Fast Reactor Core and Fuel Structural Behaviour held in Inverness in June 1990. The UKAEA is now providing the lead for a new IWGFR co-ordinated research programme on 'Acoustic Signal Processing for the Detection of Boiling or Sodium/Water Reactions in LMFBRs'.

UKAEA specialists participated in two IWGFR specialists' meetings on 'Steam Generator Failure and Failure Propagation Experience' and 'Steam Generator Acoustic/Ultrasonic Detection of Under-sodium Water Leaks' which were held, respectively, in September 1990 and October 1990 in Aix-en-Provence.

A UKAEA delegation participated in the ANS International Topical Meeting on Fast Reactor Safety, held at Snowbird in August 1990. A substantial number of papers by UK authors have been submitted, in co-operation with French and German associates, to the forthcoming International Conference on Fast Reactors and their Fuel Cycles, which is to be held in Kyoto in October.

As in previous years, the UKAEA provided the UK representative for the annual meeting of the IAEA International Working Group on Fast Reactors which was held in Vienna in April 1990.

STATUS OF LIQUID METAL REACTOR DEVELOPMENT IN THE UNITED STATES OF AMERICA

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Abstract

An existing network of government and industry research facilities and engineering test centers in the United States is currently providing test capabilities and the technical expertise required to conduct an aggressive advanced reactor development program. Subseqeunt to the directive to shut down the Fast Flux Test Facility in early 1990, a variety of activities were undertaken to provide support for continued operation. The United States has made substantial progress in achieving ALMR program objectives. The metal fuel cycle is designed to recycle and burn its own actiniums, and has the potential to be a very effective burner of actiniums generated in the LWRs. The current emphasis in the IFR Program is on the comprehensive development of the IFR technology, to be followed by a period of technology demonstration which would verify the economic feasibility of the concept. The United States has been active in international cooperative activities in the fast reactor sector since 1969.

1. OVERVIEW

The U.S. civilian nuclear power research and development program continues to focus on advanced large and mid-size light water reactors, modular high temperature gas cooled reactors and modular liquid metal fast reactors. This paper discusses the Advanced Liquid Metal Reactor program, which is composed of a small, passively safe fast reactor coupled with a metal fuel cycle that incorporates actinide recycle, and an emerging effort to process LWR spent fuel for LMR fissile material, and to enhance the LWR waste management.

The liquid metal reactor concept has a sound technology base, with some three decades of research and development both in this and other countries. An existing network of government and industry research facilities and engineering test centers in the United States is currently providing test capabilities and the technical expertise required to conduct an aggressive advanced reactor development program. Notable among the research facilities is the Experimental Breeder Reactor-II (EBR-II) at Argonne National Laboratory (ANL) in Idaho and the Fast Flux Test Facility (FFTF) at Hanford, Washington. Subsequent to the directive to shut down the Fast Flux Test Facility in early 1990, a variety of activities were undertaken to provide support for continued operation. A marketing effort initiated by the State of Washington discovered sufficient potential for support such that the federal government provided funding through FY 1991. Legislation has been introduced to internationalize the facility as a Reactor Research User Complex. Further market development continues with U.S. DOE participation to develop international partnerships for continued long-term operation.

Current U.S. Advanced Liquid Metal Reactor (ALMR) activity is focused on providing a reactor and fuel cycle system with improved safety margins, better economics, and an attractive waste management (actinide recycle) option. Special attention is being directed to passive safety features, modular plant construction, standardized plant design leading to simplified licensing and shorter construction schedules, factory fabrication, advanced instrumentation and control systems, the use of high performance materials, and the option for on-site fuel processing.

The United States has made substantial progress in achieving ALMR program objectives. After a competitive period, a decision was made in 1988 to select the General Electric ALMR concept known as PRISM (Power Reactor Innovative Small Module) for advanced conceptual design. Substantial progress has been made in design, licensing and economics, even though the original schedule has been stretched due to funding limitations. The Department of Energy's role is to advance the concept to a sufficient level that would enable private sector and/or international interests to support further development and possible cooperative demonstration of a prototype plant.

A key strategy within the U.S. LMR program is to evaluate the potential of metal fuel based on the Integral Fast Reactor (IFR) concept developed at ANL. The technology supports practical actinide recycling. The metal fuel cycle is designed to recycle and burn its own actinides, and has the potential to be a very effective burner of actinides generated in the LWRs. The entire ALMR system can thus extend uranium resources by a hundred-fold, making nuclear essentially the same as a renewable energy source.

The scientific principles involved in the IFR concept have already been shown to be soundly-based, even surpassing expectations in some instances. The current emphasis in the IFR Program is on the comprehensive development of the IFR technology, to be followed by a period of technology demonstration which would verify the economic feasibility of the concept. The development effort is presently focused on parametric investigation of the performance of the U-Pu-Zr ternary alloy metallic fuel; optimization of the flowsheet for the IFR pyroprocessing method for efficient fuel recycle and waste management; design and testing of plant-scale pyroprocessing equipment; and characterization of the many inherent passive safety aspects of the IFR systems for most effective exploitation of these characteristics in the future.

The United States has been active in international cooperative activities in the fast reactor sector since 1969. Over the ensuing years, joint programs evolved which benefitted all parties and lowered research and development costs. Such cooperation continues, at an increasing pace, even though the fast reactor program direction in the U.S. has diverged from the primary direction in Europe and Japan.

2. ALMR PLANT DESIGN, DEVELOPMENT, AND LICENSING

The objective of the power plant design and licensing work is to complete sufficient conceptual and preliminary design activities to:

- determine commercial plant systems economics, safety margins, licensability, and develop acceptable waste management options
- establish the licensability of the evolving design

2.1 Reference Concept

In late 1988, DOE focused its future LMR activities on the Power Reactor Innovative Safe Module (PRISM) design concept. Accordingly, GE was awarded a 5-year contract for Advanced Conceptual Design and Preliminary Design for DOE's Advanced Liquid Metal Reactor (ALMR) Program. During the subsequent period to date GE consolidated an industrial ALMR Team to continue conceptual design and nuclear licensing assessment work. The industrial team includes GE, Babcock and Wilcox, Bechtel, Burns and Roe, United Engineers and Constructors, and Westinghouse, with Argonne National Laboratory (ANL), Energy Technology Engineering Center (ETEC), Idaho National Engineering Laboratory (INEL), Oak Ridge National Laboratory (ORNL), and Westinghouse Hanford Company (WHC) providing technology and test support. Arrangements also have been made for substantial international participation in the program.

An overall project goal is to demonstrate the safety and performance features of the ALMR by test of a full-size prototype reactor and thereby provide a basis for Nuclear Regulatory Commission (NRC) certification of the design as soon as the year 2006.

2.1.1 The Advanced Liquid Metal Reactor (ALMR)

2.1.1.1 Overall Plant

The ALMR design has evolved from the PRISM design initiated by GE in 1980. The fundamentals of this design remain unchanged, and the enhancements made since it's selection as the ALMR in 1988 improve its economic viability and licensability. The overall design approach includes:

- Compact reactor modules sized to enable factory fabrication, economical shipment to both inland and water-side sites, and affordable full-scale prototype testing to confirm safety and performance features;
- Use of liquid sodium as a coolant, which permits operation at atmospheric pressure, a design margin to boiling greater than 400 degrees C (700 degrees F), and elimination of the need for a pressurized primary system and thick-wall pressure vessels;
- Passive shutdown heat removal for loss-of-cooling events, designed to be invulnerable to operator error and equipment failures;
- Passive reactivity reduction to safe, stable state for undercooling and overpower events with failure to scram, to provide abundant time for ultimate shutdown to cold conditions by subsequent operator action;
- By a combination of prevention and mitigation, protection against severe accidents such that NRC protective action guidelines are met with sufficient margin to make the exercise of formal public evacuation plans unnecessary;
- Capability for substantial breeding to provide security for the United States against long-term fissile uranium shortage;
- Capability to utilize as fuel the long-life actinides from spent LWR fuel.

The target commercial ALMR plant utilizes nine reactor modules arranged in three identical 465 MWe power blocks for an overall plant net electrical rating of 1395 MWe (Figure 1). A power block has three identical reactor modules, each with its own steam generator, that jointly supply steam to a single turbine generator (Figures 2 and 3). Table 1 lists general design data. Smaller plant sizes of 465 MWe and 930 MWe would use one or two of the standard power blocks, thus providing size flexibility to the utility in meeting its projected load growth. The reactor module, the intermediate heat transport system (IHTS), and most of the steam generator system are underground, an approach that has an estimated cost benefit in meeting requirements for radioactivity containment, seismic design, sodium fire mitigation, and protection from external threats such as sabotage and missiles.

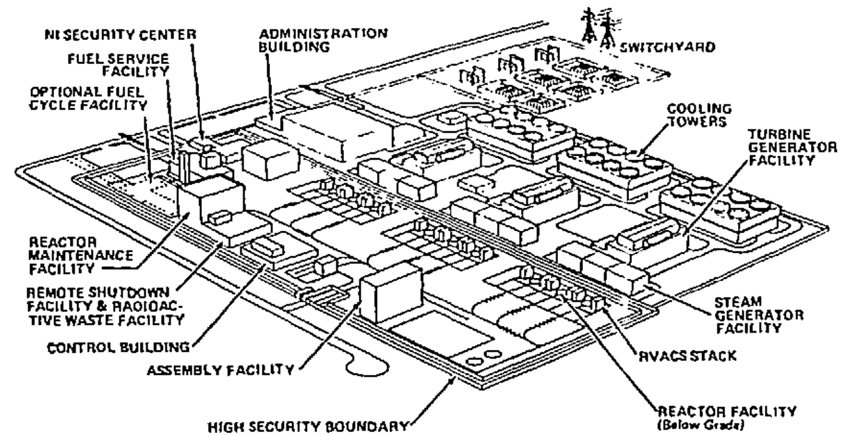


FIG. 1. ALMR power plant (three power blocks), 1395 MWe.

