactor environment is necessary to fully characterize core-cladding interactions.

Fission Gas Accommodation. The accumulation of fission gas in high burnup fuels may lead to failure through excessive straining of the cladding. A satisfactory fuel pin design must provide means for preventing fission gas pressure from distorting or rupturing the fuel pin. Inordinately high fission gas pressure in fuel pins can be prevented by providing a fission gas reservoir or by venting the fission gas as it is formed. Fission gas release is dependent on burnup and on fuel temperature which in turn is spectrum-dependent. Fission gas migration rates are spectrum-dependent via the spectrum effect on temperature gradients and, probably to a lesser extent, on diffusion rates induced by high concentrations of vacancies.

Design and location of fission gas reservoirs must be verified using prototypic fuel pins. Venting of the fission gas to the coolant or to an out-of-reactor reservoir affects reactor operation beyond the usual regime of the fuel element (e.g., contamination of the coolant, solidification of volatile fission products in the vent line, etc.). Verification of fuel pin designs which incorporate fission gas reservoirs or venting systems requires testing of full-sized fuel pins to the high burnups prototypic of fast reactor operation.

Fuel Component Design and Properties. Fuel pin tests will be used to establish fuel component designs and compositions. The type of fuel-cladding bond (or gap) will have to be established under heat transfer conditions peculiar to fast reactors. Irradiation tests must be designed to determine the effect of such variables as fuel density and power density on thermal conductivity, swelling, and mechanical properties. With certain types of fuel the design for axial compartmentation must be tested. Fuels which rely on a physical property response to provide inherent safety features (e.g., thermal expansion to provide a negative temperature coefficient) must be rigorously checked to assure that the desired physical properties (response) are maintained, or at least known, throughout the irradiation. Solid fission product accumulation and microstructural changes must be examined to determine the manner in which they affect the physical and mechanical properties of the fuel. The effects of fabrication process variables and chemical composition of the fuel on performance must be determined. Proof testing of core design and composition require the use of prototypic fuel pins irradiated in a fast reactor environment.

Failure Behavior. Fast reactor fuels, like thermal reactor fuels, must be designed to minimize damage and loss of reactor operating time in the event of a failure. This requires understanding failure mechanisms, establishing intrinsic limits of the design, and determining postfailure behavior of the fuel. Tests should be performed to establish limiting conditions on such "failure promoters" as fuel swelling, core-cladding interactions, cladding strain, and hot spots. These tests and run-to-failure tests are required to establish realistic operating conditions which will reduce the probability of failure to an acceptable level. Postfailure consequences to be studied should include:

- The exposure of fuel to the coolant
- The release of fission gas and other radioactive materials to the coolant
- Swelling of the fuel resulting in coolant flow perturbations
- Failure propagation among pins
- Structural damage to reactor components

Capability for postfailure operation must be determined. The ability to operate after cladding penetration is a desirable and perhaps necessary feature for fast reactor fuels.

Failure behavior tests must be performed in special facilities possessing an independent coolant loop, failure detection devices, and loop clean-up (decontamination) equipment. It will be necessary to establish (1) when failure occurs, (2) temperature changes caused by the failure, (3) fission product and fuel release to the coolant during and after failure, (4) the extent of contamination spread, and (5) loop clean-up characteristics.

Subassembly Proof Testing

The principal purpose of full-scale subassembly testing in a fast flux reactor is to demonstrate performance in an environment simulating that of the reactor in which the subassembly is eventually to be used. To this end, it is desirable to have the same coolant, mechanical restraints, coolant flow characteristics, and the same temperature and flux gradient both axially and radially in the test facility as in the reactor for which the subassembly is designed. The subassembly should be fabricated in the same manner as the production units to assure a proper proof test of the production subassembly. However, the test should not be primarily designed for obtaining information on individual fuel pin behavior within the element, or on the effect of failure of a pin upon adjacent pin elements. This information can be obtained from either single pin tests or tests of small bundles of pin elements.

Subassembly irradiations should be conducted primarily to test the structural members and design of the fuel element assembly rather than the fuel. The effect of stresses imposed by power and/or temperature gradients in the subassembly must be ascertained. With a properly designed fuel assembly such stresses do not distort fuel pins or structural members. Creep rates at the load conditions existing in the fuel assembly should be established through subassembly proof testing. Also, such tests should provide for evaluating the effect of neutron damage on the mechanical properties of the subassembly structural materials. Subassembly irradiations are also needed to verify heat transfer and flow characteristics of the subassembly.

Irradiation facilities for subassembly testing should have an appropriately large test section volume and should provide flux and thermal environments prototypic of fast reactors. These facilities should be instrumented to monitor flow, pressure, and temperature as well as dimensional changes which may occur in the subassembly.

Fuel Behavior at Transient and Overpower Conditions

Certain characteristics derived from basic properties of the fuel will be exploited to ensure that a power-limiting response is obtained in the event of a power excursion. Two mechanisms are available to provide this response - thermal expansion and the Doppler effect. In metallic and cermet fuels, thermal expansion of the fuel can be used as the primary means of limiting power excursions. With ceramic fuels, the principal means of limiting power excursions can be derived from a negative Doppler response.

Fast flux irradiation facilities capable of fully demonstrating these safety features are not available; therefore, tests must be designed and performed in thermal reactors to permit inferential demonstration of the desired response in a fast reactor. The SPERT Reactor can be used for directly measuring the excursion-limiting characteristics of mixed-oxide fuels during power transients. Because the response time of a Doppler effect is determined by the rate of heat transfer to U^{238} and because that rate is independent of the degree of moderation of a neutron environment, the results from these tests would be applicable to fast reactors as well as thermal reactors.

Another useful facility for investigating transient effects is the TREAT Reactor. Fuel slumping and/or axial fuel relocation are often considered to be the major potential problems with mixed-oxide fuel. TREAT experiments using irradiated pins of UO_2 - PuO_2 would allow investigation of those phenomena. Pins incorporating mechanical design features to restrict this axial relocation could be tested.

In cermet fuel, a potentially serious problem is the embrittlement of the matrix during fast flux irradiation. The thermal expansion of the cermet during a power excursion, due to its negative temperature coefficient, is relied on to limit the excursion. However, if embrittlement of the matrix has

occurred thermal shock may cause cracking or shattering of the cermet, rather than reliable thermal expansion. TREAT experiments of fuel pins previously irradiated in a fast flux, such as in EBR-II or EFFBR, would yield valuable information. Similar experiments could be performed on metallic fuel candidates. The effect of burnup, severity of excursions, and of repeated transients would be investigated.

