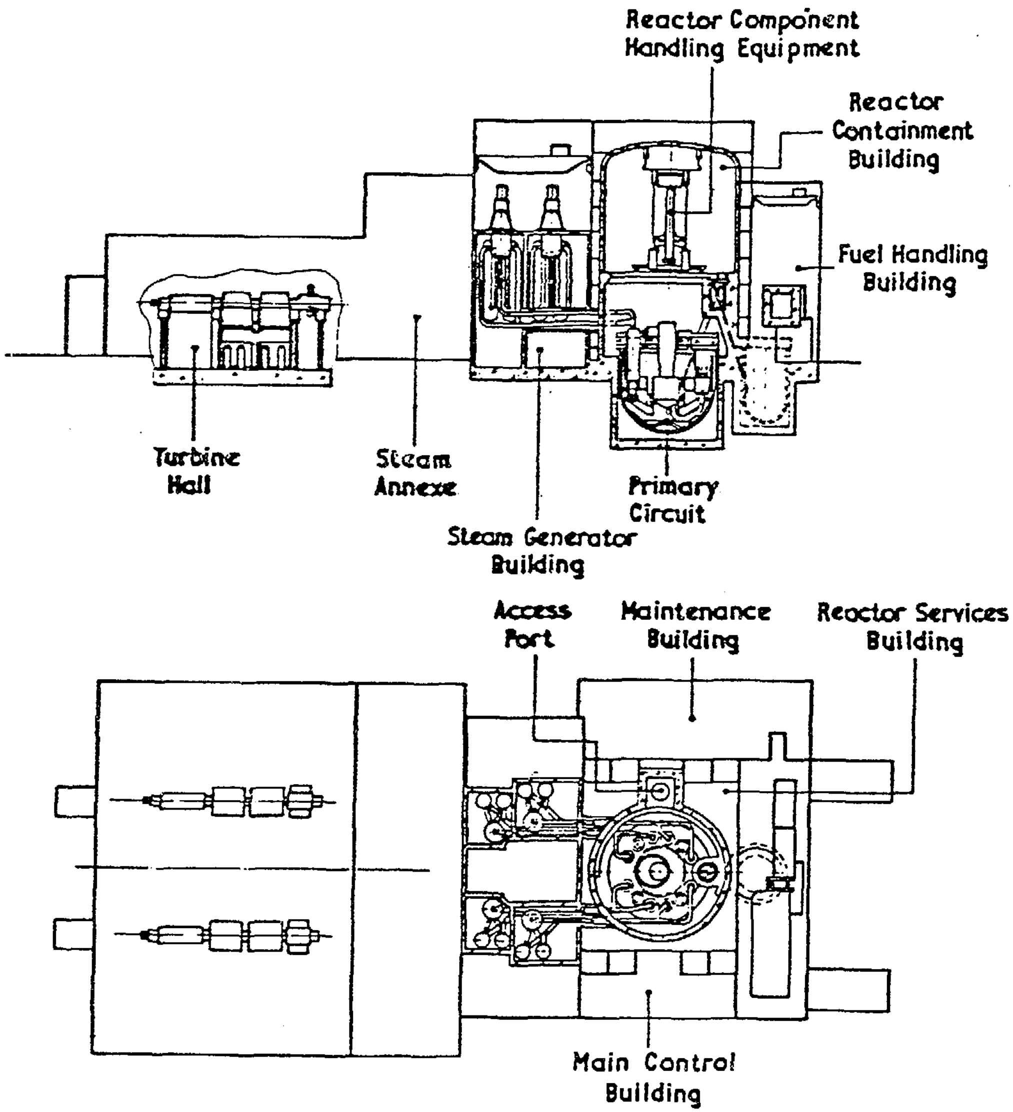

FIG. 2.

STATUS OF FAST BREEDER REACTOR DEVELOPMENT IN THE UNITED STATES

K. HORTON
U.S. Department of Energy
Washington, D.C.
United States of America

INTRODUCTION

The energy policy of the United States is aimed at shifting as rapidly as practicable from an oil dependent economy to one that relies heavily on other fuels and energy sources. Nuclear power is now and is expected to continue to be an important factor in achieving this goal. If nuclear power is to contribute to a solution of future energy needs, demonstration of the breeder reactor as a viable source of essentially inexhaustible energy supply is essential.

The United States Department of Energy program for development of the fast breeder reactor has witnessed some notable events in the past year. Foremost among these is the successful operational testing of the Fast Flux Test Facility (FFTF), located at the Hanford Engineering Development Laboratory. The reactor reached full design power of 400 MW(t) on December 21, 1980, and has performed remarkably close to design specifications.

Design of the Clinch River Breeder Reactor Plant (CRBRP), a 375 MW(e) LMFBR, is now over 80 percent complete. About $530 million in components have been ordered; component deliveries total approximately $124 million; work-in-process totals another $204 million. Construction of the plant, however, has been suspended since 1977. With the concurrence of the U.S. Congress and approvals from the appropriate authorities work on the safety review and site clearing for construction can resume.

The Conceptual Design Study (CDS) for a large, 1000 MW(e) LMFBR Large Developmental Plant (LDP) was recently completed on a schedule commensurate with submission of a full report to the Congress at the end of March, 1981. This report is the culmination of a study which began in October, 1978 and involved contributions from U.S. reactor manufacturers and USDOE laboratories.

