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PROGRESS AND STATUS OF THE INTEGRAL FAST REACTOR (IFR) DEVELOPMENT PROGRAM

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INTRODUCTION

In the Integral Fast Reactor (IFR) development program, the entire reactor system--reactor, fuel cycle, and waste process is being developed and optimized at the same time as a single integral entity.

The ALMR reactor plant design is being developed by an industrial team headed by General Electric and is presented in a companion paper. Detailed discussions on the present status of the IFR technology development activities in the areas of fuels, pyroprocessing, safety, core design, and fuel cycle demonstration are presented in the other two companion papers that follow this.

EBR-II OPERATING EXPERIENCE

Much of the IFR technology base stems from the Experimental Breeder Reactor No. II (EBR-II). EBR-II is a fast spectrum liquid metal cooled reactor (LMR) operating at a thermal power of 62.5 MW, producing ~20 MW of electricity. It was designed by Argonne National Laboratory and constructed at the Idaho National Engineering Laboratory in the late 1950s. Criticality was reached on November 11, 1963, and power operation began in 1964.

EBR-II has now entered its 28th year of operation. Many years of excellent operating experience have demonstrated the feasibility of conducting routine and special maintenance on sodium systems and components. In general, the concerns earlier expressed for the maintenance, modification, and repair of intrinsically radioactive and sodium-contaminated components have not materialized.

Sodium is noncorrosive to the metals used in the LMR reactor structures and components. Radioactive corrosion products are not formed in any significant amounts. Radioactive corrosion products circulating and depositing around the system make access for maintenance difficult. This is an increasingly important problem in the light water reactors. World-wide experience has now demonstrated that this is not a serious problem in the experience has now demonstrated that this is not a serious problem in the LMR. Because of this characteristic, access for maintenance is easy and radiation exposures to plant personnel are expected to be very low. For example, no exposures are expected during maintenance and inspection of the steam generator, turbine generator, steam and feedwater pumps and equipment, etc. Even after more than 27 years of operation, EBR-II as well as other LMR reactors abroad, are all experiencing less than 0.2 man-Sv/a personnel exposure, as compared to some LWRs that now approach 10 man-Sv/a.

Noncorrosive coolant also implies reliable sodium components performance and improved plant availability. For example, the LWR steam generator tube failures are mostly caused by the water chemistry and the accumulating corrosion products in the shell-side crevices, etc. In LMR steam generators, noncorrosive sodium flows through the shell-side and the corrosion product accumulation in crevices is minimal. Steam flows inside the tube where the simple geometry prevents corrosion product accumulation.

A unique feature to the design of EBR-II is the pool-type concept, in which the entire primary systems including the core, primary pumps, intermediate heat exchangers and primary pipings are submerged in a molten sodium pool. The regular conformation of the primary containment tank and the lack of nozzles and penetrations greatly simplify design, construction and inspection activities. All systems that contain primary (radioactive) sodium are located within the primary containment barrier. The effects of sodium leakage from primary

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components are effectively confined to the primary tank. The submersion of all primary components under constant temperature sodium greatly reduces problems of thermal stress. The effects of loss of pumped coolant flow are considerably mitigated, since the core and blanket will always remain covered with sodium. The pool configuration eliminates the large number of inerted steel-lined cells to house the individual primary system components, and it also provides thermal inertia necessary to achieve passive safety.

The pool-type design has been adopted in PFR, Phenix, SuperPhenix, and BN-600. However, the EBR-II pool design is unique in that the cold pool provides a constant low-temperature, low-pressure reactor vessel boundary. The cold pool and coolant boundary eliminate the need to attach redan structures to the vessel wall and also eliminates complex baffle arrangements to cool the hot pool and reactor vessel interface.

Due to a small size, EBR-II has a negative sodium void reactivity. This helped in the various safety analyses for the reactor operation. It is difficult to achieve negative sodium void reactivity in a large core and it usually involves tradeoffs with other performance indices, such as burnup reactivity swing. These tradeoffs require further evaluation. Quite independent of this, the metallic fueled EBR-II core is a key factor in achieving inherent passive safety, which was demonstrated by two landmark tests conducted in EBR-II on April 3, 1986. The first test was loss-of-flow without scram and the other loss-of-heat-sink without scram. These tests demonstrated that the unique combination of the high heat conductivity of metallic fuel and the thermal inertia of the large sodium pool can shut the reactor down during these potentially very severe accident situations without depending on human intervention or operation of active, engineered components.