STRUCTURAL AND CLADDING MATERIALS

Irradiation Effects

Reactor structural and fuel cladding materials are damaged by relatively long exposures to neutrons. Generally, such damage manifests itself as embrittlement and a loss of ductility. Damage may be produced in two ways: (a) by neutron bombardment, which results in various defect configurations in the lattice, and (b) by neutron reactions, which create gas atoms and generate other impurities through nuclear transmutation. Although these damage mechanisms occur in both thermal and fast reactors, the extent to which they occur in a fast reactor is ordinarily orders of magnitude greater than in thermal reactors.

Displacement Damage

Displacement damage results from the collision of fast neutrons with atoms. As the number of atoms displaced becomes large, disordered regions are formed, particularly at low irradiation temperatures, which significantly affect physical and mechanical properties. At high irradiation temperatures, those representative of fast reactor conditions, partial annealing of the damage occurs; however, sufficient damage can remain to degrade the properties of the material.

Conceivably, the fluxes in fast reactors could result in displacement rates sufficiently high to cause dynamic mixing. The short or long range order of precipitation hardened alloys can be altered as the atoms are forced to migrate. Homogenization of alloys can be induced by effective acceleration of diffusion controlled processes. Overaging in precipitation hardened alloys through an increase in the average precipitate size could be caused by intensive neutron bombardment as smaller particles are dissolved and the fragments become attached to larger particles.

In addition to a dynamic mixing of the atoms in the alloy through displacement, diffusion processes are enhanced by the number of excess vacancies created by the irradiation. The type and extent of displacement damage is known to be a function of the neutron spectrum involved and thus will vary between thermal and fast reactors. The higher flux levels in fast reactors could promote higher displacement rates whereby diffusion processes may be accelerated. As an example, an estimate of the displacement rate which could occur in the FTR indicates that each atom in an in-core position may be displaced on an average of once per day! The results of such high flux irradiations may promote carbide and nitride precipitation and the formation of sigma phase in excess of what might appear in thermal reactor irradiations of stainless steel.

Neutron Reactions

Transmutations which involve (n, α) and (n, p) reactions promote damage to the mechanical properties of the structural materials through the formation of helium and hydrogen atoms. Helium atoms are formed as a result of (n, α) reactions by fast neutrons, and also by (n, α) reactions in B^{10} by thermal neutrons. The initial spatial distribution of the helium generated in a reactor component must be considered in assessing (n, α) damage mechanisms. The nature of this distribution is directly determined by the characteristic distribution of isotopes participating in the (n, α) reaction. For the iron and nickel-base alloys, the principal fast neutron reactants are the major alloying constituents. As these are essentially in homogeneous solid solution, the resulting helium is generated uniformly throughout the metallic matrix.

The distribution of boron in these alloys is usually not homogeneous. For example, the equilibrium segregation of boron to the grain boundaries could result in an intergranular boron concentration as much as three orders of magnitude greater than the corresponding intragranular concentration. Additionally, the solubility of boron in austenite is quite low (about 6 ppm at 800 °C). Thus, an appreciable fraction of the thermal helium generated in a reactor component occurs at the grain boundaries, lattice imperfections,

and in intragranular carbide precipitates. The resulting distributions of the thermal helium is therefore heterogeneous with helium-rich regions surrounding imperfections and carbides (see Figure 3).

FIGURE 3

Helium Production Characteristics in Selected Irradiation Facilities

A thermal neutron environment not only fails to reproduce the homogeneous helium distribution characteristic of a fast neutron environment, but also results in a particular distribution that is distinctly undesirable. This is especially true when an attempt is made to thermally simulate various fast neutron effects by "doping" specimens with boron.

The helium transport process itself may be a direct function of the neutron environment. The process could be influenced by the damage rate and damage state which, in turn, are a function of the neutron spectrum.

The swelling of the material of interest as a result of helium bubble formation and growth is another damage mechanism. The amount of

swelling accompanying a given helium concentration is related to such spectral-dependent factors as the number of nucleation sites present, the production characteristics of the helium, and the availability of vacancies.

In a fast neutron environment, the reactions $\text{Fe}^{54} (n, p) \text{Mn}^{54}$, $\text{Ni}^{58} (n, p) \text{Co}^{58}$, and $\text{Cr}^{50} (n, p) \text{V}^{50}$ are the principal sources of the hydrogen produced in iron-nickel based alloys. These reactions all have effective energies greater than 4 MeV. Thus, the thermal component of the flux has no significant effect on hydrogen production. The fast fluxes in thermal reactors are not sufficient to yield enough (n, p) reactions to significantly affect the mechanical or physical properties of structural materials during a typical in-reactor lifetime.

Irradiation Testing of Established Materials

An evaluation must be made of the adequacy of established alloys for fast reactor service. These evaluations should include materials used in-core for bearing surfaces, seals, shields, and in sensing, safety, and control devices as well as materials used for cladding and fuel subassembly structural members. Tests must be performed to provide answers to such questions as:

- To what extent are mechanical and physical properties impaired by lattice defects resulting from an integrated neutron dose of 10^{24} n/cm^2 ?
- Will the high flux in fast reactors cause displacement rates sufficiently high to cause dynamic mixing?
- To what extent will high flux enhance diffusion processes, which, in turn, might destroy alloy integrity by promoting grain boundary precipitation?
- To what extent will properties be degraded by the transmutations which will occur during exposures to 10^{24} n/cm^2 ?
- Will the gas generated by these transmutations be uniformly distributed throughout the lattice, or will the gas migrate to grain boundaries and other traps?
- Will the gas created by transmutations form bubbles and coalesce to inordinately stress the alloy?

- To what extent will radiation damage be mitigated (or aggravated) by high operating temperatures?

Some tentative answers can be obtained from simple screening tests in which pre- and postirradiation measurements of alloy properties are made. Long exposure tests will be required to establish the effect of flux and the effect of differences among alloys. High flux on the order of $1 \text{ to } 3 \times 10^{16} \text{ n/cm}^2\text{-sec}$ is desired to obtain high neutron doses as rapidly as possible. Most of these tests can be performed using relatively small specimens in open loops in fast reactors. Temperature and flux conditions should be held constant throughout the test.

More elaborate tests are required to measure the alloy property changes from and during irradiation. These tests should be designed to measure creep, stress-to-rupture, and the effect of strain cycling at typical fast reactor operating conditions. Duplicate out-of-reactor tests in the same coolant and at the same temperature will be required to provide a reference. Instrumented facilities capable of monitoring and controlling temperature and pressure, and of measuring dimensional changes of the specimen are required for these tests. Most of the tests can be performed in open loops using specially designed capsules to provide the temperature conditions and instrumentation required. Some specimens (or irradiation facilities) should be equipped with in-reactor heaters. Also, some closed loop testing will be required to obtain special environmental conditions. Specimens should be exposed to a variety of neutron doses to 10^{24} n/cm^2 at fluxes between 10^{15} and $2 \times 10^{16} \text{ n/cm}^2\text{-sec}$.