The USDOE is carrying forward a comprehensive technology development program. This effort provides direct support to the FFTF and CRBRP projects and to the LDP. It also supports technology development which is generic to the overall LMFBR program. Funding for breeder technology activities exceeds $150 million in FY 1981. Major test facilities supporting the breeder technology efforts are also continuing operation. Several new facilities are under construction or in advanced stages of planning.

The United States Department of Energy continues to be actively involved in international exchanges of breeder reactor technology. This is facilitated through bilateral agreements with other countries engaged in breeder reactor development, as well as through international agencies such as IAEA and OECD.

Plant Projects

Current USDOE plant projects include the Fast Flux Test Facility, the Clinch River Breeder Reactor Plant and the Conceptual Design Study (CDS) Large Developmental Plant. Relative emphasis on these projects has been a controversial issue for the past few years but now appears to be approaching resolution.

Fast Flux Test Facility (FFTF) Project

The FFTF is a 400-megawatt (thermal) sodium-cooled fast reactor specifically designed for development and testing of fast breeder reactor fuels and components. The reactor is a loop-type plant, with three heat transport system loops. The reactor is not provided with blanket assemblies for fissile breeding, further consistent with its role as a test reactor. The outer three rows of core assemblies are stainless steel radial reflector assemblies which serve to enhance the neutron flux in the core interior.

The FFTF is equipped with a great deal of instrumentation. Each core assembly is provided with instruments for measurement of sodium flow rates and outlet sodium temperature. Three instrument trees, one servicing each of the three core sectors, provide this outlet instrumentation for all fuel assemblies, control and safety assemblies, and for selected reflector assemblies. In addition, eight of the 73 core positions are equipped for full in-core instrumentation. Two of these eight positions are closed-loop facilties having independent sodium loops, intermediate heat exchangers, pumps and dump heat exchangers. The other six fully-instrumented positions are open to reactor primary coolant.

Loading of fuel assemblies in FFTF began in November, 1979, and initial criticality was achieved with about 80 percent of the core loaded, on February 9, 1980. Following completion of core loading, a comprehensive series of very low power reactor physics characterization tests was carried out. The extent and precision of this characterization is unparalleled for a large reactor.

Upon completion of low power characterization, nuclear operation of FFTF was terminated to permit plant activities in preparation for the initial ascent to full power. These activities included replacement of a primary sodium

pump which had developed vibrations in excess of acceptable limits. The source of the vibrations was subsequently found to be deflection of the pump shaft, apparently as a consequence of thermal distortion which occurred when the sodium level was allowed to become too high during early operation. Pump replacement was completed in May, 1980.

Other activities during the preparation for power ascent included heating and ventilating system modifications, rework of primary piping insulation, and replacement of a control rod drive mechanism. The integrity of heat transport system cells was assured prior to closeout and inerting. Testing of the heat transport system at the maximum isothermal system temperature was completed in August, 1980. Shield plugs were installed by mid-September, and the cells were inerted. Nuclear operation was resumed on October 1, 1980.

The FFTF ascent to power began on November 17, 1980. Reactor power reached 60 MWt (15%) on December 3 and 140 MWt (35%) on December 5. Extensive reactor physics characterization was conducted at both power levels. On December 10, the reactor was scrammed from 35 percent power in a natural circulation cooling mode. Power ascension was resumed, reaching 300 MWt (75%) on December 15. A normal scram test was conducted from that power level; thereafter, the ascent to 100 percent power resumed. Full power of 400 MWt was attained on December 21, 1980 at 1946 hours. About 48 hours later, a normal reactor shutdown was initiated.

The reactor was returned to the refueling mode in late December and remained in that state during January and February, 1981. During that period, the In-Reactor Thimble was removed and special test assemblies, such as the Absorber Open Test Assembly (AOTA), were installed in the core. Nuclear operation was resumed on March 1, 1981, and a natural circulation test from 400 MWt (100% of full power) was carried out on March 18, 1981.

During reactor operation and plant testing, FFTF parameters have been shown to be well within design specification limits. Fuel assembly outlet sodium temperatures were uniform and within limits. Power coefficients remained negative and within limits over the full power range. There were no reactivity anomalies. Primary and secondary heat transport system loop hydraulics were fully satisfactory, with no flow instability problems observed. Cooling capacity of the intermediate heat exchangers and dump heat exchangers was very close to predictions. The heating and ventilating systems functioned properly, maintaining cell temperatures within specified limits. Sodium purity remained very high, with the highest measured sodium plugging temperature being 275°F (146°C). Functioning of reactor systems during both normal and natural circulation scram tests was excellent. Monitoring of reactor cover gas to date has shown no measurable xenon or krypton levels, indicating continuing integrity of all fuel rods.

Near-future plans for FFTF include the completion of loading of experiments to form the complement of tests for the first complete full-power cycle.

This should begin in June, 1981. An eight day full-power demonstration run is presently planned for August, 1981. Normal full-power operation (Cycle 1) should begin in December, 1981. During the coming year, the FFTF is expected to generate a wealth of data in support of the follow-on plant projects and the generic technology program.