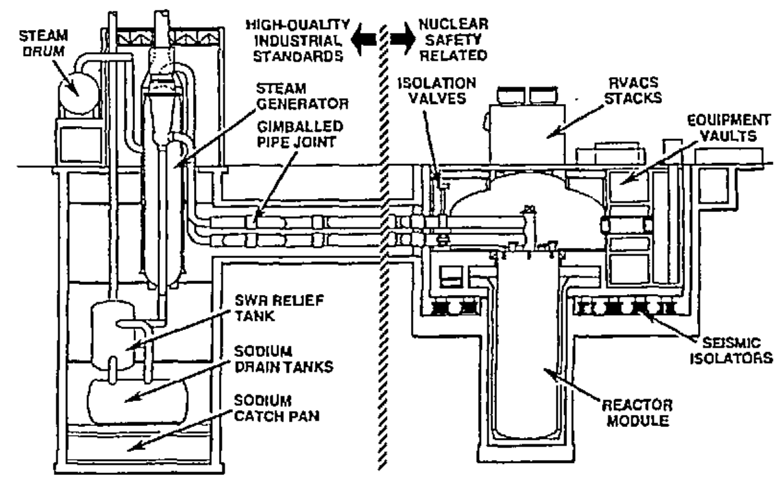


FIG. 2. Nuclear steam supply system (three per power block).

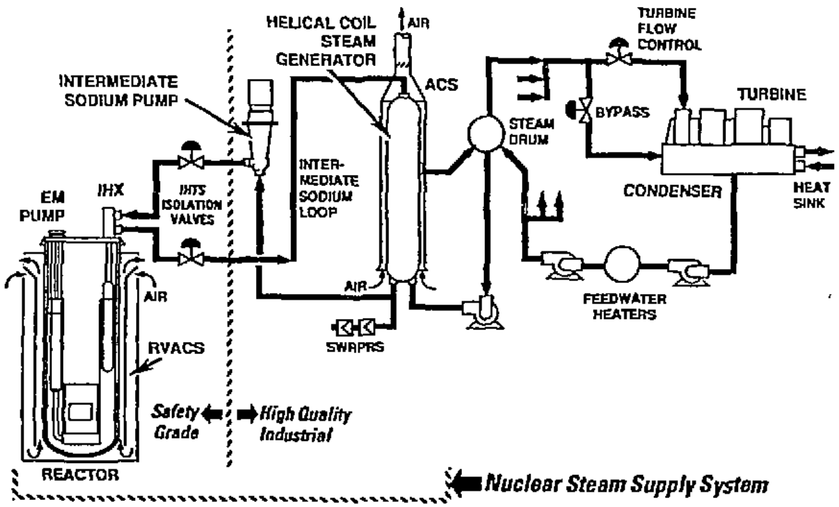


FIG. 3. ALMR main power system.

All nuclear safety-related systems and buildings are enclosed within a double-fenced and barricaded high security area. The steam generator system is physically separated from the nuclear portion of the plant; this system and the IHTS connecting it will be built to upgraded industrial standards.

The reactor module, (Figure 4), is about six meters in diameter and has a shipment weight of about 725 tonnes, not including removable internal components that are shipped separately.

The primary boundary for the core coolant and cover gas, which operate at approximately atmospheric pressure, is comprised of the reactor vessel, the reactor head closure and fittings, and the intermediate heat exchangers. The reactor containment (Figure 4) is a second leaktight pressure-retaining boundary that backs up the primary boundary; it is comprised of a lower containment vessel backing up the reactor vessel and an upper containment vessel backing up the closure head.

TABLE 1

PLANT PERFORMANCE CHARACTERISTICS

Overall Plant	
- Net Electrical Output	1395 MWe
- Net Station Efficiency	32.9%
- Number of Power Blocks	Three
- Number of Reactor Modules:	
per power block	Three
per plant	Nine
Power Block	
- Number of Reactor Modules	Three
- Net Electrical Output	465 MWe
- Steam Generator Number	Three
- Steam Generator Type	Helical Coil
- Turbine Type	1800 rpm, Tandem Compound
	Four Flow - 38-inch Last Stage Bucket
- Turbine Throttle Conditions	965 psia/540 F
- Feedwater Temperature	420 F
Reactor Module	
- Thermal Power (Core)	471 MWt
- Primary Sodium Inlet/Outlet Temperature	640 F/905 F
- Primary Sodium Flow Rate	46,000 gpm
- Intermediate Sodium Inlet/Outlet Temperature	548 F/830 F
- Intermediate Sodium Flow Rate	41,250 gpm
Reactor Core	
- Fuel	U-Pu-Zr Metal
	(Oxide Backup)
- Refueling Interval	18 months
	(12 Mo. for backup)
- Compound System Doubling Time for Breeding	Approximately 100 years

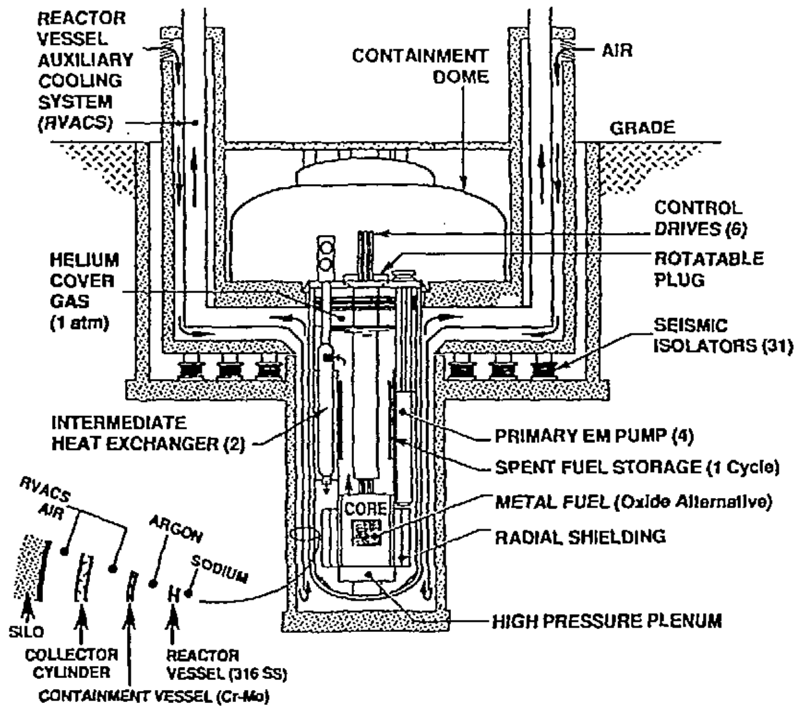


FIG. 4. ALMR reactor module in silo.

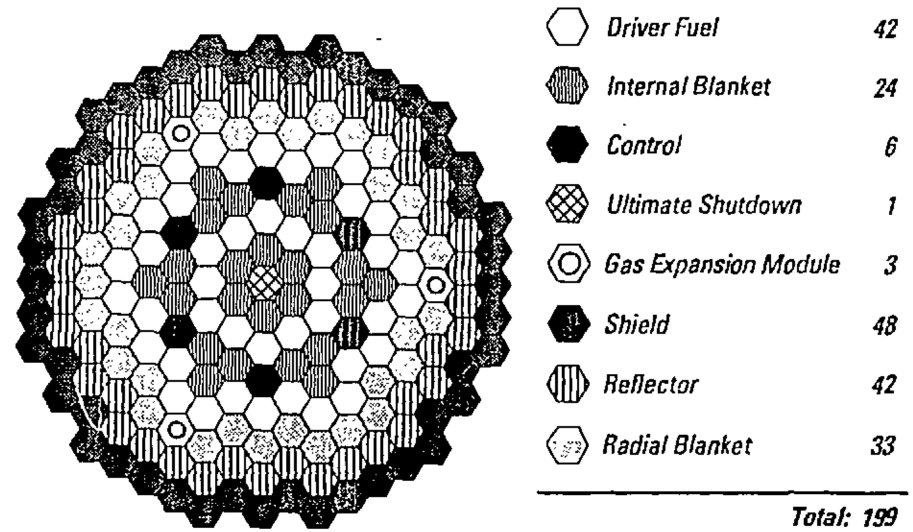


FIG. 5. Reference metal fueled core.

The reference fuel for the ALMR is metallic uranium-plutonium-zirconium alloy. The ferritic alloy HT9 is used for cladding and assembly ducts to minimize swelling associated with long burnups. A heterogeneous arrangement of blanket and driver fuel is used, with six control rods (Figure 5). Refueling occurs after 18 months of operation, with one-third of the core being changed each time; this results in a 4-5 year fuel life (150 MWd/kg peak burnup). Metal fuel provides excellent negative reactivity feedback for loss of cooling and transient overpower events. Metal fuel also provides competitive fuel costs.