Irradiation Testing of New Alloys

New alloys can be initially evaluated by simple screening tests in which pre- and postirradiation measurements of mechanical and physical properties are made. Measurement of such properties as hardness, tensile strength, creep, etc., should be made in these types of tests. To accomplish this initial evaluation as rapidly as possible, specimens should be irradiated in the highest flux available to quickly build up high neutron doses. The tests can be performed with relatively small, encapsulated specimens in open test loops. An important test condition is a high flux.

More thorough evaluation will be required of the candidate alloys selected through screening tests. As in the case with the established alloys, in-reactor creep measurements and stress-to-rupture measurements should be made. In-reactor measurements should be augmented by pre- and post-irradiation measurements of hardness and various tensile properties. These tests will require well instrumented facilities. In-reactor heaters for constant temperature conditions may be required for some of these tests. Test conditions should provide for fluxes of about 10^{16} n/cm²-sec and exposures up to 5×10^{23} n/cm².

IRRADIATION FACILITIES FOR FAST REACTOR FUEL
AND MATERIAL DEVELOPMENT

CRITERIA

Several estimates⁽¹⁻⁴⁾ have been made of the type and amount of irradiation testing space required for pursuing a vigorous fast breeder reactor development program. While these estimates differ as to specific needs there is general agreement that existing facilities are inadequate. The AEC has set forth criteria for irradiation testing facilities for the Liquid Metal-Fast Breeder Reactor (LM-FBR) Program.⁽¹⁾ Other testing requirements for fast fuel and material development are presented or may be derived from the Atomic Power Development Associates (APDA) Study,⁽³⁾ the four 1000 MW_e Design Studies,⁽²⁾ the LM-FBR Fuel Evaluation Report,⁽⁴⁾ and the FFTF Program Directive M. Shaw to D. G. Williams, 1/28/66).⁽⁵⁾ The irradiation testing criteria formulated by the AEC along with the range of operating conditions given in the four 1000 MW_e Design Studies are summarized in Table II.

In addition to the criteria listed in Table II, some testing facilities will be needed in which high risk and failure tests can be performed. Such facilities should have the capabilities of monitoring flow, pressure, and temperature; detecting fuel failure and accommodating fuel meltdown. A rapid transit irradiation ("Rabbit") facility is also needed for very short term (a few seconds to several hours) tests.

A fast reactor for fuel and materials testing should also possess certain facilities and design features to accommodate a broad, long-term fast reactor development program. These include:

- Wide range core reactivity control to compensate for variations in core nuclear characteristics under diverse test conditions.
- Special safety features to offset inherent risk involved in tests of previously untried fuel components.
- Specialized handling facilities for test fuel and test loop equipment.

TABLE II
CRITERIA FOR IRRADIATION TESTING FACILITIES
FOR FAST FUEL AND MATERIAL DEVELOPMENT

	LM-FBR Program Criteria ^(1, 5)	Operating Conditions, Four 1000 MW ^e Design Studies ⁽²⁾
Specific Power, W/g fiss. atoms	500-2000	555-2800
Peak Flux, n/cm ² -sec	10 ¹⁶ (min)	to 1.5 x 10 ¹⁶
Exposure, MWd/ton	100,000-200,000	100 0 ,000-225,000
Temperature, °F		
Coolant Inlet	---	800-979
Coolant Outlet	800-1200	1200-1372
Closed Loops	to 1400	---
Maximum Heat Flux, Btu/hr-ft ²	---	570,000- 1,870,000
Length of Test Fuel Pins, ft	2-4	2-6*
Diameter of Test Fuel Pins, in.	0.2-0.4	0.25-0.301*
Open Loop Testing Capacity		
Foot pins/yr	1000-2000	---
Subassemblies/yr	5-10	---
Closed Loop Testing Capacity		
Foot pins/yr	600-1500	---
Subassemblies/yr	4-7	---
Diameter of Closed Loop Test Space		
Diameter, in.	6 (min)	---
Length, ft	6 (min)	---
Number of Instrumented Loops		
Open	6-8	---
Closed	6-8	---

* Dimensions of fuel pins used in design studies.

- Instrumented driver fuel coolant channels in addition to typical test fuel instrumentation to assure known experimental environment for fuel testing.
- Core design to accommodate a wide range of test fuel assembly sizes, lengths, and geometries.
- Both wet and dry examination cells for interim and postirradiation viewing, measuring, and manipulating of test specimens.
- A critical facility to evaluate each core and core modification (including test elements) in advance of reactor operation to assure reactor safety and performance.

FAST REACTOR FACILITIES

EBR-II

The Experimental Breeder Reactor-II (EBR-II) is a sodium-cooled fast breeder with a 62.5 MW_t rating. Its original purpose was to demonstrate the feasibility of power reactor operation with a closed fuel cycle. The plant includes a complete remote fuel processing and fuel element fabrication facility. Reactor core and blanket is contained in a steel vessel which is submerged in a 26 ft diameter primary tank. The primary tank is filled with about 90,000 gal of sodium at 370 °C. With a maximum coolant flow rate of about 9,000 gal/min, the large sodium reservoir ensures that bulk coolant temperature transients are very slowly transmitted to the reactor.

Coolant for fuel and blanket is taken from the bulk sodium in the primary tank, passed through two mechanical pumps and introduced near the bottom of the reactor vessel. Flow is upward through individual fuel and breeder reflector subassemblies. In order to achieve nearly uniform coolant outlet temperatures, the coolant flow rate is adjusted to be consistent with axial and radial power density gradients and discontinuities. The hot coolant leaves the reactor near the top of the reactor vessel and passes through primary heat exchangers submerged in the primary sodium. Sodium is also the working fluid of the intermediate secondary cooling system.

The reactor core is made up of an hexagonal array of subassemblies each containing 91 fuel pins. The fuel, a uranium-fissium alloy 49% enriched in U^{235} , is sodium bonded to stainless steel cladding. The reactor core is 14 in. high and 18 in. diameter. Reactor control is effected by moving fuel in and out of the core from below. There are 12 control and 2 safety subassemblies.

A summary of EBR-II characteristics is given in Table III.

TABLE III
SUMMARY OF EBR-II CHARACTERISTICS

Power:	
Total	62.5 MW _t
Specific	~ 400 W/g fiss. atoms
Peak Flux:	
Total	3.6×10^{15} n/cm ² -sec
E > 0.18 MeV	2.8×10^{15} n/cm ² -sec
Coolant Temperature:	
Inlet	700 °F
Outlet	900 °F
Heat Flux	680,000 Btu/hr-ft ²
Core:	
Length	14 in.
Diameter	19 in.
Subassembly:	
Shape	Hexagonal
Size	2.25 across flats
Size of Test Fuel Pin:	
Length	to ~30 in.
Diameter	to ~0.3 in.