Clinch River Breeder Reactor Plant (CRBRP) Project

Design of the Clinch River Breeder Reactor Plant, a 975 MW(t)/375 MW(e) LMFBR, is now over 80 percent complete. The CRBRP is a three-loop plant which will operate at a vessel outlet sodium temperature of 995°F (535°C). The plant incorporates a "hockey stick" steam generator design, with identical evaporator and superheater modules. Each loop includes two evaporators and one superheater. The steam generator module is a 757-tube counterflow heat exchanger with a dry weight of about 106 tons. The fuel material is mixed uranium-plutonium oxide, and the blanket material is depleted uranium oxide. It is presently planned to operate with a heterogeneous core configuration which consists of a single fuel enrichment zone interspersed with inner blanket assemblies. This fuel and blanket array is surrounded peripherally by radial blanket assemblies. Upper and lower axial blankets are fourteen inches (35.6 cm) in length. Fifteen control assemblies (nine primary and six secondary) are employed. An equilibrium-cycle breeding ratio in the range 1.22 - 1.29 is expected with the heterogeneous core configuration.

The CRBRP Project was authorized by the Congress of the United States in June, 1970. The lead reactor manufacturer (Westinghouse Electric Corporation) was selected in November, 1972, and a formal authorization for preliminary design work was issued in March, 1973. The Final Environmental Statement for the CRBRP was issued by the Nuclear Regulatory Commission (NRC) in February, 1977, and hearings before the Atomic Safety and Licensing Board were scheduled to begin in June, 1977. This schedule was perturbed by an Executive Branch decision, announced on April 20, 1977, to defer LMFBR demonstration indefinitely. Accordingly, on April 25, 1977, licensing proceedings necessary to secure the Limited Work Authorization (LWA) were suspended. After the LWA is obtained, government approval for a Construction Permit must then be obtained in order to begin construction.

Although the future of the Project was in doubt after April, 1977, continuing funding was provided. Although formal licensing proceedings were suspended other project activities continued, including design, analysis and component fabrication. The CRBRP Project has instituted design changes for enhanced licensability and conformance to regulatory requirements. Documentation previously submitted to the NRC, including the CRBRP Preliminary Safety Analysis Report, has been reviewed and updated as necessary.

Design of the CRBRP is now well advanced and components valued at approximately \$75 million were completed and stored during Fiscal Year 1980, which ended September 30, 1980. The 470-ton stainless steel reactor vessel was completed in December, 1979 and is now under storage in Indiana. The cumulative value of equipment delivered through the end of Fiscal Year 1980 is approximately \$105 million, and includes such major components as intermediate heat exchangers and primary sodium pump drive systems. As of the end of Fiscal Year 1980, a total of \$533 million in equipment orders had been placed. Fabrication of the prototype steam generator and the 33,700 gallon per minute sodium pump has been completed. The sodium pump and drive system have been successfully tested in a water test facility. The steam generator will be delivered in late 1981 for testing in the 70 MW(t) Sodium Components Test Installation facility at the Energy Technology Engineering Center (ETEC). Final steps leading to procurement of replaceable core components were initiated in March, 1981. Pending approvals from appropriate authorities NRC will resume the safety review of the plant. The current schedule calls for the plant to be operational in 1990.

Development Plant Conceptual Design Study (CDS)

The Conceptual Design Study for a Large Developmental Plant began in October, 1978 with the objective of defining an LMFBR conceptual design incorporating experience gained in design of FFTF and CRBRP as well as that developed in prior studies, such as that for the PLBR (Prototype Large Breeder Reactor). The study is intended to produce an advanced conceptual design which will serve as a logical next step past CRBRP in the USDOE breeder development program. The conceptual design also serves to define and focus the course to be followed in LMFBR base technology development.

The scope of the Conceptual Design Study included development of a complete LMFBR conceptual design and establishment of cost and schedule estimates for design, construction and operation. In concert with key managers from the base technology programs, a broad plan for developing the necessary base technology was also established.

The study was carried out in two phases. Phase I, of 14 months duration, was the screening, or pre-conceptual design, phase. It extended from October, 1978 to December 1979 and consisted of critical evaluations of various systems options. Technical parameters evaluated included: fuel cycle, plant size, loop or pool design, reactor configuration, steam cycle, turbine generator, fuel handling, and many others. At the conclusion of the screening phase, selection of key plant design parameters was accomplished, and the CDS proceeded to Phase II, the conceptual design phase. This phase extended through March, 1981 and involved development of an integrated plant conceptual design, based on design selections and criteria established during Phase I.

The conceptual design which has emerged is for a loop-type plant, 1000 MWe (gross), with four heat transport system loops and operating at a mixed mean outlet sodium temperature of 950°F (510°C). The plant is to have a service lifetime of 40 years. Befitting its role as a developmental plant, the reactor is provided with a great degree of design flexibility, in that the core, although optimized for use of a mixed uranium-plutonium oxide fuel, can accommodate advanced fuels such as advanced oxides or mixed carbides. The basic core configuration is heterogeneous, but use of a homogeneous configuration is possible. An evolutionary development of the fuel system is planned, proceeding to ever greater burnup levels while maintaining doubling time and plant availability requirements. The plant employs two steam generators per loop and operates on a combined steam cycle. The plant can accomodate any of the candidate steam generator designs currently under development. Two auxiliary cooling systems are provided, one in the reactor vessel and one in the intermediate heat transport system.