For safety margin in the event of loss of the primary sodium pumps, three gas expansion modules (GEMs) have been added at the core periphery (Figure 5). The GEMs are hollow assembly ducts, closed at the top and open at the bottom, and are filled with helium, the same as the vessel cover gas. When the pumps are running, the gas is compressed into the upper portion of the duct, and sodium extends up into the active core region. When the pumps are not running and the inlet pressure therefore is lower, the gas expands and pushes the sodium level down to below the active core region, which increases the neutron leakage from the core and, with the small ALMR core, significantly reduced reactivity.

2.1.1.2 Innovtive Safety Feature of the ALMR Plant

Modular Reactor in Underground Silo

The reactor modules and the intermediate heat transport systems are underground, providing improved protection from tornadoes, missiles, and sabotage. The small thermal ratings of the individual reactor modules ease the task of decay heat removal, and reduce the potential consequences of a core damaging accident.

Seismic Isolation

The reactor module and its safety related systems rest on 31 seismic isolators (Figure 4). The isolators decouple the system from horizontal accelerations in the high frequency range, which is of greatest importance in establishing design margins. The system is being designed to have the capability to meet 0.5g safe-shutdown earthquake criteria.

Passive Decay Heat Removal

Normal reactor shutdown heat removal is through the IHTS and steam generator to the turbine condenser and external heat sink (Figure 3). The auxiliary cooling system (ACS) provides an alternative means of heat removal by direct cooling of the steam generator with forced or natural convection atmospheric air around the steam generator shell. An extraordinary reliable safety-grade backup is provided by the reactor vessel auxiliary cooling system (RVACS). In the rare event that the IHTS becomes unusable during power operation, for example because of a main sodium pipe break or sodium dump, the reactor will scram and the RVACS will automatically and passively come into full operation. Temperatures will rise and heat transfer to the atmospheric air that is always circulating upward around the containment vessel (Figure 4) will increase until equilibrium between reactor heat generation and RVACS cooling is established. The core bulk outlet sodium temperature peaks at about 607 degrees C (1125 degrees F) after about 30 hours, with substantial margin below structural limits and over 345 degrees C (620 degrees F) margin to boiling (Figure 6).

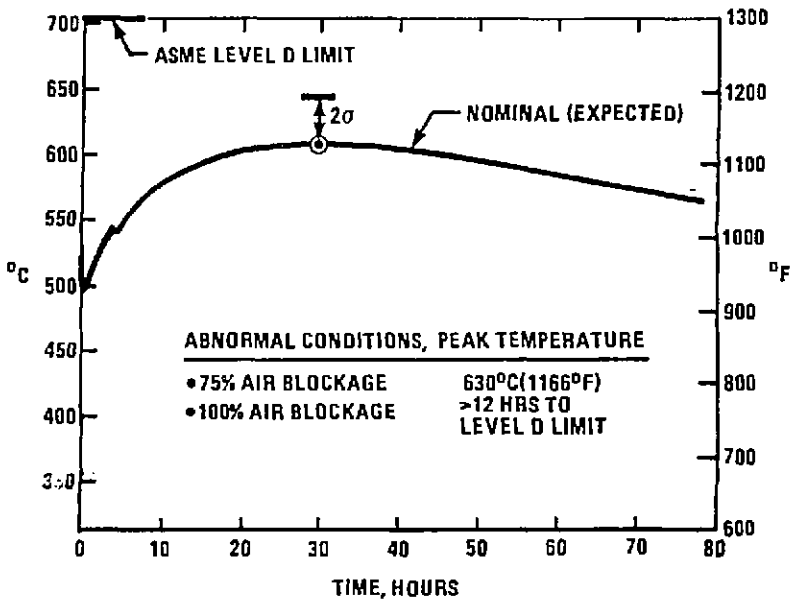


FIG. 6. Core average outlet sodium temperature after loss of all cooling by intermediate heat transport system at full power, with scram.

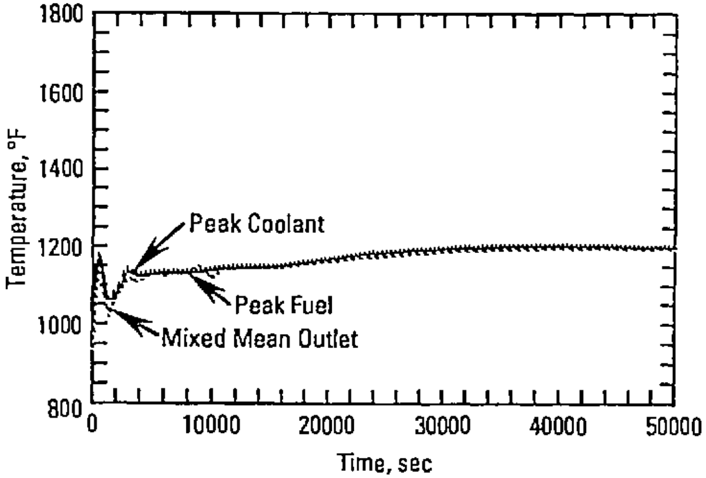
Redundancy of the air flow passages, the characteristic of thermal radiation (dominant heat transfer mode) increasing much more than proportional to rising temperatures, and substantial margins in the design, make the RVACS extremely tolerant of accidental events, such as flow blockages and surface fouling. For example, a massive structural collapse that blocked the equivalent of 75 percent of the air ducting would result in peak temperatures below the ASME Level D structural limit set for the design. An even more extreme case involving complete blockage of the air passages could be accommodated without exceeding ASME Level D limits for over twelve hours.

Reactivity Shutdown and Control

The requirement established for the reactivity shutdown and control system is that the probability of failure to shut down be less than 10^{-6} per demand. The insertion of any one of the six rods will bring the core to shutdown. Each rod can be inserted into the core in three different ways: rod run-in by the plant control system, gravity drop initiated by the reactor protection system, and fast, forceful run-in initiated by the protection system. The reactor protection system is safety grade, automatic, well separated from the non-safety grade plant control system, and located entirely in the reactor module vaults away from the control room. A second and diverse reactivity shutdown system is provided for the very unlikely event that all control rods fail to insert. This "ultimate" shutdown system when activated by the operator, drops boron balls into the central core location (Figure 5) and will bring the reactor to cold shutdown independent of the control rods.

The requirement established for the selected ATWS (anticipated transients, without scram) events are: no significant fuel failures, high margin to sodium boiling, and long-term structural temperatures maintained below the ASME Level D limit (700 degrees C, 1300 degrees F). Figure 7 shows the reactor behavior under a combined condition of loss of forced primary flow and loss of heat removal by the IHTS at full power and without scram. Through the passive reactivity feedbacks and the passive air flow heat removal, the core outlet sodium temperature settles below the established temperature limit of 700 degrees C (1300 degrees F), and high margin to sodium boiling is maintained. Although the fuel clad interface temperature exceeds the 780 degrees C limit for fuel-clad eutectic formation for several minutes, no fuel failures are expected.

Figure 8 shows the reactor behavior under an accidental rod withdrawal at full power without scram, adding 40 cents reactivity at the maximum capability of the control system, 2 cents/sec. The power peaks at about 170 percent and settles at about 135 percent of rated power. The sodium and fuel temperatures again settle out below the 700 degrees C (1300 degrees F) limit, and no fuel failures are expected.



F'G. 7. Core outlet sodium and peak assembly temperatures after loss of primary forced flow ·. d loss of cooling by the intermediate heat transport system at full power and with failure to scram.

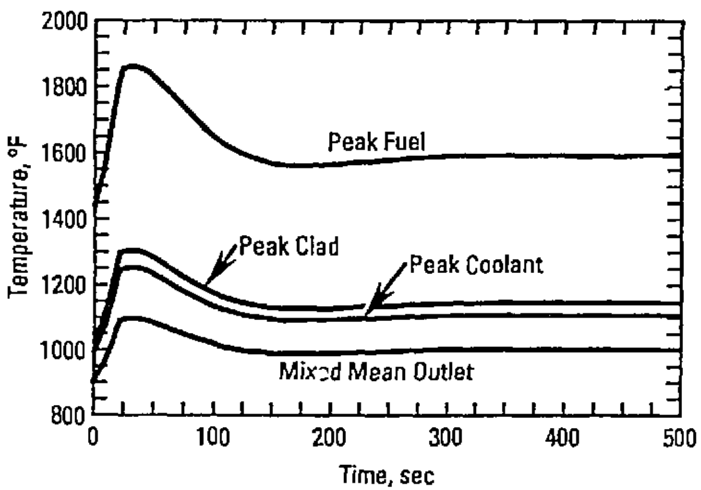


FIG. 8. Core outlet sodium and peak assembly temperatures resulting from withdrawal of control rods at full power and with failure to scram.

Containment

The primary system boundary consists of the reactor vessel, the seal-welded reactor head closure, associated isolation valves, control rod drive housings, instrument drywells and the surfaces of the intermediate heat exchangers (IHXs). The prin. .ry system is completely sealed during power operation and provides a strong barrier designed to contain severe core disruptive events without leakage. Preliminary assessments indicate that energetic events producing several hundred megajoules of mechanical energy and gross core melting can be contained by the core support and primary system structures. The expected energy release from a severe event in the small, metal-fueled core is not more than a few megajoules, and the metal fuel is expected to resolidify under most conditions in a porous, coolable form.

The containment completely surrounds the primary system and represents a separate pressure retaining, essentially leak-tight boundary. It consists of the containment vessel, which backs up the primary vessel, and the upper containment dome, which backs up the head closure (Figure 4).