The EBR-II steam generators have operated without a single tube leak over 27 years of continuous service. The EBR-II steam generator system consisted originally of eight natural-circulation evaporators, two once-through superheaters, and a single horizontal steam drum. The evaporators were arranged in two rows of four each and connected to the steam drum by individual risers and downcomers. Steam separation takes place within the drum, and dry saturated steam is routed from the top of the drum through parallel-connected superheaters to a common header supplying the turbine.

The evaporators and superheaters are straight-tube heat exchangers with sodium flowing through the baffled shells. Except for the baffle-nest material of Type 304 stainless-steel, all construction material is 2-1/4 Cr-1 Mo ferritic-steel. The only difference between the evaporators and superheaters are the core tubes that are installed in the heat-transfer tubes of the superheaters to increase the steam velocity, and the superheaters are inverted in relation to the evaporators.

The design minimizes the possibility of interaction between sodium and water/steam by using "duplex" tubes and double tubesheets. The outer tube is welded to the sodium tubesheet to form the sodium cavity, and the inner tube is welded to steam tubesheet. With this design no single weld, tube, or tubesheet separates the sodium from the water/steam.

Two types of duplex tubing were used in the fabrication of the units. Four evaporators and one superheater contain "mechanically-bonded" tubes, and the other units contain "metallurgically-bonded" tubes. Fabrication of both types of tubes consisted of placing the outer tue over the inner tube and drawing the tubes together through a die and over a pin, leaving the double-tube undersized. This operation was followed by expanding the duplexed tube to its specified outside diameter by drawing a pin through the tube. In the case of the metallurgical tubes, prior to drawing the tubes, the outside of the inner tube was first coated (electroplated) with 0.13 mm of pure nickel and the inside of the outer tube with 0.013 mm of KANOGEN nickel (11-13 wt% phosphorous). Following the drawing operation, metallurgical bonding required a final operation of heating to flow the nickel nickel-phosphorous alloy between the two tubes to produce a brazed tube-to-tube bond. The heating operation annealed out the radial prestress, which was introduced during the drawing operation. The mechanically bonded tubes were left in the stressed condition of the outer tube in tension and the inner tube in compression.

TECHNOLOGY DEVELOPMENT PRIORITIES

Metal fuel is key to realizing the inherent passive safety potential of the IFR and, importantly, it also allows compact pyroprocessing. Therefore, development of a complete technical database on metal fuel performance is an essential element of the IFR Program. The irradiation test program has included a comprehensive range of design and operating conditions. The U-Pu-Zr composition has varied from no plutonium to 28 wt% plutonium with zirconium variation from 2 to 14 wt%. Three cladding materials have been used, which include the two austenitic claddings, SS316 and D9, and the ferritic cladding, HT9.

The lead irradiation tests began in February 1985, and reached a burn up level of 18.4 atom%, far exceeding their design target of 10% burnup. Interlinkage of gas bubbles and rapid gas release at 1 to 2% burnup mitigates fuel-clad mechanical interaction because the resulting porous fuel is plastically compliant at temperature. This is the key factor for achieving a high burnup with metallic fuel.

Irradiation tests to date have clearly demonstrated excellent performance characteristics of the metallic fuel in both steady-state and off-normal operating conditions. EBR-II is now fully loaded with the IFR fuel alloys and statistically significant fuel performance data are being generated.

Pyroprocessing, which utilizes high temperatures and molten salt and metal solvents, can be advantageously utilized for processing metal fuels because metal is both a suitable feed for such processes and the product as well and is thus suitable for fabrication into new fuel elements. Direct production of metal as the product avoids the further chemical conversion steps necessary in reprocessing by the conventional Purex solvent extraction process.