The participants in the CDS included reactor manufacturers (Atomics International, Babcock & Wilcox, Combustion Engineering, General Electric and Westinghouse) and architect-engineers (Bechtel, Burns & Rowe, Stone & Webster). Technical integration was provided by the Boeing Engineering and Construction Company in conjunction with the USDOE. An overall review of the final conceptual design was conducted in November 1980, and detailed design reviews on major reactor and plant systems have been proceeding since January 1981. The various reports comprising the final results of the CDS, covering technical, schedular, and economic aspects have been compiled and prepared for submission to the Congress of the United States. Requirements for base technology supporting the Large Developmental Plant design, construction and operation have been established and factored into planning for the base technology programs. Whether the project proceeds with detailed design during the next fiscal year and the schedule on which the plant is built will be deliberated by Congress and the Administration over the next few months.

BREEDER TECHNOLOGY

Present LMFBR technology efforts support the FFTF Project, the CRBRP Project, the Large Developmental Plant, and generic technology which is common to all three. Work under these groups is most conveniently subdivided into four general categories: Safety and Physics, Engineered Systems and Components, Materials and Structures, and Fuel and Materials.

Safety and Physics

The program for development of fast reactor safety technology is organized according to program elements. These elements include core safety considerations, categorized as four lines of assurance (LOA), and plant safety considerations. The four LOA's (LOA-1: prevent accidents; LOA-2: limit core damage; LOA-3: maintain containment integrity; LOA-4: attenuate radiological consequences) emphasize prevention of severe accidents (LOA-1 and 2) as well as mitigation of accident consequences (LOA-3 and 4).

The objective of LOA-1 is to demonstrate that LMFBRs can be designed, constructed and operated in such a manner as to present an extremely low probability of accident initiation. Activities in this LOA address the capability of the reactor system to reliably prevent events requiring reactor shutdown system (RSS) and shutdown heat removal system (SHRS) operation. Other work is intended to ensure that the RSS and SHRS function reliably in termination of plant transients. Recent activities in the LOA-1 category

have included reliability testing of the CRBRP primary and secondary control rod systems. A program for reliability verification of RSS and SHRS is presently in progress.

The second line of assurance, LOA-2, provides a demonstration that the inherent response of the reactor system will limit core damage if accident initiation cannot be prevented. Such initiation could occur if the RSS and/or SHRS fail to function, or if local faults are permitted to propagate beyond Design Basis limits. Activities in LOA-2 focus on (1) providing that RSS faults can be accommodated through the operation of a self-actuated shutdown system; (2) demonstrating inherent or degraded-mode SHRS operation, so that SHRS faults can be accommodated; and (3) showing that local faults which propagate beyond design basis limits can be accommodated without initiation of bulk sodium boiling. During the past year, over 40 tests of a self-actuated shutdown system (SASS) were conducted. Several candidate SASS concepts are currently under development. These include: Temperature-sensitive electromagnetic (TSEM) latch with internal heat source; TSEM latch with external heat source; thermionic-switched electromagnetic latch; and hydrostatically-supported absorber with electromagnetic valve. A full analysis of the shutdown heat removal system was performed in conjunction with the Conceptual Design Study. A test program for evaluation of degraded-mode SHRS operation will be established shortly.

A number of unterminated transient tests with fresh pins have been performed in the Transient Reactor Test facility (TREAT) under the US-UK collaborative program. Future tests will employ pins irradiated in the Prototype Fast Reactor (PFR).

Two tests have been conducted in the Engineering Test Reactor-Sodium Loop Safety Facility (ETR-SLSF) to examine pin bundle behavior under different operational transients. The W-1 test consisted of numerous LOPI-type (Loss of Piping Integrity) transients modeled on the CRBRP design basis accident conditions. The test utilized extensive instrumentation and was carried out in steps of increasing severity. The final run was unterminated, and carried through loss of fuel pin integrity, sodium boiling, dryout, pin failure and flow reversal. The test showed a considerable conservatism in predictive capabilities. The W-2 test in SLSF was conducted in the slow transient overpower regime on a 10¢/sec ramp from a steady-state power level at which the pin bundle had been fully preconditioned. Multiple disruptive failure occurred, leading to a molten fuel-coolant interaction event and flow reversal. Pre-test predictions of pin failures were accurate to within about one second. The next test scheduled for SLSF is the P-4 test, which will examine flow blockage and failure propagation within the pin bundle.

The objective of the third line of assurance, LOA-3, is to demonstrate a high probability of maintaining long-term containment integrity under whole-core disruption conditions. Accommodation of accident energetics is provided by the primary system boundary and by containment. Work in this category has involved modeling and analysis of initation phase and transition phase events and energetics. An updated code, SAS4A, describing initiation phase events, was recently issued. Considerable work was done in evaluation of hypothetical core disruptive accident energetics and debris accommodation for the CDS. Molten UO_2 - MgO interaction tests were conducted. Work is now underway to select and analyze passive core debris retention systems.

The fourth line of assurance, LOA-4, demonstrates a high probability of attenuating the consequences of a radioactivity release inside containment by inherent attenuation mechanisms, such as rapid high-density aerosol depletion, and by operation of engineered cleanup systems. Large-scale sodium aerosol containment tests were recently completed; two submerged-bed gravel scrubber tests were conducted. Plans have been developed for further testing of attenuation and cleanup systems.

Plant Safety Considerations involve the establishment of design criteria for balance-of-plant aspects such that these present no hazards to public health and safety. The BOP aspects considered are fuel handling and storage, liquid metal spills and interactions with construction materials, and auxiliary systems. During the past year, a number of sodium-concrete interaction tests were performed and evaluated.