During power operation, all sodium and cover gas service lines are closed wi.·- double isolation valves at their penetrations through the containment, and all other penetrations in the reactor head closure are seal-welded. There are no penetrations in the reactor or containment vessels. If a leak should occur in the reactor vessel, the containment vessel will retain the primary sodium. The two vessels are sized such that the reactor core, the stored spent fuel assemblies, and the IHX inlets will always remain covered; thus normal flow paths are reserved for cooling.

Even though the primary system boundary is designed to contain severe core disruptive events, the assumed design basis for the containment is such an event with the simultaneous breach of the primary system boundary. It is further assumed that the cover gas escapes through the reactor head closure and carries with it 100 percent of the fission gases and lesser fractions of the other radioactive materials and that air enters the primary system, resulting in a pool fire that consumes all the oxygen available in the containment dome. The resulting pressures and temperatures are within the containment dome design levels of 1.7 bar (25 psig) over-pressure and 370 degrees C (700 degrees F) temperature. The less than 1 percent volume per day design leak rate results in radiation dose levels of less than 1 rem whole body at the plant boundary for the first 36 hours.

2.1.1.3 Safety Approach

There are several elements in the safety design approach for the ALMR.

First, the basic ALMR design is conservative. During power operation the reactor is hermetically sealed, all sodium and cover gas service lines in the head closure are closed with double isolation valves, and all other penetrations are seal-welded. There are no penetrations in the reactor vessel below the head closure. The pressure in the reactor cover gas is approximately atmospheric during power operation. At full power the bulk sodium temperature exiting from the core is 485 degrees C (905 degrees F) and the peak fuel pin linear power is 305 W/cm (9.3 kW/ft). These selections give substantial temperature margins to design limits: over 400 degrees C (720 degrees F) to sodium boiling, 140 degrees C (250 degrees F) to fuel centerline melting, and 150 degrees C (250 degrees F) to fuel clad eutectic formation.

The reactor and its safety-related systems are seismically isolated in the horizontal direction. The relatively small reactor diameter results in a structure that is stiff in the vertical direction and eliminates the need for vertical isolation. The isolator bearings are each an assembly of steel plates laminated with rubber, a design that has been commercially developed and used in non-nuclear applications, and is being further developed and tested to qualify for nuclear service. The reactor system seismic design basis is 0.5 g for safe-shutdown earthquakes, with structural margins to accommodate safely more severe, very low probability earthquakes that approach 1 g acceleration.

Second, very reliable engineered protective systems are provided in the design, for example, the highly automated plant control system (PCS) will maintain the core outlet sodium temperature within specified limits during power operations, including automatic "runback" of the power to a low level if necessary. If an emergency event develops too rapidly for the PCS to control it, then the safety-grade reactor protection system (RPS) located at the reactor module will independently respond by "scramming" the reactor (rapid insertion of the six control rods). The RPS includes substantial internal diversity and redundancy and, thus, is extremely reliable; the probability of it failing to shut down the reactor is estimated to be less than 10^{-6} per demand.

Third, the ALMR is designed so that passive heat removal and reactivity shutdown characteristics will come automatically into play to bring the reactor to a safe, stable state in the unlikely event that there is failure of the active systems.

Fourth, the reactor primary coolant and cover gas boundary and internal structures are designed with large strength margins so that even in the extremely unlikely event that sodium boiling or fuel melting occurs, the reactor primary coolant and cover gas boundary will not be breached and there will be no leak of radioactive materials from the reactor.

Fifth, the ALMR design includes full containment around the reactor to provide barriers against radiation release in the extremely unlikely event that there is a breach of the reactor primary coolant and cover gas boundary.

This approach utilizes both accident prevention and accident mitigation features, and is consistent with the NRC policy concerning advanced nuclear power plants. The objective is to achieve for the ALMR design a probability less than 10^{-6} per plant-year that there will be an accident with a radiation release at the plant site boundary greater than 1 rem whole body dose for the first 36 hours after initiation of the accident or greater than 25 rem whole body dose at a distance of ten miles from the plant thereafter. The 1 rem limit over the first 36 hours is consistent with release limit guidelines proposed by the NRC staff for advanced nuclear reactor designs intended to not require a rehearsed public evacuation plant. The subsequent 25 rem limit is consistent with current federal regulations.

2.1.1.4 Fuel Cycle and Waste Management

The reference fuel for ALMR is metallic U-Pu-Zr alloy being developed by Argonne National Laboratory as described in Section 3.

During power operation plutonium in the driver fuel will be consumed by fissioning, and new plutonium will be created by U238 atoms in the blanket and fuel assemblies capturing neutrons to make Pu239. With the present reference design, slightly more fissile Pu will be made than will be consumed. The estimated compound system doubling time (CSDT) for breeding Pu is about 120 years, which is consistent with current costs of fissile material and DOE cost estimating guidelines for the ALMR program. However, the ALMR has a capability to achieve a CSDT of about 50 years, for utilization when expected future increases of fissile material costs occur and extension of nuclear fuel resources is desired.

At each refueling shutdown, new driver fuel assemblies (containing Pu239 and U238) and new blanket assemblies (containing U238) will be put into the reactor, and spent fuel and blankets will be taken from the reactor to the fuel cycle facility. In this facility, Pu239 will be recovered and used to enrich new fuel, more U238 will be added to provide fertile material, and radioactive wastes will be prepared for disposal. Startup cores of new ALMR plants will use plutonium from processing of spent fuel from LWRs and/or from other ALMR plants in operation.

Spent fuel from reactors contain minor actinides, specifically americium, curium and neptunium, which have extremely long radioactive lives. In the reference IFR metal fuel cycle, the minor actinides will be processed with the plutonium and returned in the new fuel to the reactor. In a fast neutron spectrum, such as in the ALMR, the minor actinides will fission as part of the fuel, creating thermal energy while being reduced to short-lived fission products. Ultimately, these and the other fission products will be taken from the fuel cycle as waste products.

If LWR spent fuel is used as the Pu source for ALMR startup fuel, the minor actinides can be included with the Pu for disposal by fissioning them as a constituent in the ALMR fuel. At the present time there are approximately 100 GWe of LWRs in the U.S. The rate of production of Pu and minor actinides as spent fuel constituents from these plants and their potential future replacements is estimated to be sufficient to provide for startup of the equivalent of approximately one 1395 MWe ALMR per year. Thus, with a relatively moderate rate of deployment of ALMRs in the next century, this approach could provide a means to reduce substantially the challenge of disposal of long life, high level wastes coming from LWRs.

2.2 Licensing

2.2.1 Regulatory Review

Both the ALMR design team and the NRC recognize the desirability of interaction with each other during the design process to assure regulatory approval of the final product. A Preliminary Safety Information Document (PSID) was submitted to the NRC for review in November 1986. This document is similar to a Preliminary Safety Analysis Report (PSAR), but with less detail because of the conceptual nature of the design. During 1987 and 1988, numerous meeting and discussions were held among the design team, the NRC staff, and the Advisory Committee for Reactor Safeguards (ACRS) in the course of the review. The results of the review are the draft Safety Evaluation Report (SER) prepared by the NRC Staff and the review letter by the ACRS reporting the findings.

However, they also expressed concerns about the emphasis on accident prevention to the exclusion of mitigation, and the lack of conventional containment. Therefore, amendments to the PSID were prepared and submitted to the NRC in 1990 augmenting the design with additional accident mitigation evaluations and features, and also adding a low leakage pressure retaining containment dome. The NRC is currently reviewing these amendments.

2.3 Development Strategy

The current advanced conceptual design contract was scheduled for completion in 1992, however, funding curtailment has affected completion of the scope of work on that schedule. Nevertheless, it remains the Government's design for the private sector to come forth with a firm interest to proceed with a preliminary and final ALMR design, as well as constructing a prototypic module, on a cost-shared basis. To that end the industrial design contract team is defining, and with DOE authorization, will implement a plan leading to commercialization of the ALMR. It envisions a team effort between U.S. industry, its international partners, utilities, and DOE. It includes implementation of an ALMR prototype and certification of the design by the NRC. Performance of the detail design, and construction and safety testing of a single reactor prototype would be jointly funded by the DOE and the private sector. Commencement of the detailed design phase by 1995 could permit completion of safety testing of the prototype and certification of the design by 2006. Therefore, the DOE plans to continue development of the ALMR and the associated actinide recycle at a level of effort consistent with budget constraints. International cooperation will continue to be an important part of this program.

3. Integral Fast Reactor Development

The Integral Fast Reactor (IFR) is an innovative liquid metal reactor concept being developed at ANL. This concept exploits the inherent properties of liquid metal cooling and metallic fuel in a way that leads to substantial improvements in the characteristics of the complete reactor system. The IFR concept consists of five major technical features: (1) liquid metal (sodium) cooling; (2) pool-type reactor configuration; (3) metallic fuel; (4) an integral fuel cycle, based on spent fuel pyrometallurgical/electrochemical processing and injection-cast fuel refabrication; and (5) an option for a compact, collocated fuel cycle facility. The reference power plant design, PRISM, incorporates many of the features of the IFR system being tested and proven in the ALMR Program.

3.1 IFR Technology Development

The goals of IFR technology development in the U.S. ALMR program are to confirm metal fuel performance capabilities, establish a reference pyroprocessing method for recycle of metal fuel, evaluate actinide burning benefits as part of U.S. waste management options, assess passive reactor safety characteristics, and demonstrate fuel cycle economics. There is sufficient evidence to indicate that, when utilized in conjunction with the advanced PRISM ALMR design, metal fuel has the potential for improved economics and passive safety performance. It also favors improved waste management options, including the burning of a very large fraction of the longer-life actinides present in reactor spent fuel, which could facilitate future waste disposal activities. A ternary alloy metal fuel (U-Pu-Zr) is the current reference fuel for the IFR.