The EBR-II has a potential for large volume (approximately 10 subassemblies) dynamic sodium testing in a maximum total flux of 3.6×10^{15} n/cm²-sec (2.8×10^{15} n/cm²-sec > 0.18 MeV). Future potential exists for testing as many as 20 subassemblies. Lack of capability for accommodating instrumented tests and lack of individual subassembly failure detection equipment are the major drawbacks to fuel testing in the EBR-II. Other limitations are:

- Lack of closed loop facilities.

- Necessity for encapsulating test fuel pins.
- Restricted control of sodium coolant chemistry, temperature, and flow.
- Lack of provisions for single pin testing.
- Lack of facilities for examining and recharging irradiated fuel pins.
- Inability to significantly raise temperature of the coolant to an individual subassembly above that of the reactor coolant temperature.

Efforts have been initiated to eliminate some of these limitations. Anticipated improved performance of the Mark IA fuel and the elimination of equipment operational problems are expected to improve operating efficiency. Installation of facilities have been proposed which will permit interim and postirradiation examination of test fuel pins.

Fermi

The Enrico Fermi Fast Breeder (Fermi) Reactor, built as a prototype central station unit, is privately owned and operated by the Power Reactor Development Company. The reactor has a thermal rating of 200 MW, however, with Core A, power level will be limited to 110 MW. At 110 MW inlet and outlet temperatures of the sodium coolant are 450 °F and 620 °F, respectively. The cylindrical core is 31 in. high by 31 in. diameter and contains 105 subassemblies. Each fuel subassembly contains 140 pins in a square array. The individual fuel pins are 0.158 in. OD and are clad with Zircaloy. The fuel is a U-10 wt% Mo alloy metallurgically bonded to Zircaloy cladding. Each subassembly is about $2\frac{1}{2}$ in. on a side.

Loading or unloading of the core and blanket is accomplished by a handling system that consists of a rotating shield plug and offset handling mechanism. A holddown plate is located below the plug to maintain radial alignment of the core fuel elements and to hold the elements down against the pressure caused by the upward flow of coolant. This holddown plate also acts as a guide for the control element drive.

Space is provided in the central part of the reactor core for eight safety rods and two control rods. All contain boron-carbide as the nuclear poison.

The general characteristics of the Fermi reactor are summarized in Table IV.

TABLE IV
SUMMARY OF FERMI CHARACTERISTICS

	<u>Design</u>	<u>Current</u>
Power:		
Total	200 MW _t	110 MW _t
Specific	675 W/g _t fiss. atoms	355 W/g _t fiss. atoms
Peak Flux:		
Total	4.5×10^{15} n/cm ² -sec	2.5×10^{15} n/cm ² -sec
Fast	3.6×10^{15} n/cm ² -sec	2.0×10^{15} n/cm ² -sec
Coolant Temperature:		
Inlet	550 °F	450 °F
Outlet	800 °F	620 °F
Heat Flux	650,000 Btu/hr-ft ²	~330,000 Btu/hr-ft ²
Core:		
Length	31 in.	
Diameter	31 in.	
Subassembly:		
Shape	Square	
Size	2.65 in. on side	
Size of Test Fuel Pin		
Length	to 50 in.	
Diameter	~0.25 in.	

The Fermi reactor contains large in-core dynamic sodium testing space (5 to 10 subassemblies) with a maximum flux (at 110 MW) of about 2×10^{15} n/cm²-sec. Tests can be performed on unencapsulated fuel pins to investigate pin swelling characteristics, fission gas release rates, fission product migration, fuel relocation characteristics, and fuel-cladding interactions.

Present problems associated with irradiation testing in the Fermi reactor stem from (1) low plant factor, (2) limited ability to accommodate instrumented tests, (3) lack of failure detection equipment, and (4) absence of

closed loops. In addition, the existing core of U-10 wt% Mo fuel alloy is only capable of an additional 50,000 MWd/ton of operation and is expected to be depleted in September of 1968. The scheduled availability of a replacement core is mid-1970.

SEFOR

The Southwest Experimental Fast Oxide Reactor (SEFOR) is designed to obtain physics and engineering data, using fuel compositions and temperatures characteristic of fast power reactor operating conditions. SEFOR is specifically designed for determining the Doppler coefficient of reactivity of ceramic fuels at temperatures extending to fuel melting.

SEFOR will be used to:

- "(a) Investigate operating characteristics of large $\text{PuO}_2\text{-UO}_2$ fast reactors.
- (b) Measure and investigate the Doppler effect during normal operating conditions.
- (c) Measure the Doppler shutdown effect under transient conditions.
- (d) Establish engineering design safety criteria for large economic units."⁽⁶⁾

SEFOR, now under construction, will be sodium-cooled and is designed to operate at power levels up to 20 MW_t with provisions for subsequent conversion to 50 MW_t . The reactor core is approximately 33 in. long and 33 in. diameter and contains approximately 14 to 20 vol% coolant, 20 to 25 vol% structure and the balance is fuel. BeO will be substituted for about 10% of the fuel rods to soften the spectrum to obtain a sufficiently large negative Doppler response. The reactor fuel is mixed $\text{PuO}_2\text{-UO}_2$ clad with a 0.060 in. thick stainless steel. Fuel rod diameter is 1 in. Fuel composition is approximately 86% UO_2 and 14% PuO_2 . The total fuel loading will contain about 300 kg of Pu^{239} . Eighteen movable radial reflector segments plus three internal poison rods and one special reactivity insertion device will be used to control the reactor.

A summary of SEFOR characteristics is given in Table V.

TABLE V
SUMMARY OF SEFOR CHARACTERISTICS

Reactor Power	20 MW _t
Total Peak Flux	6.3×10^{14} n/cm ² -sec
Core Dimensions:	
Height	33 in.
Diameter	33 in.
Fuel	14 % PuO ₂ -86% UO ₂
Fuel Cladding	Stainless Steel
Coolant	Sodium
Core Composition:	
Fuel	50 vol%
Structure	24 vol%
Coolant	14 vol%
Beryllia	12 vol%
Fuel Rod Diameter	1 in.
Fuel Linear Power Generation:	
Maximum	24.2 kW/ft
Fuel Linear Power Generation:	
Average	12.1 kW/ft
Coolant Temperature (steady state):	
Reactor Inlet	700 °F
Reactor Outlet	820 °F
Coolant Flow Rate	4500 gal/min
Coolant Velocity, Average	
Fuel Channel	11.6 ft/sec

SEFOR is considered inadequate as an irradiation facility for fast reactor fuels and materials because of low flux and limited testing capabilities.

FTR

The Fast Test Reactor (FTR), as now conceived, will provide a versatile test facility for high energy neutron irradiation of fuels and structural materials for fast reactors. The reactor will be equipped with a variety of instrumented open and closed test loops and specimen irradiation stations for performing

tests under conditions which closely simulate the core environments of proposed fast breeder power reactors.