The Physics program encompasses integral measurements of core and shield properties and detailed nuclear data measurements employing energy-dependent cross section experiments. Computational methods and codes are being developed to enable more accurate calculations of fuel utilization, reactor control parameters, core reactivity coefficients, nuclear heat sources, blanket performance, transient response, and radiation shielding. Advanced experimental and analytical dosimetry methods are being developed and applied to flux/fluence spectral measurements supporting reactor fuels and materials testing. Selected experts in Physics laboratories were directed in 1980 toward analysis and interpretation of the results of nuclear characterization of FFTF at low power. Additional effort provided corroborative reactor physics calculations of the CDS design.

Engineered Systems and Components

The Engineered Systems and Components program is broad in scope, covering the major components of LMFBR heat transport systems and auxiliary systems, as well as the development of instrumentation and control systems, analytical/design methods, and other data supporting component and system design. Primary emphasis has been placed on the development of large steam generators and pumps.

Steam generator development activities include the development, design, fabrication, and testing of two different once-through steam generator designs for future large-plant applications: a single-wall, helically coiled tube concept and a double-wall, straight tube concept. Both designs have been completed and fabrication of 70 MW models is in progress. The helical-coil model will be delivered near the end of Fiscal Year 1982, with the straight tube model to follow about one year later. Testing will be conducted in the ETEC Sodium Components Test Installation (SCTI) in the 1984-1985 period. In addition, the 117-MWt "hockey stick" CRBRP steam generator will be tested in the SCTI shortly. The steam generator effort also includes a series of large sodium-water reaction tests, now in progress, to simulate the effects on an LMFBR steam generator in the unlikely event of the complete rupture of a steam/water tube surrounded by sodium.

The development of large (85,000 gpm) primary-system and intermediate-system sodium pumps is proceeding. Water tests of the reference primary pump model tests have been completed, and fabrication of a prototype pump is progressing toward a delivery date in mid-1983. Model testing and fabrication of a prototype of the reference intermediate pump began recently. An inducer pump element has been installed on a spare FFTF primary pump for testing in the Sodium Pump Test Facility (SPTF) at ETEC this spring. This will be followed by sodium testing of the prototype CRBRP primary pump, also in SPTF.

Other continuing activities in the area of plant components include sodium valve development and performance testing of the CRBRP in-vessel transfer machine. Candidate steam generator leak detection systems, both chemical and acoustic types, will be tested on steam generators in SCTI in 1982 - 1985.

Materials and Structures

The technologies addressed in the Materials and Structures Program are associated with the heat transport system, the reactor vessel, and structures internal to the reactor vessel but outside of the fuel subassemblies. The Program consists of the following eight technical program elements:

- High Temperature Structural Design
- Seismic Design
- Mechanical Properties Design Data
- Fabrication
- Nondestructive Testing
- Corrosion/Tribology
- Advanced Alloys
- Coolants (Sodium Systems)

The High Temperature Structural Design program element is developing design methods and criteria for components that operate in the temperature regime where time-dependent (creep) effects are significant. Tasks included in this element are: (a) inelastic analysis equations and computer programs, (b) time-dependent structural damage and failure criteria, (c) simplified design analysis rules and criteria, (d) weldment design technology, (e) validation of analysis methods and criteria, and (f) flawed component assessment.

The Seismic Design program element is developing verified and cost-effective seismic analysis methods and criteria for breeder reactor facilities. Tasks included in this element are: (a) earthquake ground motion definition, (b) soil/structure interaction analysis, (c) seismic analysis and design criteria, and (d) verification analysis and test.

The Mechanical Properties Design Data program element is providing data on the four principal structural materials - types 304 and 316 stainless steel, 2 1/4 Cr-1Mo steel and Alloy 718. A reference data base is being generated which includes characterization of the effects of sodium exposure, irradiation, and high-temperature service.

The Fabrication program element has investigated methods for manufacturing large-diameter, thin-wall pipe and fittings typical of that required for large LMFBR heat transport systems. Current efforts include development of (a) mechanized welding for stainless steel pipe, (b) stainless steel weld metal with improved high temperature properties, and (c) transition joints to connect stainless steel and ferritic steel piping.

The Nondestructive Testing program element has been developing techniques and equipment required for both fabrication and inservice inspections of LMFBR components. Emphasis is being given to austenitic stainless steel welds, tube-to-tubesheet welds in steam generators, and steam generator tubing.

The Corrosion/Tribology program element is developing: (a) specifications and analytical instrumentation for water chemistry for steam generators, (b) data on the effect of carbon transport, e.g., decarburization of steam generator materials, and (c) data to qualify various material couples for tribological applications.

The Advanced Alloy program element is currently focused on development of a modified 9 Cr-Mo ferritic steel for potential use as an alternative to the present reference structural materials (austenitic stainless steel and 2 1/4 Cr-1Mo steel).

The Coolants program element has been addressing the operation and maintenance of sodium systems with emphasis on analytical and impurity monitoring methods, impurity behavior in sodium, materials compatibility in sodium, cold trap optimization, and radionuclide behavior. Also included in this element is the development of methods and processes for sodium removal and decontamination.

Fuels and Materials

Work in this category involves the design, development, testing, and demonstration of breeder reactor replaceable core components, which includes fuel, blanket, and absorber assemblies. Performance capabilities of various reactor assemblies and their respective component materials are evaluated and improved, utilizing tests covering a broad range of steady-state and design transient conditions. Fabrication of reactor materials and components is also an important development activity in this base technology category.