The DOE IFR technology development program includes five elements:

- Fuel Performance Testing. Fabrication, irradiation testing, performance evaluation, and modeling of metal fuel to demonstrate its performance under normal and upset conditions.

- Core Design R&D. Optimization of the passive inherent safety and economics of metal-fueled cores.

- Safety Tests and Analysis. Demonstration of IFR and metal fuel safety potential, through testing in EBR-II and TREAT.

- Pyroprocess Technology. Development of pyroprocessing and waste treatment methods for incorporation as the reference IFR processing flowsheet.

- Spent Fuel Recycle and Waste Treatment Demonstration. Operation of the full IFR fuel cycle to optimize processes and demonstrate technical and economic feasibility of the system.

3.1.1 Fuel Performance Demonstration

The primary objectives of metal fuel performance demonstration are to demonstrate, through irradiation of individual test assemblies and whole core loadings, the economic and safety performance potential of metallic fuels, and to develop a technology data base, as required, to support advanced reactor design and licensing processes. The demonstration includes the fabrication of fresh fuel and components for irradiation in EBR-II AND FFTF.

The basic physical properties of the ternary alloy IFR fuel and the fuel/cladding interactions over a broad range of compositions and operating conditions are being established. Out-of-reactor experiments are underway to establish the compatibility of the metal fuel with advanced cladding materials, to characterize the distribution of the alloy elements within the fuel, to measure the thermal and physical properties of the fuel, and to validate calculational methods of modeling the fuel behavior.

The irradiation test program has included a comprehensive range of design and operating parameters. The U-Pu-Zr fuel composition has been varied from 0 percent Pu to 28 percent Pu with zirconium variations from 2 percent to 14 percent. Three cladding materials have been used, including the two austenitic materials 316 and D9, and the ferritic/martensitic alloy HT9. The plenum-to-fuel volume ratio, smear density, linear power, and fuel/cladding temperatures have covered a wide range. The maximum burnup achieved to date is 18.4 atom percent and, in general, the steady-state irradiation performance of the IFR fuel has been excellent.

In order to generate a statistically significant data base with which to judge IFR fuel reliability, the standard EBR-II driver fuel alloy, U-5Fs, has been steadily phased out even though fully reliable to its 8 percent burnup goal level. The EBR-II core conversion is now complete, using MK-III subassemblies with D9-clad, U-10Zr driver fuel and 316SS-clad, U-10Zr safety/control rod fuel. The current MK-III driver fuel burnup goal of 10 percent has already been exceeded by on the first qualification subassembly (14.7 percent burnup). A parallel effort to qualify HT9-clad (MK-IV) fuel is underway and to date has achieved a maximum burnup of 8.4 percent. This effort will qualify not only HT9-clad U-10Zr but also HT9-clad U-26Pu-10Zr for ultimate feed to the IFR fuel cycle demonstration.

Irradiation testing in FFTF was initiated to demonstrate that the data base generated in the shorter EBR-II core (34.3 cm core height) is directly applicable in gauging performance of commercial LMR cores approximately three times this height. Axial fuel growth might be expected to show some core-height dependence, and this was the principal driving force for the FFTF tests. Irradiation of two 169-pin FFTF fuel assemblies is now complete. The first test, consisting of D9-clad pins with U-(0,8,19) Pu-10Zr fuel, was discharged after reaching its goal burnup of 10 atom percent. The second test, consisting of a limited number of HT9-clad pins with U-10Zr fuel, was discharged at 5 atom percent burnup. Neither test experienced fuel pin failure and both will soon undergo postirradiation examination. Follow-on tests of HT9-clad U-10Zr fuel pins are now under irradiation in FFTF and have achieved a maximum burnup of over 11 atom percent without breach. Unique instrumentation in the FFTF has confirmed the magnitude of axial growth expected in this metal fuel. These are the only tests currently under irradiation using HT9, a metal fuel and a pin length similar to those proposed for the ALMR. The bulk of these tests will continue irradiation until pin breach or other life-limiting conditions are reached.

Considerable effort is being placed on characterizing the thermal and physical properties of the IFR fuel, in particular the swelling and axial growth of the fuel and the compatibility of various fuel compositions with different cladding materials under operating conditions. Anisotropic fuel swelling has been observed, with the degree of anisotropy a complex function of fuel composition, temperature and fission rate. Axial growth of the fuel column is an approximately linear function of burnup beyond the point of initial fuel/cladding contact, at about 2 percent burnup. Figure 9 shows the average length change of the fuel column as a function of burnup for three fuel compositions: U-10Zr, U-8Pu-10Zr, and U-19Pu-10Zr. The compatibility of fuel and cladding alloys is being studied by a variety of means, including differential thermal analysis, diffusion couples with unirradiated materials, and temperature transient testing of segments of irradiated fuel elements or complete fuel elements. The latter tests have shown generally higher temperatures for fuel-cladding interaction than for those tests done with unirradiated samples. This is most likely a consequence of the complex interaction between fuel and cladding under irradiation. Results to date indicate that the Pu content of the fuel does not seem to lower the fuel-cladding eutectic temperature.

3.1.2 Safety Tests and Analysis

The overall objective of the IFR safety analysis task is to provide the experimental data to validate the unique safety features of the IFR and to fully characterize the totality of safety features associated with metallic fuel. Primary activities are: (1) to conduct TREAT tests to establish the margins to failure for

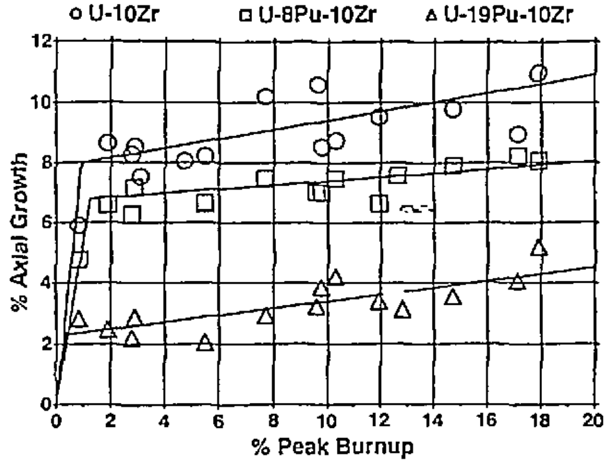


FIG. 9. Burnup dependence of the average axial fuel elongation.

metal fuel and validate the modelling and analysis of the transient behavior of metallic fuel, (2) to conduct analyses to demonstrate the safety margin of metallic fuel for a wide range of reactor sizes and to apply the analyses to the reference reactor concept, (3) to conduct out-of-reactor tests on both unirradiated and irradiated fuel to establish key fuel behavior data under upset conditions, and (4) to conduct analyses of reactor operational transients, anticipated-transient-without-scram (ATWS) events, and local faults to establish margins of safety for metal-fueled IFRs.

No further TREAT tests have been performed during the past year, but analyses of previous tests with metal fuel have shown that transient heating of this fuel under accidents conditions produces cladding loading dominated by the plenum pressure. The similarity of the thermal expansion of fuel and cladding, and the compliant nature of the porous fuel, lead to negligible fuel-cladding mechanical interaction (FCMI) damage of the cladding. Although the FCMI stresses in the cladding may be significant early in the transient, little plastic strain accumulates before fuel creep relaxes the cladding loading to a hydrostatic state that follows the transient increase in plenum pressure. Should the accident sequence proceed to fuel melting, the high fuel porosity, low gas retention, and small fuel density decrease on melting lead to little pressurization of the pin before melting at the top of the fuel column allows molten fuel to expand into the plenum region. Besides delaying g fuel failure (to about four times normal power in an 8-second period overpower transient), this molten fuel extrusion can provide a significant source of negative reactivity feedback.

Following the strikingly successful Inherent Safety Demonstration test series conducted at EBR-II in 1986, a wide range of design basis accidents (including anticipated, unlikely and some extremely unlikely transients) have been evaluated for the IFR concept. These events are found to lead to consequences well within conservatively interpreted acceptance guidelines. The improved passive safety capability of the pool configuration and the improved reactivity feedback response of metal fuel lead to the availability of large design margins of safety. In pool systems, the large primary system heat capacity buffers the primary system so that no reactor scram is required for any combination of balance-of-plant (BOP) faults. In the metal-fueled IFR, the reactivity decrement associated with changing power level is small compared to oxide-fueled reactors. These basic characteristics and the availability of large margins can be exploited to develop simplifications in the plant protection system (PPS) and plant control system (PCS) configurations, leading to the emergence of a new optimum control strategy that could reduce event frequencies and scram demands.

To support further inherent safety demonstrations in EBR-II, to be conducted with the full IFR core in place, the EBR-II PCS has been substantially automated. The reactor is serving as a test bed for advanced control and diagnostic system technology, and a number of advanced diagnostic systems are now in place. These systems are proving highly useful in monitoring safety margins and in providing operating simplicity and flexibility.

3.1.3 Pyroprocess Development

A key element of the IFR concept is its unique fuel cycle, based on a combination of pyrometallurgical and electrochemical processing, or "pyroprocessing." This element of the IFR Program deals with the development of a compact process for recovering plutonium and uranium from the irradiated metallic core and blanket materials, for extracting fission products from them, and for re-enriching the core fuel with plutonium bred in the blanket. To accomplish this, major development efforts are directed toward flowsheet development and process chemistry, process development studies, engineering-scale demonstration of the electrorefining process, and waste treatment/management.