The FTR is rated at 400 MW_t but will initially be operated at 300 MW_t . The design concept fuels being considered are a PuO_2 -SS cermet and a mixed U-Pu oxide. The core will initially develop a maximum neutron flux of about $1 \times 10^{16} \text{ n/cm}^2\text{-sec}$. A versatile flux spectrum can be obtained ranging from high hard flux to nominal fast flux; flux spectrum modifications can be made by in-core moderation.

The core of the FTR is composed of 61 hexagonal flow channels (tubes) in a closely packed, skewed, hexagonal array. The tubes extend above and below the core zone in expanding arrays (as dictated by the skewed geometry). Test and driver fuel assemblies are contained in the tubes which serve as flow channels for the liquid sodium coolant. The core zone formed by fuel in the pressure tubes is approximately a right circular cylinder 36 in. high by 37 in. diameter. The FTR conceptual design provides 20 irradiation test facilities. (Through expenditure of additional capital funds, it is possible to provide a maximum of 27 irradiation test facilities in the reactor.) The closed test loops are designed for 1400°F maximum coolant outlet temperature.

The FTR conceptual design provides a high degree of accessibility to the core for both open and closed loop test facilities. The process tubes can be instrumented for coolant flow, pressure, and temperature as well as for sensors applied directly to the fuel assembly. The core has relatively high neutron flux and power to permit exposure of developmental fuel assemblies to environments more intense than they would encounter in currently anticipated power reactor service.

Proposed characteristics of the FTR are summarized in Table VI.

Because the FTR is designed specifically for fuels and materials testing it will include features which are highly desirable in a test reactor. The closed loops will accommodate deliberate fuel element failures and other high risk tests. These loops can also be used to test vented fuel pins. The reactor will be able to accommodate a wide range of test fuel assembly sizes and geometries. The reactor design provides for sufficient reactivity control

TABLE VISUMMARY OF PROPOSED FAST TEST REACTOR CHARACTERISTICS

(Values Given in Parentheses Represent Design Capabilities)

Basic Core (Driver Fueled)

Power	300 MW _t (400 MW _t)
Coolant:	
Fluid	Sodium
Inlet Temperature	500 °F (700 °F)
Outlet Temperature	800 °F (1000 °F)
Fuel (Driver):	
Core	PuO ₂ -SS cermet or mixed U-Pu oxides
Clad	Stainless Steel
Subassembly Shape	Hexagonal assembly of cylindrical rods
Core:	
Length	36 in.
Volume	600 liters
Number of Tubes	61
Tube Shape	Hexagonal
Tube Size	4.5 in. (across flats)
Total Peak Flux	10 ¹⁶ n/cm ² -sec (1.32 x 10 ¹⁶ n/cm ² -sec)
Peak Flux >0.1 MeV	0.71 x 10 ¹⁶ n/cm ² -sec (0.94 x 10 ¹⁶ n/cm ² -sec)
Operating Period	6 weeks operational 1 to 2 weeks downtime

In-Core Test Loops

	<u>Open Loops</u>	<u>Closed Loops</u>
Number of Loops	6 (10)	4 (8)
Power/Loop	6 MW _t (10 MW _t)	4 MW _t (>10 MW _t)
Coolant:		
Fluid	Sodium	Sodium (optional)
Inlet Temperature	500 °F (700 °F)	< 600 to 1200 °F (<600 to 1200 °F)
Outlet Temperature	800 °F (1000 °F)	1400 °F (1400 °F)
Net Velocity	25 ft/sec (60 ft/sec)	30 ft/sec (50 ft/sec)
Fuel (Test):		
Size (diameter)	4.5 in. (8 in.)	4.5 in. (8 in.)
Length	36 in. (48 in.)	24-48 in. (24-48 in.)

Reflector Test Facilities

Number of Open Loop Material Tests	8 (4)
Number of Package Loop Material Tests	0 (4)
Rapid Transit (Rabbit) Test	1 (1)

and safety features to compensate for variations in core nuclear characteristics under a wide range of test conditions. Capabilities for interim and postirradiation viewing, measuring, and manipulation of test specimens in both wet and dry cells are included as part of the reactor facility. In addition, a critical facility will be available for assessing the effect of each core loading (as altered by modifications or test assemblies) on reactor safety and performance.

COMPARISON OF FAST REACTOR IRRADIATION FACILITIES WITH AEC CRITERIA

Fuel and material development programs for fast reactors will require testing conditions which are peculiar to fast reactor service conditions. Basically these conditions are:

- High operating temperatures
- High specific powers
- High fuel burnups
- Sodium (or other liquid metal coolants)
- High flux
- Hard spectrum
- High neutron doses to cladding administered at a high rate.

The FTR will be designed to provide all of these conditions, as well as temperature, flux, and exposure conditions well in excess of those considered minimal. Existing fast reactor facilities (and thermal reactor facilities modified for fast reactor testing) are seriously deficient in providing for fast reactor testing in one or more of the following ways:

- Thermal conditions
- Flux intensity
- Prototypic fast reactor spectrum
- Coolant environment
- Test volume
- Test specimen size
- Failure detection instrumentation
- Instrumentation for monitoring and controlling temperature, pressure, and flow

- Closed loop facilities
- Facilities for interim examination and reinsertion (in the reactor) of fuel elements
- Availability.

In Tables VII and VIII, some of the test requirements specified by the AEC criteria⁽¹⁾ and the operating conditions described in the four 1000 MW Design Studies⁽²⁾ are presented for comparison with the conditions attainable in the FTR, the EBR-II, Fermi, and SEFOR. A comparison can be made between the criteria and the operating conditions, presented in Table VII, of these four reactors. Data on fuel specimen sizes and the size and number of test facilities specified in the criteria and available in the above four reactors are presented in Table VIII.

Although EBR-II, Fermi, and SEFOR are deficient in many of these requirements, the FTR will be designed to meet or exceed all of them.⁽⁸⁾ These deficiencies exist in the earlier reactors because they were designed and built for purposes other than fuel and materials testing. The flux in the EBR-II and Fermi reactors is less than one-half of the minimum called for in the criteria and the specific power is about 20 to 25% less than the minimum.

The existing reactors have very limited capabilities for conducting instrumented tests. The FTR as presently conceived will have closed loops capable of accommodating high risk tests which may involve a fuel meltdown. These loops will also provide a means of (1) testing vented fuels in a flowing sodium system and (2) significantly altering test environments. The EBR-II, SEFOR, and EFFBR (Fermi) do not have closed loops; as a consequence they cannot provide for, or are severely limited in providing for:

- Failure and high risk tests
- Vented fuel tests
- Coolant test temperatures significantly different from reactor coolant temperature
- Coolant compositions other than the reactor coolant.