With the advent of FFTF operation, activities in this area are accelerating. Presently there are 36 full-scale test assemblies in the FFTF core. This total includes 20 standard driver fuel assemblies which are earmarked for special examination upon discharge after one, two, three, or four full cycles of reactor operation (an FFTF cycle is 100 equivalent full-power days). An

additional five standard driver fuel assemblies are designated for operation to cladding breach, covering both central and peripheral core regions. Other tests presently installed include an oxide axial blanket test assembly, three reference absorber assemblies (in control rod positions), four structural materials surveillance assemblies (in reflector positions), and three fully-instrumented test assemblies. There are two instrumented fuels test assemblies in the core, in Rows 2 and 6. These assemblies contain a distribution of thermocouples which map out axial and radial temperature distributions in a full 217-pin reference driver fuel assembly. The third instrumented test now in the core is the counterpart of the fuels tests, in a reference absorber assembly. The absorber test is located in Row 6. These instrumented tests will provide valuable data on assembly thermal-hydraulics characteristics over the full service life of reference FFTF core components.

Beginning in June, 1981, an additional 28 test assemblies will be installed in the FFTF core, prepatory to the first complete full-power cycle (Cycle 1). These tests include ten assemblies with fuel pin spacer variables, four of which are grid-spaced; five assemblies with reference fuel pins and advanced alloy ducts; four oxide fuel assemblies with advanced alloy cladding and ducts; a special test assembly containing helium-bonded carbide fuel pins; oxide radial blanket and inner blanket assemblies; an advanced absorber assembly; two safety-related fuel assemblies with annular axial blanket pellets; a standard driver fuel assembly for operation to cladding breach; a reference driver fuel assembly fitted with an outlet radionuclide trap; and the first of two fully-instrumented test assemblies for cladding/duct materials development, providing active control of irradiation temperature and sensing methods for detection of in-reactor specimen ruptures.

While expending a great deal of effort in preparation for testing, the Fuels and Materials program has also continued steady-state irradiations of fuel pins in EBR-II. These tests are directed primarly toward assessment of the performance of advance cladding alloys. A number of non-fueled tests with advanced alloys are also continuing, to further define the high-fluence swelling, creep, and fracture toughness characteristics of these materials. Run-beyond-cladding-breach (RBCB) tests are also being conducted in EBR-II, some utilizing the Breached Fuel Test Facility (BFTF). The RBCB testing is applied to both reference and advanced cladding alloys.

Transient testing of fuel pins with reference and advanced alloy cladding is being carried on in the TREAT facility. Fuel pin designs, pre-irradiated in EBR-II, are being subjected to transient overpower conditions over a range of ramp rates from 3¢/sec to 50¢/sec. These tests are intended to define operational limits and margins for operation under such transient conditions.

Concurrent with efforts to characterize and improve the performance capabilities of replaceable core components, activities are underway to assure continued, safe, and economical capabilities for manufacture of these components. The USDOE program interfaces closely with primary producers and metal fabricators to optimize manufacturing processes which will provide high-quality materials with reproducible characteristics and product forms which meet design specifications at the lowest possible manufacturing costs.

A major initiative is now underway to develop the requisite technology for fabrication of plutonium-bearing fuel elements in a manner consistent with the need of the nuclear industry and the general public for the next several decades. This technology development activity will provide fabrication methods, known as Secure Automated Fabrication (SAF), which offer high productivity while affording maximum protection to operating personnel and reducing to the lowest practicable limits the potential for accidental or subversive escape of nuclear materials from a fabrication facility. The concept involves full automation of unit processes, from initial powder preparation to the final step in fuel pin fabrication. In addition to these attributes, SAF technology will provide a more uniform product of even higher quality than now produced.

Equipment developed for automated fabrication includes pellet pressing, batch sintering furnace, stoichiometry adjustment system, fuel pellet inspection station and pulsed magnetic welding system for end cap welds. Work is proceeding on powder preparation processes and automated pin loading stations. Advanced process control computer systems are being interfaced with the unit operations, and substantial progress has been made in integration of the automated unit processes.

Process control will be provided by a centralized computer linked to distributed local control systems. The overall system is designed for real-time process operations control, maintenance diagnostics, and accountability reporting. Close-coupled analytical chemistry methods (e.g., fuel assay, oxygen-to-metal ratio, moisture content, gas content, etc.) are in advanced development stages and will support the real-time process control required.

Process integration of SAF technology will be demonstrated in a Cold Test Facility (CTF) in 1981. This demonstration will be carried out without the use of plutonium. Testing of the fully-integrated system will be completed by the end of 1983. Installation of a complete SAF line in the Fuels and Materials Examination Facility (FMEF) will begin in mid-1983, to be followed by testing and debugging operations. Full operation is scheduled for early 1986. The SAF line will have a fuel pin production capacity of six metric tons (U + Pu) per year. This capacity will be sufficient to fulfill the fuel supply requirements for both FFTF and CRBRP. The fabrication line will be fully-automated and remotized. Contact maintenance will be required initially, but ongoing development of remote maintenance techniques in the FMEF will permit introduction of remote maintenance procedures by 1990.

Low level waste processing and wet scrap recycle capabilities are being developed in conjunction with the SAF technology. This will provide well over 99 percent plutonium recovery and a great reduction in waste volume arising from fabrication operations.