Electrorefining is the key step in the IFR pyroprocessing. Following disassembly of the fuel assemblies, the fuel pins are chopped into short lengths, which are packaged to form a batch for dissolution in the electrorefiner. Bulk uranium, especially from the blanket, is electrorefined by deposition on a solid cathode. The remaining uranium along with plutonium and other minor actinides are electrorefined by deposition into a liquid cadmium cathode. The cathodes are removed from the electrorefiner cell, the cadmium and occluded salt removed by retorting, and the uranium or uranium-plutonium-actinide product is consolidated by melting.

To assure criticality-safe configuration during retorting process, the cathode deposit size of 10 kg has been selected as the reference. Uranium deposits at the 10-kg scale are now routinely produced in the engineering-scale electrorefiner facility. The deposit rate of 10 kg in about 24-hour period promises economic throughput for commercial-scale operation when multiple electrodes are employed to increase the batch size.

The plutonium electrotransport to liquid cadmium cathode has been demonstrated on laboratory-scale involving few hundreds of grams. This combined with simulated uranium deposits in liquid cadmium cathode indicate that large-scale plutonium electrotransport would be successful in the EBR-II Fuel Cycle Facility as well.

APPLICATION OF PYROPROCESS TO LWR OXIDE FUEL

In IFR pyroprocessing, minor actinides accompany the plutonium product stream. Therefore, actinide recycling occurs naturally in the IFR, which is another important attribute as a next-generation reactor concept. A logical question is whether the pyroprocessing approach can be extended to extract actinides from LWR spent fuel. And, in fact, it turns out that there is an extensive experience base at Argonne in applying pyrochemical processes to oxide fuel.

The earlier pyrochemical process development efforts were discontinued because there was no clear advantage over the traditional Purex process that produces a pure plutonium product stream. Today, however, the goal has changed. In traditional reprocessing based on Purex, the goal was to produce a highly decontaminated pure Pu product stream. However, when LWR processing is viewed as a waste management strategy, the goal is quite different. Neither a pure Pu product stream nor a high decontamination factor is required. In fact, just the opposite is desirable. The new process goals, when LWR spent fuel processing is viewed as a waste management strategy, are as follows:

- Direct extraction of all actinides (Pu, Np, Am, Cm, etc.) from the spent fuel as a single product stream.
- An actinide recovery target of 99.9%.
- The process should be incapable of producing pure Pu product.
- The process should be incapable of achieving a high decontamination factor for fission products.
- The process should be simple enough to achieve acceptable economics.

A preliminary assessment has been made to investigate the feasibility of using pyrochemical processes for directly extracting actinides from LWR spent fuel, satisfying the new process goals. It appears that pyrochemical processes are compatible with the new process goals and two promising flowsheet options have been identified: (1) a salt transport process and (2) a magnesium extraction process.

Pyrochemical processes appear to be a natural fit to the LWR actinide extraction application and should provide significant advantages over the traditional Purex-based processes. First, potentially all actinide elements are extracted in a single product stream, along with most rare earth fission products. A pure plutonium product is not possible. The product is highly radioactive and is not much more attractive than the original spent fuel as far as the diversion risk is concerned. The process as such therefore provides some nonproliferation protection.

Second, in these processes uranium remains as metal ingots with some noble fission product contamination. In this form the uranium can be easily stored for later recovery and use in the IFRs. The actinide extraction processes then deal with only 1 or 2% of the total heavy metal. This small mass flow and the few process steps involved lead to compact equipment systems and small facility size, and should portend favorable economics.

As a spin-off from the IFR technology, a small-scale R&D program has been initiated at Argonne to develop the necessary database to judge technical feasibility of the proposed flowsheets by the end of 1995.

FUTURE DIRECTIONS

The IFR Program is now at a critical juncture. Technical feasibility has been demonstrated and the technology database has been established to support its practicality. The Program is entering its demonstration phase. EBR-II is now completely fueled with IFR fuel alloys, and rapid progress is being made on the refurbishment of its Fuel Cycle Facility. When completed later this year, EBR-II and its Fuel Cycle Facility will then start operating as a pilot-plant scale IFR prototype.

This prototype demonstration, expected to be completed by the end of 1995, will include near-commercial-scale IFR fuel cycle processes, actinide recycle technology, and substantial progress in waste form certification. The IFR prototype fuel cycle demonstration is the crucial step in proving the practicality of the next-generation nuclear fuel cycle.

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