TABLE VII
TEST CONDITIONS-VARIOUS FAST REACTORS
(Compared to AEC Criteria)

	<u>AEC Criteria*</u>	<u>1000 MW_e Studies**</u>	<u>FTR</u>	<u>EBR-II</u>	<u>Fermi***</u>	<u>SEFOR</u>
Specific Power, W/g fiss. atoms	500-2000	555-2800	to 1400	~400	355	165
Sodium Outlet Temp, °F						
Open Loops	800-1200	1200-1372	800-1000	900	620	820
Closed Loops	to 1400	---	to 1400	---	---	---
Peak Neutron Flux, n/cm ² -sec	10 ¹⁶ (min)	0.3-1.5 x 10 ¹⁶	10 ¹⁶ (initial) 1.35 x 10 ¹⁶ (design)	3.6 x 10 ¹⁵	2.5 x 10 ¹⁵	6.3 x 10 ¹⁴

* See Reference (1)

** See Reference (2)

*** Based on reactor power level of 110 MW

TABLE VIII
TEST FACILITIES AND TEST SPECIMEN SIZE-
VARIOUS FAST REACTORS COMPARED TO AEC CRITERIA

	<u>AEC[†]</u> <u>Criteria</u>	<u>1000 MW^{††}</u> <u>Studies</u>	<u>FTR</u>	<u>EBR-II</u>	<u>Fermi</u>	<u>SEFOR</u>
<u>Number of Instrumented Loops:</u>						
Closed	6-8	NA	4-8	0	0	0
Open	6-8	NA	6-10	0	0	0
<u>Dimensions of Closed Loops:</u>						
Diameter, in. [‡]	6	NA	4.5 H 4.5-8 R	-- --	-- --	-- --
Length, ft	6	NA	8	--	--	--
<u>Dimensions of Test Fuel Pins:</u>						
Diameter, in.	0.2-0.4	0.25- 0.3 ^{‡‡}	0.2- 0.4	0.16- 0.59	to 0.3	to 1.0
Length, ft	2-4	2-6 ^{‡‡}	2-4	2-5	1.1	3

[†] See Reference (1)

^{††} See Reference (2)

[‡] "H" indicates hexagonal shaped channel. Value = distance between flats.
 "R" indicates round channel. Value = diameter.

^{‡‡} Size of fuel pins specified for design studied.

There are other testing capabilities which are highly desirable and perhaps even essential for fast reactor fuel and material development. A rapid transit irradiation (Rabbit) facility would provide a means of obtaining very short irradiations (a few seconds to a few hours) on test specimens. The reactor operating efficiency (plant factor) should be reasonably high to minimize time to obtain goal exposures. The Conceptual FTR Design is aimed at an operating efficiency of about 75%—a difficult goal for any test reactor. EBR-II and Fermi have goal operating efficiencies of about 50%.

THERMAL REACTOR FACILITIES

Existing facilities are inadequate to meet the testing needs described above. Although the FTR, now in conceptual design, will provide for much of these needs, extensive use of existing facilities, both fast and thermal, will be required to meet the needs of a vigorous fast reactor fuels and materials development program. In spite of their deficiencies, thermal reactor irradiation facilities will probably be used for the bulk of the fast reactor fuels and materials testing until the FTR is in operation. Although these facilities have fluxes, coolants, and temperatures which are not representative of fast reactors, they do provide (1) large volumes of test space capable of receiving highly instrumented tests, (2) relatively high burnup rates, and (3) loops for failure behavior tests.

Testing in thermal reactors is expected to contribute significantly to fast reactor development in still another way. A decade of experience has been obtained with fuels and structural materials in thermal reactor systems. While it is yet imprudent to extrapolate this experience to fast reactor service, such experience can be used as a starting point for developing fast reactor fuel and material technology. To provide a sound basis will require highly sophisticated tests from which fundamental irradiation effects can be identified. Fuels and materials being tested must be fully characterized, temperature conditions rigidly controlled, and neutron spectra well defined. Information from these tests, supplemented by information from comparable tests performed in fast reactor environments, may provide a means for analytically correlating radiation effects in thermal reactor environments with those obtained in fast reactors.

Companion tests performed in fast reactors and thermal reactors will be used to empirically verify such correlations.

Irradiation facilities in thermal reactors can be modified to provide "tailored" flux conditions. For example, thermal neutron shields can be used to reduce the thermal component of the flux, or converter rings can be used to increase fast flux.

ETR

The Engineering Test Reactor (ETR) is a light water reactor with a rated power level of 175 MW_t. The active section of the core is 36 in. high by 30 in. square and is loaded with 52 fuel elements and 16 control rods. Test positions are located both in the core and the reflector. The peak thermal and fast ($E > 1$ MeV) neutron fluxes in the core are 5×10^{14} and 1.5×10^{14} n/cm²-sec, respectively.

Exclusive of test loop volume, the reactor contains about twenty 3 x 3 in. test positions in the core, thirty 1 in. diameter test holes in the beryllium reflector region, and about forty-seven 3 x 3 in. test positions in the aluminum reflector. (These 47 test positions are the ones closest to the reactor core. Other positions in the aluminum reflector have fluxes too low to be of significant value for fast fuel and materials testing.) Unperturbed peak fluxes in the test positions in the beryllium and aluminum reflectors are 2 to 4×10^{14} and 2.5×10^{14} n/cm²-sec, respectively. The fast component ($E > 1$ MeV) of the neutron flux in the reflector positions is about one-tenth of the total flux. Test positions in the core and reflector can accommodate, to a limited extent, instrumented test assemblies. The maximum permissible coolant temperatures for these positions is 120 °F.

The ETR contains twelve major loop facilities, of which eight are committed to Naval Reactor programs. These loops are all water cooled and have test sections 36 in. long. Loop diameters range from about 1.5 to 3.5 in. and have a combined test volume of about 1500 in³. Nine of the loops are located in the reactor core and have peak neutron fluxes of about 5×10^{14} n/cm²-sec with a fast component of 1.5×10^{14} n/cm²-sec. The three loops located in the reflector have peak neutron fluxes of 1 to 1.8×10^{14} n/cm²-sec.

The fast component of this neutron flux is about one-tenth of the total flux. Only one of the loops (P-7) is designed for testing failed fuel elements. A description of the operating conditions and test capabilities of the 12 loops in ETR is given in Table IX (next page).

MTR

The Materials Test Reactor (MTR) is a light water cooled 40 MW_t reactor with a core 16 by 28 in. in cross section and 36 in. high. The reactor core and beryllium reflector are contained in a 35 in. diameter aluminum tank. Irradiation positions are located in both the core and the reflector. Reactor fuel is an aluminum-highly enriched uranium alloy. The reactor has an average and maximum neutron flux of $2.5 \times 10^{14} \text{ n/cm}^2\text{-sec}$ and $5 \times 10^{14} \text{ n/cm}^2\text{-sec}$, respectively. The fast component ($E > 1 \text{ MeV}$) is about one-fourth of the total flux.