The electrorefining process consists first of the batch dissolution of chopped fuel element segments. An oxidant, $CdCl_2$, is added to convert sodium from the fuel element thermal bond and active fission product metals to their chlorides, which become a part of the molten chloride electrolyte. Uranium is electrorefined by deposition on a solid cathode, and U-Pu fuel elements are electrorefined by deposition in a liquid cadmium cathode. These cathodes are removed from the electrorefiner cell, the cadmium and occluded salt are removed by retorting, and the uranium and/or uranium-plutonium product are consolidated by melting. The residual solids in the cadmium pool constitute the noble metal fission product waste from the process. The salt, which can be made sufficiently free of actinides to be regarded as non-transuranic (non-TRU) material, is a process waste. The rare earth and noble metal fission products may be consolidated in to a metal matrix such as copper to provide a permanent disposable waste form (non-TRU) waste form. Flowsheet variations are being tested with a laboratory-scale electrorefining apparatus, with results being compared against calculations based on an electrochemical model of the overall process. Good agrement has been obtained between the measured and calculated compositions of the salt and metal phases. An engineering-scale version of the electrorefiner is being used for process development and system optimization.

3.1.4 Fuel Cycle Demonstration

The objective of this program element is to verify and quantify the economic potential of the IFR metal fuel cycle by means of a comprehensive demonstration of all aspects of the fuel cycle under conditions representative of future commercial practice. The demonstration will be carried out in the refurbished Fuel Cycle Facility at the ANL-Idaho Site over the period 1991-1995, using spent IFR fuel discharged from the EBR-II reactor. Modifications to the former Hot Fuel Examination Facility-South to include installation of full-scale IFR pyroprocessing unit operations are now planned to commence in October, 1991. The pyroprocessing demonstration will take place on a scale and at a throughput rate which facilitate a reliable economic evaluation of the process. Once in full hot operation, about 60-90 EBR-II fuel subassemblies will be processed per year. Unit operations to be demonstrated range from initial fuel subassembly breakdown to final waste treatment and packaging for disposal. At this time, the EBR-II complex will then be in full operation as an IFR prototype, with fuel at goal burnup levels, fuel being recycled (i.e., processed and refabricated) to the reactor in a closed fuel cycle, and low-volume non-TRU waste products being shipped off-site for disposal.

3.2 IFR Concept Development

The objective of this program is to verify design features, safety margins, and cost savings of the IFR concept and to confirm performance expectations of the required new components and materials. Extensive use is made of the national laboratories and engineering test centers to verify predicted behavior of components, materials, and systems. With the selection of the reference concept, resources will be concentrated on the technologies required for the chosen advanced reactor concept and remaining technical uncertainties will be resolved.

3.2.1 Systems Technology

Systems technology includes work on components, advanced instruments and controls, and auxiliary systems. This R&D advances the performance of current technology, which is needed to assure that desired cost savings are realized in the construction and operation of the proposed reference advanced ALMR.

At the Energy Technology Engineering Center (ETEC), major emphasis is currently on steam generator development. Helical coil steam generator testing has been suspended, and activities now encompass initiation of a double wall tube steam generator test program under a cooperative contract with the Japanese Atomic Power Company (JAPC) that includes testing of a 70 megawatt model steam generator built by Westinghouse, and of a JAPC few-tube model steam generator, and testing of advanced technology and materials (e.g., modified 9 Cr-1 Mo steel and improved leak detection systems). Activities at ETEC also include testing and test preparations in support of other (non-FBR) reactor concepts, including the gas-cooled and heavy water cooled New Production Reactors, and SP-100, an advanced power plant for space applications.

The advanced instruments and control program element includes the design, test, and performance demonstration of new, state-of-the-art instrumentation and control systems, such as radiation detection, sodium pressure and temperature measuring, automated plant control, "smart" sensors and diagnostics, nondestructive testing and in-core neutron flux monitoring, which will reduce plant cost and improve the licensability of advanced liquid metal reactors.

Work continued on the long-term task of providing for testing and validation of advanced control system designs by simulation, with emphasis on applications of parallel processing to improvement of simulation speeds and to real-time simulation techniques. A large reactor simulation program was converted to parallel processor code and is in the final stages of debugging. Significant speedups were obtained with parallelization, with the code currently running up to 600 times real time. Lessons and techniques developed in this exercise will be applied to the advanced controls program demonstration projects, as appropriate.

Research and development is also required to determine the best design for major auxiliary systems, including fuel handling, vessel support, in-service inspection, sodium leak detection, and remote maintenance. Sufficient test and design verification data are needed so these systems can be clearly specified and accurately costed. R&D is directed at those critical features that differ significantly from FFTF or EBR-II operating systems and that offer potential for significant future cost savings, improved reliability, and/or increases in plant availability. One area generating interest is a bottom-support plant design that offers the potential of reduced seismic loads and lower cost.

4. OXIDE FUEL CYCLE R&D

Oxide fuels remain the primary backup for metallic fuels in both the IFR and PRISM concepts. Oxide fuels continue to be the preferred fuel for most other nations who are developing the fast breeder reactor. For many years, the U.S. and Japan have had extensive cooperative efforts underway. Development efforts have included long-life oxide fuel performance and an advanced fuel cycle system based on the well known Purex process. This cooperation continues.

The oxide fuel cycle R&D program currently consists of collaborative efforts with Japan in advanced reprocessing systems development (see 4.2) and fuel performance testing of mixed oxide fuel in FFTF and EBR-II for Japan's MONJU reactor.

4.1 Oxide Fuel Development

The primary effort of the FFTF fuel development program during 1990 was additional development of ALMR fuel and blanket assembly designs that concentrated on controlling pin and duct distortion by reducing the swelling behavior of the structural materials. The Core Demonstration Experiment (CDE), currently being irradiated in the FFTF, clearly demonstrates the capability of mixed-oxide fuel to achieve burnups in excess of 200,000 MWd/MTM and fast neutron fluences in excess of 30×10^{22} n/cm^2 using the very low swelling ferritic-martensitic alloy HT9.

The CDE consists of three parts: (1) a partial prototypic LMR core loading of ten fuel and six blanket assemblies arranged in a heterogeneous configuration in the center of the FFTF core, (2) two lead fuel tests designed to operate under one and two sigma thermal conditions, and (3) two low power fuel tests designed to verify concepts of an ultra-long life LMR core. Each CDE fuel assembly contains of 169 large diameter (6.858 mm) pins with mixed-oxide annular fuel pellets and each blanket assembly contains 91 large diameter (9.906 mm) pins with solid pellets of depleted uranium dioxide. The duct, cladding and wire wrap are HT9.

The CDE partial core loading began irradiation in September 1986 and successfully completed irradiation in April 1990 after a 3.5-year resident time. This pioneering work conclusively demonstrates the advantages of low swelling materials. The fuel attained peak exposures of 163,900 MWd/MTM and 23.3×10^{22} n/cm^2 and the blanket attained peak exposures of 43,100 MWd/MTM and 22.8×10^{22} n/cm^2. No duct elongation was observed and no distortion sufficient to cause an increase in withdrawal loads of the assemblies was observed. One fuel assembly from this partial loading continues irradiation in FFTF. As of March 1991, this fuel assembly has attained a peak exposure of 194,000 MWd/MTM and 29×10^{22} n/cm^2.

The CDE lead tests continue their irradiation in FFTF and have attained a peak exposure of 218,000 MWd/MTM and 34×10^{22} n/cm^2 as of March 1991. Concurrently, the CDE low power tests have attained a peak exposure of 109,000 MWd/MTM and $13 \times 10^{2}2$ n/cm^2. At this time, although very high exposures have been reached, none of the CDE test pins have breached.

Additionally, irradiation continues on the two prototypic fuel and one prototypic blanket assemblies for Japan's MONJU LMR. These tests utilize cladding and duct components provided by Japan. As of March 1991, the fuel assemblies have attained a peak exposure of 134,000 MWd/MTM and 19×10^{22} n/cm^2. The MONJU blanket assembly has attained a peak exposure of 21,000 MWd/MTM and 21×10^{22} n/cm^2 and is planned for removal at this time.

4.2 Oxide Fuel Reprocessing

Fuel reprocessing activities on the Shear-Leach-Purex aqueous process are continuing in the Consolidated Fuel Reprocessing Program (CFRP) at the Oak Ridge National Laboratory (ORNL). Much of the work is carried out under a jointly funded collaborative agreement with PNC of Japan. The goal of the program is continued development of reprocessing technology and support of the design and construction of the Recycle Equipment Test Facility (RETF) in Japan. Each piece of equipment in the RETF will be prototypical of those of a future FBR fuel recycling pilot plant in Japan and will provide a valid technology demonstration under hot operating conditions. The reprocessing program at the CFRP is focusing primarily on this collaborative program while completing ongoing activities with the United Kingdom Atomic Energy Authority and the Commissariat a l'Energie Atomique of France.

The concepts that have been and are being developed at ORNL are compact, high throughput devices that are designed both to be reliable and readily maintainable by remote means. Equipment components, except for the heavy mechanical head-end devices, will be mounted on racks that facilitate in situ remote maintenance and also provide for the rapid replacement of individual components or entire racks, if necessary. An electromechanical master-slave manipulator is in operation at CFRP that is highly dexterous, has force feed back to the operator and is itself remotely maintainable. In-cell samplers have been developed and tested that will minimize cell penetrations, reduce exposure to analytical personnel and simplify operations. The entire process will be fully instrumented and will provide for both enhanced process control and enhanced safeguards. Because cell penetrations are minimized and in cell maintenance maximized, a sealed cell concept is possible that will allow a low-flow ventilation system to be used that could include an inert cell gas. This in turn would essentially eliminate any in-cell fire hazard.