TEST FACILITIES

The major separately-funded test facilities of the USDOE breeder reactor program are listed below. The facilities are grouped according to site location.

Facility	Site*	Status
Engineering Test Reactor (ETR)	INEL	Operational
Experimental Breeder Reactor II (EBR-II)	ANL-W	Operational
Hot Fuel Examination Facility (HFEF)	ANL-W	Operational
Transient Reactor Test Facility (TREAT)	ANL-W	Operational**
Zero Power Plutonium Reactor (ZPPR)	ANL-W	Operational
Zero Power Reactor 6 (ZPR-6)	ANL-E	Operational
Zero Power Reactor 9 (ZPR-9)	ANL-E	Standby
Fuels and Materials Examination Facility (FMEF)	HEDL	Under Construction
Maintenance and Storage Facility (MASF)	HEDL	Under Construction
Fuel Storage Facility (FSF)	HEDL	Completed
Sodium Components Test Installation (SCTI)	ETEC	Operational
Thermal Transient Facility (TTF)	ETEC	Operational
Sodium Pump Test Facility (SPTF)	ETEC	Operational
Large Leak Test Rig (LLTR)	ETEC	Operational
Small Components Test Loop (SCTL)	ETEC	Operational

*INEL: Idaho National Engineering Laboratory, Idaho
ANL-W: Argonne National Laboratory-West, Idaho
ANL-E: Argonne National Laboratory-East, Illinois
HEDL: Hanford Engineering Development Laboratory, Washington
ETEC: Energy Technology Engineering Center, California
**A major modification (TREAT Upgrade) is in progress. Completion scheduled for 1985.

INEL Facilities

ETR

The ETR is a 175-MW(t) test reactor that has been in operation since 1957. It has operated with as many as 14 different in-pile loops installed and in operation at one time. The main use for ETR until 1973 was for irradiation of fuel and materials for light water reactor applications, however, now it performs as a key test facility in support of the Department's breeder reactor safety program. In this role, ETR conducts tests contributing significantly toward verification of the safety characteristics of reactor fuel and core designs for reactors using liquid metal as the primary coolant. The ETR accepts a series of tests in large, packaged, sodium in-reactor loops under the Sodium Loop Safety Facility (SLSF) program. Each SLSF loop provides the controlled environment in which reactor safety irradiation experiments can be conducted under conditions closely simulating the thermal-hydraulic conditions, both steady-state and transient, in fast breeder reactor systems.

ANL-W Facilities

The EBR-II is a 62.5 MW(t) - 20 MW(e) sodium-cooled pool-type fast reactor at the Argonne National Laboratory (ANL)-West site of the Idaho National Engineering Laboratory, operated by the ANL for the Department of Energy. EBR-II is currently used primarily for safety, fuels, and materials irradiation experiments. The operation has continued to be highly satisfactory, with a plant capacity factor of 77.1 percent in 1980. The reactor is now at Run 113 and has accumulated a total of nearly 200,000 megawatt-days of operation. A number of specialized tests have been conducted during the past year, including a flow blockage test and several run-beyond-cladding breach (RBCB) tests. Successful operation of the Breached Fuel Test Facility (BFTF) has been demonstrated. Run 112 was conducted as a special slow ramp rate overpower transient test run. Installation of the second in-core test facility, the Fuels Performance Test Facility, is scheduled for mid-1981. Modification of the reactor shutdown system is planned for late-1981. These test facilities and plant modifcations are part of the overall plan for future utilization of EBR-II for operation safety and reliability testing.

Design of fuel assemblies for the TREAT Upgrade project is continuing. The Upgrade will allow the testing of large breeder reactor fuel pin bundles under more severe transient conditions. The SLSF tests W-1 and W-2 were also completed in ETR during the year. Analysis of the results of these tests is proceeding.

HFEF comprises two adjacent hot cell complexes--HFEF-North and HFEF-South--located at INEL and operated by ANL. The South complex contains two heavily shielded hot cells, one with an argon inert atmosphere. Each complex includes unshielded repair areas, test laboratories, equipment rooms, storage areas, offices, and other facilities. The HFEF provides the operational support to EBR-II, the capability to examine and reconstitute experimental subassemblies irradiated in EBR-II for breeder reactor programs on fuels and materials. HFEF will be used for FFTF post irradiation examinations pending completion of the Fuels and Materials Examination Facility at the FFTF site.

TREAT is an air-cooled, thermal, heterogeneous reactor located at the ANL-West site and operated by ANL. TREAT currently provides the capability to irradiate reactor fuels and structural materials under conditions simulating various types of nuclear excursions and transient undercooling situations. It is also available for limited neutron radiography. The facility is being modified to permit transient testing of larger safety experiments in a more prototypic fast reactor environment.

ZPPR is operated by ANL at the INEL in Idaho with capacity for experimental reactor assemblies up to 14 feet x 14 feet x ten feet. ZPPR is the largest and best suited critical facility in the world for accurately modeling large fast reactors. Extensive benchmark and mockup experiments needed for both homogeneous and heterogeneous CRBR core designs are nearly complete. Further benchmark and mockup experiments will be needed at ZPPR during the 1980's to guide and confirm the designs of large-scale developmental cores which may incorporate such additional features as heterogeneous configuration, advanced fuels, and long-lived capabilities. ZPPR is equipped with an automated fuel loading machine, which supplants manual loading/unloading operations and thereby minimizes personnel exposure to radiation in accordance with the Department of Energy policy.