Exclusive of test loop volume the reactor contains nine 3 by 3 by 36 in. test positions in the core and 43 similarly sized test positions in the reflector. Of the reflector test positions, only nine have an unperturbed neutron flux equal to or greater than $1 \times 10^{14} \text{ n/cm}^2\text{-sec}$. Two of these nine positions have thermal neutron fluxes of $2.7 \times 10^{14} \text{ n/cm}^2\text{-sec}$. The fast component ($E > 1 \text{ MeV}$) in the reflector positions is about one-tenth of the total neutron flux. Test positions in the core and reflector can accommodate instrumented test assemblies to a limited extent. Maximum permissible coolant temperature is 120°F .

The four principal test loops in the MTR (B-3, VH-3, B-4, and L-42) are described in Table X. The latter three loops are currently committed to Naval Reactor programs. All four loops have water-cooled test sections 36 in. long. Loop diameters range from 1 to $2\frac{3}{4}$ in. The combined test volume of these loops is about 370 in^3 .

The MTR contains two rapid transit irradiation (Rabbit) facilities each with test sections about 1 in. in diameter.

TABLE IX
ETR LOOPS

Loop and Principal User	Peak Flux, n/cm ² -sec		Diam., in.	Length, in.	Vol., in. ³	Press., psi	Temp., °F	Heat Removal Capacity, MW	Remarks
	Thermal	Fast (E > 1 MeV)							
In Core:									
L-10 West	~5x10 ¹⁴	~1.5x10 ¹⁴	~1.5	36	63.5/ loop	2500	650	~1.5	Reentrant loop with line heaters. Limited capacity for failure tests. Committed to Naval Reactor Program.
J-10 West									
L-8 West									
L-12 West									
M-13 West									
F-10 KAPL									
H-10 KAPL									
G-7	5x10 ¹⁴	1.5x10 ¹⁴	3.4	36	326	2500	650	1.5	Heat removal capacity tied in with M-3. Com- bined capacity 2 MW.
G-12 Phillips	5x10 ¹⁴	1.5x10 ¹⁴	~2.0	36	113	600	350	0.3	-----
Reflector:									
M-3 BNW	1x10 ¹⁴	1x10 ¹³	3.4	36	326	2200	570	~0.5	Heat removal capacity tied in with G-7. Com- bined capacity 2 MW.
P-7 BNW	1.8x10 ¹⁴	1.8x10 ¹³	2.7	36	206	1600	500	0.5	Equipped for fuel failure tests.
C-7 West	1.8x10 ¹⁴	1.8x10 ¹³	~1.5	36	63.5	2500	650	~1.5	-----

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TABLE X
MTR LOOPS

	<u>B-3 (BNW)</u>	<u>VH-3 (West)</u>	<u>B-4 (West)</u>	<u>L-42 (KAPL)</u>
Flux, nV				
Thermal	1×10^{14}	1.7×10^{14}	1.5×10^{14}	2×10^{14}
Fast (E > 1 MeV)	1×10^{13}	5×10^{12}	1.5×10^{13}	2×10^{13}
Size,				
Diameter, in.	1.5	~1.0	2.75	1.5
Length, in.	36	36	36	36
Volume, in. ³	63.5	~28.2	214	63.5
Design Press.,				
psi	300	2500	2500	2500
Temperature, °F	290	600	600	600
Heat Removal				
Capacity, kW	300	1	500	~500

GETR

The General Electric Test Reactor (GETR) is a light water cooled and moderated reactor which normally operates at a power level of 30 MW_t. It is privately owned and operated. The reactor core is a 24 by 24 in. matrix with an active length of 36 in. Appropriately shaped beryllium and aluminum reflector pieces round out the core to a right circular cylindrical shape. The reactor is controlled by six bottom mounted, top entry control rods each comprising a poison section and a fuel section. The reactor core is forced convection cooled by demineralized water circulated in the primary system. Irradiation facilities are located both in the pressure vessel and the reflector pool.

The reactor core has an assortment of special filler pieces designed to accept both capsule and loop experiments. A typical experimental loading of the core includes eight large capsule locations in the inner ring, eight large capsule locations in the outer ring, eight small capsule locations (isotope holes) in the peripheral filler pieces, and three in-core loop locations. At times this arrangement is changed by moving experimental filler pieces and adding additional fuel elements. The in-core loop locations can also be used for capsule experiments. Core capsules are cooled by primary reactor water.

The reactor contains about 24 in-core capsule locations. Sixteen of these are in filler blocks containing 1.5 in. diameter holes and eight are in 0.634 in. diameter "isotope holes" located near the periphery of the core. Gross test volume available in the in-core capsule locations (assuming test section length equal to core length) is about 1100 in³. The maximum unperturbed thermal flux is about 2×10^{14} n/cm²-sec. Inlet and outlet coolant water temperatures for the capsule locations are 120 and 140 °F respectively and the maximum permissible capsule surface temperature is 330 °F. Instrumented capsules and loops can be irradiated in these in-core locations.

There are 37 capsule tubes located outside of the reactor core region. These tubes are removable and can be replaced by pool loop facility tubes, large diameter capsule tubes, or other special facilities. The basic pool capsule tube is 1½ in. in diameter and about 45 in. long and is cooled by 100 °F inlet water

supplied by the pool coolant pump. Maximum permissible capsule surface temperature is 220 °F. Gross testing volume available in these 37 pool capsule tubes (assuming test section length is equal to core length) is about 2280 in³. Instrumented test assemblies can be accommodated in the pool capsule locations. The average unperturbed thermal flux in the pool capsule facilities is about 7.5×10^{13} n/cm²-sec. Peak fluxes of 1×10^{14} n/cm²-sec can be obtained in certain locations.

Currently there are two major loop facilities in the GETR - the Boiling Water Loop and the High Temperature Helium Loop. These loops have been designed for specific test purposes or test programs. However, they can be modified to produce different test conditions for new experimental programs. In addition to the two existing loops, there are three locations within the core designated for loop facilities. Typical operating conditions and capacities of the existing loops are described in Table XI.

TABLE XI
GETR LOOPS

	<u>Boiling Water Loop</u>	<u>High Temperature Helium Loop</u>
Thermal Neutron Flux, avg.	5×10^{13} n/cm ² -sec	1×10^{13} n/cm ² -sec
Inlet Coolant Temperature	510 °F	500 °F
Outlet Coolant Temperature	---	1800 °F
Loop Pressure	1300 psia	350 psia
Element Peak Heat Flux	300,000 Btu/hr-ft ²	150,000 Btu/hr-ft ²
Fuel Element Power	140 KW	76 KW
Coolant Flow Rate	45 gal/min	162 lb/hr
Test Section Size:		
Diameter	~3.0 in.	~3.0 in.
Length	36 in.	36 in.
Volume	255 in. ³	255 in. ³

The Boiling Water Loop located in the reactor pool is a general purpose loop equipped to operate either as a boiling water loop or as a pressurized water loop. The facility tube is a radially adjustable U-tube. With the appropriate selection of fuel enrichment and design, heat fluxes in the range of 500,000 Btu/hr-ft² can be accommodated.