Many of the concepts jointly developed by the CFRP and PNC are being incorporated into the reference design of the RETF. Testing of these concepts under prototypic conditions will provide the basis for a proven reprocessing facility that can be made with low capital and operating costs, minimal personnel exposure, improved effluent control and increased safety and safeguardability. Major components/systems currently under test and evaluation include a laser disassembly machine, a rotary dissolver, and centrifugal contactors for separations.

5. TEST FACILITIES

The major facilities advancing the LMR effort are located at three sites within the United States:

- Argonne National Laboratory-West (ANL-W), Idaho Falls, Idaho
- Westinghouse Hanford Company (WHC), Richland, Washington
- Energy Technology Engineering Center (ETEC), Santa Susana, California

In addition to the above facilities, the program utilizes other facilities for advanced reactor R&D. One of the more prominent of these facilities is the Integrated Equipment Test (IET) facility at Oak Ridge National Laboratory (ORNL). The IET provides a key capability in fuel reprocessing technology and is used in a number of international collaborative efforts.

As noted earlier it is probable that the FFTF will be closed down in the near future.

5.1 Argonne National Laboratory-West (ANL-W)

The ANL-W site is the technology center for the U.S. metal fuel development program, for reactor operations, for metal fuel reprocessing and refabrication, and for waste treatment demonstrations verifying the closed metal fuel cycle. The facilities are described below:

5.1.1 Experimental Breeder Reactor-II (EBR-II)

EBR-II, a metal fueled LMR power plant, has been in continuous operation since 1964. In 1989, it achieved a plant capacity factor of 37.0 percent, however, the 10-year average plant factor was 68.17 percent. The original mission of EBR-II was that of a complete pilot plant for proving the liquid metal fast breeder reactor (LMFBR) system. The reactor, in conjunction with the adjacent Fuel Cycle Facility, demonstrated the concept of a fast breeder power plant with an integral fuel cycle. During 1968-69, with the successful completion of the original mission, a substantial modification program was initiated to convert the EBR-II reactor into a fast reactor test facility.

EBR-II's long and successful operating history provides an important source of information on the long-term reliability of LMRs. Major programs being conducted in EBR-II include metal fuel irradiation testing and demonstrating the inherently safe response of a metal fueled, pool LMR to plant upsets. EBR-II also serves as an important test bed for key features of innovative LMR designs, such as flexible pipe joints, improved materials, and instrument and control system improvements. Other major tests currently being conducted include those to determine the efforts of running-beyond-cladding-breach, and response of oxide fuel to operational transients in a joint U.S./Japanese program.

The As Low As Reasonably Achievable (ALARA) approach to radiation exposure combined with other EBR-II attributes has resulted in collective man-rem exposures two orders of magnitude less than for commercial power, light water reactors.

The future utilization of EBR-II includes continued support of the IFR development and metal fuel performance demonstration program, completion of the Phase II U.S./Japanese program of oxide fuel operational reliability testing, and irradiation testing of fuels and other materials for the space and defense power program.

5.1.2 Zero Power Physics Reactor (ZPPR)

The ZPPR is an experimental critical facility in which different reactor core designs can be mocked-up, operated at a low power, and characterized according to the physics properties of the core and surrounding regions. The ZPPR provides experimental physics data for the design of fast reactor demonstration plants and large fast reactor central-station power plants. Operational and design parameters such as critical mass, control rod worth, power-generation distribution, breeding-blanket effectiveness, and neutron flux on support structures are measured for configurations that exactly duplicate the neutronics of the proposed design. Also measured and confirmed are safety-related parameters fundamental to the demonstration of a safe design, such as the Doppler coefficient and the sodium-void coefficient.

5.1.3 Transient Reactor Test Facility (TREAT)

The TREAT reactor is an air-cooled, thermal, heterogeneous reactor used to simulate postulated reactor transients and transient undercooling events. The primary mission of the TREAT reactor is to conduct safety-related tests in support of the Liquid Metal Reactor Program. Tests include overpower transient tests on fuels to determine fuel dynamic behavior during reactor excursions, overpower transient tests to investigate fuel-coolant interaction phenomena, steady-state power tests with loss-of-flow to investigate coolant expulsion and related phenomena, and combinations of loss-of-flow and transient-overpower tests. TREAT also provides neutron radiography services for experimental fuel irradiation programs and other experiments.

5.1.4 Hot Fuel Examination Facility (HFEF)

The HFEF consists of an air-atmospheric and an argon-atmospheric hot cell, which provides capabilities for remote assembly and disassembly of irradiated subassemblies and loops, and for examination of fuel elements and material specimens. The HFEF examination capabilities include precision gamma-scanning profilometry and other dimensional measurements, weight determinations, metallography, photographic and visual observations, eddy current and ultrasonic nondestructive testing, and neutron radiography.

5.1.5 Fuel Cycle Facility (FCF)

The previously designated HFEF-South facility consisting of an air-atmospheric and an argon-atmospheric cell is currently being modified to provide for the development and demonstration of IFR metal fuel pyroprocessing and waste processing technology.

5.2 Westinghouse Hanford Company (WHC)

The WHC facility at Hanford serves as an irradiation testing center for LMRs, fusion, isotope production, space and defense power systems, and cooperative international programs. The key facilities used for LMR development are described below.

5.2.1 Fast Flux Test Facility (FFTF)

As mentioned earlier, the Department of Energy has announced its decision to close down the FFTF. This decision was based on the lack of a mission that justified the continued, high operating costs for the FFTF. It was further determined that the Advanced Liquid Metal Reactor development program could continue, on schedule, using the facilities at the Idaho National Engineering Laboratory. The start of shutdown has been deferred pending the outcome of efforts to secure non-DOE sources of funding for its operation.

The FFTF, a fuel and materials test reactor, began operation in 1982 and is currently nearing the end of its eleventh operating cycle. Its excellent performance is shown by the FFTF Operating Histogram (Figure 10) and Annual Operational Performance data (Figure 11). The plant achieved an Operational Efficiency Factor of 93.2 percent in 1990, a measure of ability to achieve planned activities.

The FFTF completed Cycle 11B operating in October 1990, accumulating 218.2 EFPD in its two subcycles. Cycle 11C operation started in late December 1990 and at the end of February 1991 achieved 2000 EFPD since the beginning of Cycle 1, operation, registering 62.9 EFPD during Cycle 11C. The highest burnup fuel assembly in the core during Cycle 11C achieved a burnup of 218 MWd/kgM at the end of February 1991.

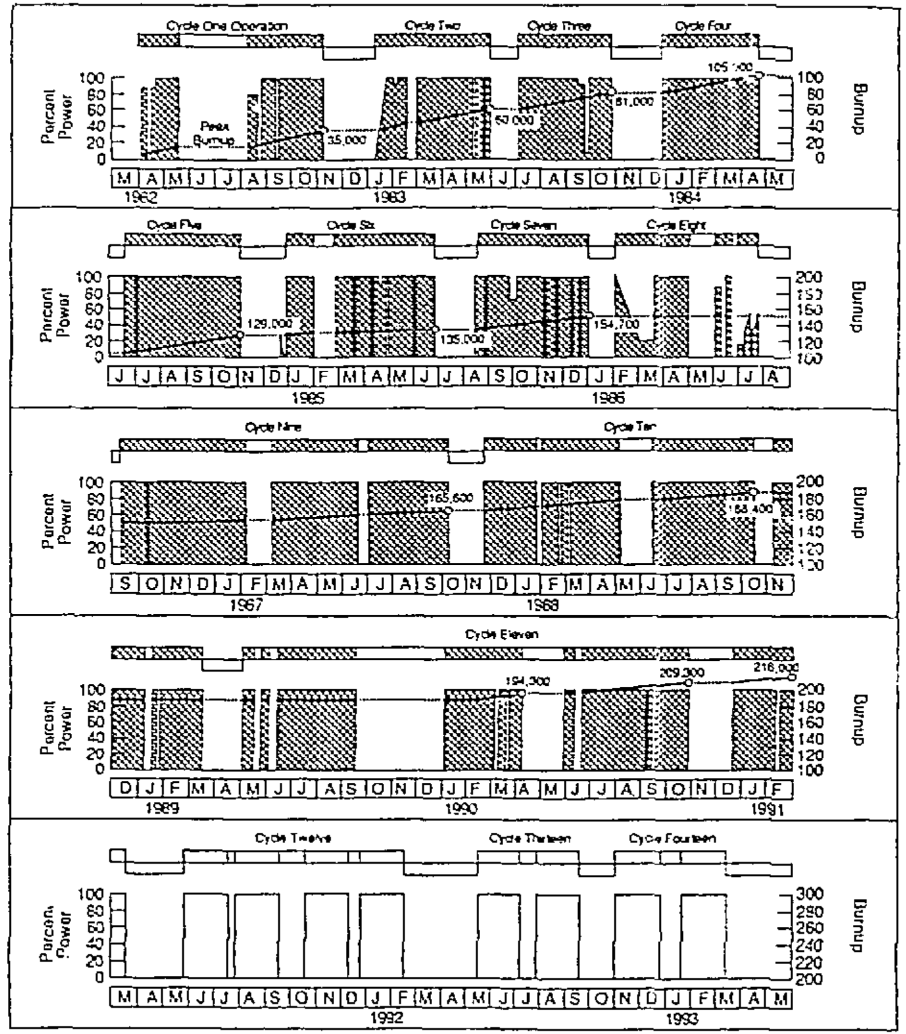
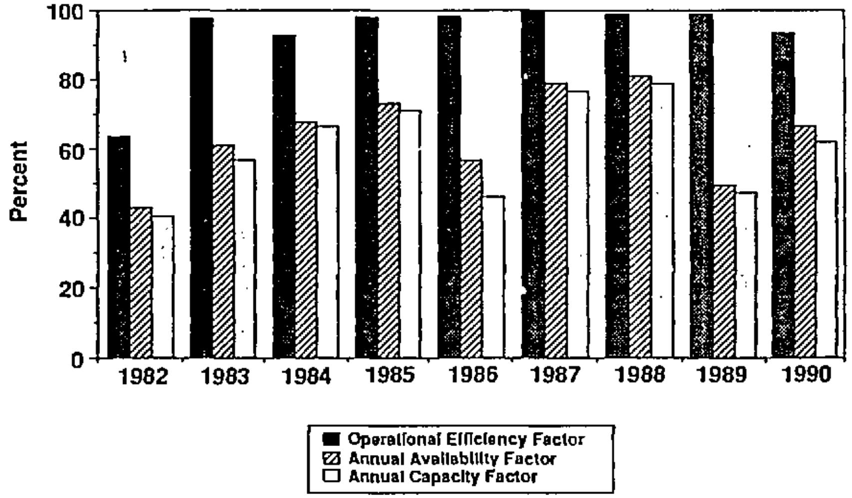