ANL-E Facilities

The ZPR-6 and -9 are located and maintained at ANL-East. As examples of their many uses, a series of critical experiments culminating in the engineering mockup criticals which established the FFTF core design were performed on ZPR-9. ZPR-9 and ZPR-6 more recently enabled safety test facility criticals, and special investigation measurements intended to resolve persistent discrepancies which appear in the analyses of ZPPR experiments. ZPR-9 is currently in standby and operation at ZPR-6 will be deferred during FY 1982.

HEDL Facilities

The FMEF is a large multi-purpose facility which will provide postirradiation examination capabilities in support of FFTF operations and testing. In addition, the facility will house the SAF line for fabrication of FFTF and CRBRP fuel pins. It will also contain a test pin fabrication line and fabrication technology development capabilities. Construction of the FMEF was 12 percent complete at the end of February 1981. The present construction schedule calls for completion of the facility by December 1983.

Construction of the FSF is complete. Acceptance testing is scheduled for completion by July 1982. FSF will provide sufficient fuel storage capacity for five to eight years of FFTF operation and includes provision for storing

FFTF driver fuel assemblies as well as FMEF canisters containing pins previously subjected to postirradiation examination. The facility uses sodium and NaK as the storage and heat transfer media. The storage vessel contains racks which will accommodate over 450 fuel assemblies or canisters.

The MASF is located adjacent to FFTF and is on schedule for completion in October 1982. This facility contains systems for removal of sodium from, and decontamination of, FFTF components. Maintenance and repair can be performed on large components at low activity levels. Self-contained cells are also provided for maintenance of contaminated small components. The facility also furnishes storage for FFTF maintenance equipment.

ETEC Facilities

The SPTF is a large non-nuclear sodium loop for testing sodium pumps, flowmeters, and other large heat transport system components. It can provide thorough flow mapping of pumps and can impose severe thermal transients on test components. The facility is being expanded to increase its capacity to 100,000 gallons per minute (gpm), in preparation for testing large sodium pumps and components. Installation of the prototype CRBRP primary pump for testing will begin in late-1981.

The SCTI is a high heat throughput sodium-cooled heat exchanger test facility. Previous tests with the 35 MW(t) capability included model steam generators and the FFTF dump heat exchanger. Expansion from 35 MW(t) to 70 MW(t) capacity was completed in mid-1980. The facility has been used for testing the 400 gpm CRBRP electromagnetic pump, a CRBRP check valve, and a large variable pressure reduction device. Delivery of the CRBRP prototype steam generator for installation and test is expected in 1982. Future plans for this facility include tests of single-wall, helical coil, and double-wall, straight tube 70 MW steam generators.

The SCTL is a 10-inch sodium loop with a 3500-gpm main pump and a 9500-gallon expansion tank that can be used for immersion testing up to 1200°F (650°C). A subscale inducer pump has been successfully tested in this loop, as has the CRBRP in-vessel transfer machine. Transient tests with CRBRP 4-inch valves were recently completed, and a CRBRP 3-inch valve is being installed for testing. The SCTL is being modified to a double-pump loop, for concurrent pump and component testing.

The LLTR is a sodium-water reaction test facility comprised of a scaled-down LMFBR secondary sodium heat transfer system, steam generator, and vent relief system. Capability is provided for water/steam injection into the sodium side of a prototypical steam generator from an intentionally faulted water/steam tube. Tests of two prototypic CRBRP steam generator geometries are now in progress, the first, a hockey stick configuration with a 16 inch diameter, and the second, a straight tube steam generator with CRBRP steam generator prototypic diameter (5 feet).

The TTF is used to simulate the effect of process fluid transient temperature conditions on plant components by directing high velocity inert gas through thermally preconditioned components. Testing of a CRBRP trimetallic transition joint and creep ratchetting tests on an 8-inch pipe are underway.

THE SWISS CONTRIBUTION TO THE FBR DEVELOPMENT

M. HUDINA, et. al. (Presented by W. SEIFRITZ)
Institut Fédéral de Recherches en Matière de Réacteurs
Würenlingen, Switzerland

1. Nuclear Energy and Future Strategy in Switzerland

By applying for membership in the "International Working Group on Fast Reactors" Switzerland has expressed its interest in the fast breeder reactor and in the LMFBR in particular. For explaining this interest it seems to be appropriate to begin with a brief review of the Swiss nuclear energy scene.

In Switzerland the use of nuclear power for electricity production began in 1969 (Beznau I) as a result of a decision taken by the Swiss electricity companies in the early 1960s to use predominantly nuclear energy to supplement the hydroelectric power. The decision was based on technical, environmental and economic evaluations, and considering the lack of fossile fuel resources in the country. The early introduction of LWRs by the electricity companies contributed to the abandonment of an HWR development project of the Swiss industry, in which the Federal Institute for Reactor Research (EIR) was participating. By 1980 the contributions of hydroelectric, nuclear and conventional thermal power to the total electricity production amounted to 69.6 %, 28.4 % and 2.0 % respectively, the nuclear power being generated by 3 PWRs and a BWR with a total installed capacity of 2 GWe.

Future energy planning in Switzerland is based on scenarios prepared by two commissions nominated by the Federal Council