The High Temperature Helium Loop is designed to test graphite clad fuel assemblies. The facility consists of: a single tube connected to a closed loop helium gas circulating system with provisions for sampling fission products released from the fuel; a fission product trapping system; and containment cubicles with ventilation system containment filters.

In addition to the capsule and loop facilities described above, the GETR contains the following irradiation facilities:

- A hydraulic shuttle facility which permits insertion and withdrawal of small capsules during reactor power operation without affecting the operation of the test reactor. Hydraulic shuttle capsules are noninstrumented and may be irradiated several minutes to several days.
- Two trail cable facilities capable of accommodating either instrumented or noninstrumented sealed capsule irradiations. These facilities are intended for relatively short exposure irradiations in which specimens are inserted and withdrawn between scheduled reactor shutdowns.
- A beam port facility which permits special purpose testing in an appreciable fast neutron flux and a fairly large irradiation space. Typical beam port experiments do not generate power to the extent that cooling becomes a matter of safety. If fissionable specimens are used, the power must be limited to a few watts.

TR-SRE

The Two Region Sodium Reactor Experiment (TR-SRE) as it presently exists is a graphite moderated, sodium cooled thermal reactor. The core has been modified from the original SRE and now contains the Power Expansion

Program (PEP) core.⁽⁹⁾ The fuel is uranium carbide enriched to 6.5% U^{235} . The fuel assembly is an eight-rod cluster.

Conversion of the SRE to a two-region reactor having a fast-spectrum center region has been proposed by Atomics International.^(10, 11) The core modifications would consist basically of: removing seven of the Zircaloy-clad, graphite moderator cells from the center of the core, replacing the outer six of these with "decoupler" element consisting of a combination of boron-stainless steel plates and depleted UO_2 rod assemblies, and replacing the central cell with a cluster of 19 hexagonal fuel assemblies to form the fast core. All the fast fuel assemblies are considered as test assemblies by AI. The outer, unmodified region of the reactor would then serve principally as a thermal driver for the fast region.

Modifications required of the reactor auxiliary and support facilities for this conversion are discussed in detail in Reference (11). Reference (10) also proposes the use of package closed loops in the reactor, portable loops of 400 kW to 1 MW_t capacity for fuel and material tests, and a fixed package loop of 1 MW capacity for fuel testing. These loops would be installed in test positions within the fast region or in the decouple elements. The fast spectrum central region of the reactor would be 43 in. long and 10.9 in. in diameter giving a test section volume of about 4,000 in.³ (see Table XII). The fast neutron flux ($E > 0.1$ MeV) obtained in the fast flux region would be about 1×10^{15} n/cm²-sec.

EOCR

The Experimental Organic Cooled Reactor (EOCR) is an organic moderated and cooled test reactor originally built for the organic reactor program and located at the National Reactor Testing Station, Idaho. It has never been operated and is presently in a moth-balled conditions. As-built, the EOCR was intended for 40 MW_t operation, utilizing fully enriched UO_2 cermet fuel with Santowax-OMP as coolant-moderator.

Conversion of the EOCR to a fast testing facility has been proposed by Atomic Power Development Associates (APDA)⁽¹⁴⁾ and independently by

TABLE XII
TEST CAPACITY OF PROPOSED LOOPS
IN TR-SRE, EOCR, AND ATR

	<u>Test Section Dimensions</u>		<u>Test Section Volume</u>		<u>Heat Removal Capacity, MW_t</u>	<u>Temperature, °F</u>	
	<u>L. - in.</u>	<u>ID - in.</u>	<u>in.³</u>	<u>Pin Inch*</u>		<u>In</u>	<u>Out</u>
TR-SRE (AI)	43	10.9	4000	15,523	22.7	650	1200
EOCR (APDA)							
1 Center Loop	36	3.16	282	2,196	3.0	800	1200
4 Pkg Loops	36	0.68	52**	720**	0.1 to 0.2 ea	---	1200
EOCR (PPC)							
1 Center Loop	36	3.2	290	2,196	3.0	---	1200
4 Reentry Loops	36	1.0	113**	1,008**	1.2 ea	---	1200
ATR (APDA)	54	1.2	61	270	0.225	600	1200

* 0.25 in. diameter fuel pin

** Total for four loops

Phillips Petroleum Company (PPC).⁽¹⁵⁾ Both APDA and PPC have proposed rather extensive modifications to the EOCR to accommodate fast fuel tests. Major changes are summarized as follows:

	APDA	PPC
Fuel	Use original fuel type with structural modification	Change to UAl_3 -APM or 2219 aluminum alloy
Coolant-Moderator	Unchanged (Santowax-OMP)	Biphenyl, reverse flow
Maximum Power	60 MW _t	78 MW _t
Number of Sodium Loops	1 large central loop 4 peripheral package loops	1 large central loop 4 peripheral loops
Core Hardware	Replace grid plates	Modify
Control System	Modify as-built system	Redesigned; burnable poison added
Heat Removal	Increased heat required, added heat exchanger capacity, modify or replace as-built pumps	Increase primary circulating rate, air cooler replaced with boiler

Loops, associated liquid metal heat removal facilities, and test support equipment are to be added.

The modification proposed by APDA would provide about 330 in.³ of test section volume. Modifications proposed by PPC would provide about 400 in.³ of test section volume (see Table XII). With either modification the fast neutron flux ($E > 0.1$ MeV) could be considerably less than 1×10^{15} n/cm²-sec.

ATR

The Advanced Test Reactor (ATR) is a light water moderated and cooled reactor providing high neutron flux and independent control of flux and spectrum in the major test facilities of the reactor. This versatility

is a consequence of the unique core arrangement of the reactor. Nine flux trap facilities are arranged in a square array and fueled in a serpentine pattern. A design study of an in-pile package loop in the ATR for fast fuel testing has been made.⁽¹⁶⁾ Such a loop could be installed in one of the corner facilities or "lobes" of the reactor. To accomplish removal of thermal neutrons, the test loop would be encircled by a thermal neutron filter. Because of the severe lobe reactivity loss due to the poisoning effect of the filter, the reactor core would be extended inward, displacing the flux trap region. The NaK-cooled loop proposed for the center of the fast lobe is intended to be a modification of the Pratt & Whitney loop (PW-19) that was operated in the ETR. Thus, the components of the package loop facility proposed for installation in the ATR are the extended core region, the neutron filter, and the loop itself. The proposed test loop would have a test section volume of 270 in.³ (see Table XII). The fast neutron flux in the loop would be about 2.5×10^{14} n/cm²-sec.

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