FIG. 10. FFTF operating histogram.



Operating Statistics

	Cycle 1	Cycle 2	Cycle 3	Cycle 4	Cycle 5	Cycle 6	Cycle 7	Cycle 8	Cycle 9	Cycle 10	Cycle 11 (2/28/91)
EFPD For Cycle:	101.5	100.5	101.5	109.5	127.7	134.0	122.8	63.0	341.8	385.3	397.4
Total Plant EFPD / End Of Cycle:	134.3	234.8	336.3	445.8	568.5	702.5	825.3	888.3	1230.1	1615.4	2012.9
Cycle Capacity Factor (%):	50.3	83.1	90.5	99.5	93.5	74.9	90.3	38.9	86.8	80.1	59.8
Availability Factor (%):	53.0	90.6	99.0	100.0	94.6	78.5	94.6	57.9	89.8	80.4	62.8
Number Of Experiments:*	62	64	52	51	52	41	31	35	46	50	48
Maximum Fuel Burnup At End Of Cycle (Mwd/MT):	35,000	60,000	81,000	105,000	129,000	135,000	154,700	154,700	165,600	188,400	216,000

*The Total Number Of Experiments Irradiated During Any Given Cycle Includes DE (Tracked Through EOL), LPMA And GEM.

	1982*	1983	1984	1985	1986	1987	1988	1989	1990
Capacity Factor (%):	40.5	56.9	66.4	71.0	46.2	76.5	78.5	47.8	61.8
Availability Factor (%):	42.8	61.1	67.6	73.0	56.8	78.7	81.2	49.7	66.3
Operational Efficiency Factor (%):	63.5	97.8	92.6	98.0	98.1	100.0	98.9	99.0	93.2

* Reporting Began At Start Of Cycle 1 On April 16, 1982

FIG. 11. FFTF annual operational performance data.

Both the LMR Materials Open Test Assembly (MOTA) and fusion materials MOTA experiments have operated very well and produced valuable data throughout the year. During April 1991 the two MOTA vehicles will be removed from the FFTF and their experimental contents reconstituted into one MOTA vehicle for continued irradiation.

The Core Demonstration Experiment (CDE) configuration successfully completed irradiation to its three-year reference lifetime goal which was reached in April, 1990. Five CDE fuel assemblies continue irradiation with three of these having burnup levels of 194 to 218 MWd/kgM and are well on their way toward a five-year or greater lifetime.

5.2.2 Fuel and Materials Examination Facility (FMEF)

The FMEF is a modern facility designed and built originally as a fuels and materials examination facility and greatly modified to provide the capability for LMR fuel fabrication and plutonium (or other fissile material) storage. The facility has never been fully activated although i . s the latest design, safety, environmental compliance, and safeguards standards. Future utilization of this facility will be influenced by DOE decisions to shut down FFTF operation.

5.3 Energy Technology Engineering Center (ETEC)

The ETEC provides testing capabilities for developmental hardware, software and instrumentation in non-nuclear environments. Key facilities are described below:

5.3.1 The Sodium Component Test Installation (SCTI)

The Sodium Components Test Installation (SCTI) is a 70 megawatt (thermal) test facility that is used primarily for steam generator tests. The sodium heat transport system contains two 35 megawatt (thermal), natural gas-fired sodium heaters, an 8000 gpm main circulating pump, and supporting services for steady state and transient test operations. SCTI includes a 70 megawatt steam and feedwater system which permits once-through or recirculating operation of test steam generators or tests of other high temperature, high pressure steam and water systems. The principal operating mode entails delivery of steam to the turbine of the Power Pak cogenerating system for production of 25.5 megawatts of electricity for distribution to the Southern California Edison power grid. The SCTI can be operated independently of Power Pak, particularly for the performance of severe transient tests; these transients would not be prudent to perform in a nuclear facility, but are safely performed in the SCTI, which has been designed for this purpose. The facility currently utilizes three separate test positions for steam generators.

5.3.2 The Sodium Pump Test Facility (SPTF)

SPTF was designed primarily for large sodium pump testing and the concurrent testing of large sodium flowmeters. The facility has a flow capacity of 100,000 gpm at 210 psig developed head and 1100 degrees Fahrenheit. The system has a turndown capacity to less than 100 gpm. Electric power is available for motor drives up to 15,000 hp. The system includes two test stands for large pumps and is designed to produce strong upramp or downramp thermal transients during pump operation. A space vacuum test capability is also available in this facility.

5.3.3 The Transient Test Facility (TTF)

TTF is used to simulate the effects of process fluid transient temperatures on plant components. Simulations for hot water, liquid metal slurries, and molten salts can be achieved by directing hot or cold high-velocity inert gas through thermally preconditioned test articles. TTF also contains a large hydraulic loading structure mounted on a massive, steel-reinforced, concrete base, which permits simultaneous mechanical and thermal stress testing of components up to 32 by 40 by 25 feet. The TTF complex also includes the Fragility Test System (FTS) and the Seismic Isolation Test Fixture (SITF). The FTS uses TTFs 500,000 lb. seismic mass, which is set in bedrock, as a base for high-level seismic failure (fragility) testing of piping systems and components. a synchronous, four-table hydraulic shaker system can provide accelerations up to 30G. The SITF is a test bed designed and constructed to simulate earthquake effects on large, flexible bearings that can isolate reactor and non-reactor components, systems and buildings and other structures from such effects.

5.3.4 The Steam Accumulator Blowdown Evaluation Rig (SABER)

SABER is used to evaluate large nuclear and fossil fired steam power plant equipment in a blowdown mode. Steam flow rate in excess of 10 million pounds per hour can be developed. Dry or wet steam can be used at pressures to 500 psi, with higher pressures available. Test article reaction loads can be accommodated to one million pounds, with built-in steam quenching and fast-acting control valve capabilities. Types of testing that can be accommodated include acceptance, qualification, design development, off-design performance, noise and vibration, and failure testing.

6. INTERNATIONAL COOPERATION

The United States has been a strong advocate of international cooperation within the fast reactor development program. Department of Energy cooperative activities with the other nations within the International Working Group for Fast Reactors dates back to 1969, and over the ensuing years the extent of cooperation has increased. Even the 1983 shift in the U.S. development direction, from large, oxide-fueled reactors to small, metal-fueled ones, did not lessen our international cooperation. The even more recent U.S. emphasis on using fast reactors for actinide recycle in anticipated to further enhance collaboration.

6.1 U.S./Japan Activities

Currently, the U.S. has fast reactor development agreements and contracts for cooperation with the Power Reactor and nuclear Fuel Development Corporation, the Central Research Institute of the Electric Power Industry, and the Japan Atomic Power Company. The activities include safety and operating experience with oxide fuels, component reliability, materials data, shielding experiments, steam generator studies, and the pyroprocess development for metal fuel reprocessing. Cooperation is very active within these technologies.

6.2 U.S./European Activities

Cooperation continues with France, the Federal Republic of Germany and the United Kingdom. Significant progress has been made towards reaching a new agreement with the existing European fast reactor consortium, taking into account the emerging European community. This past year emphasis was directed at reactor safety and the differences accruing between large, oxide-fueled plants and small, metal-fueled plants, and economic comparisons of these different approaches.

7. CONCLUSION

The U.S. continues to develop the advanced liquid metal reactor concept with a metal fuel cycle that may incorporate actinide recycle from spent light water reactor fuels. This concept is part of the National Energy Strategy that has recently been released by DOE for making nuclear power a viable part of our nation's future energy supply. For the foreseeable future, however, evolutionary and advanced light water reactors are expected to provide the major, new nuclear contribution, and for this to occur, major emphasis must be placed on reducing the uncertainties and promoting stability in the licensing process. With these actions, nuclear power could flourish and without them, nuclear power in the U.S. could all but disappear by the year 2030.