THE INTEGRAL FAST REACTOR AND
ITS ROLE IN A NEW GENERATION OF NUCLEAR POWER PLANTS*

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MEDIUM SIZE POWER REACTORS IN THE
NEXT GENERATION

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TOKAI UNIVERSITY
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BY

R.R. SMITH
ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois

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ABSTRACT

In the aftermath of two major nuclear powerplant accidents in seven years (TMI-2, March, 1979 and Chernobyl, April, 1986) efforts are being focused on the development of plants that are inherently safe. Of various concepts under consideration sodium-cooled breeder reactors are of particular interest since these are not only endowed with features conducive to inherent safety but are also environmentally acceptable and have the ability to transform an essentially worthless material, U-238, into valuable nuclear fuel, i.e. Pu-239. Additional virtues are associated with the use of sodium as a coolant since sodium is chemically passive with respect to cladding and structural materials and is characterized by a boiling point so high (1650°F) that all primary components operate at room pressure.

These virtues would by themselves qualify the breeder as a highly attractive candidate for development as an inherently safe plant. But an additional dimension is being added. Recent reconsiderations of earlier technologies based on the use of metallic fuels in breeder reactors are leading to the development of an attractive advanced concept, the IFR (Integral Fast Reactor), that has all of the virtues of conventional breeders plus those that make an inherently safe system even safer, closes the fuel cycle in on-site (colocated) reprocessing facilities, virtually eliminates the threat of fuel diversion and promises lower costs through the modification or elimination of overly complicated emergency shutdown systems. In its simplest form the IFR is little more than a sodium-cooled pool-type metal-fueled breeder that is patterned closely after EBR-II, a small 62.5 MWt breeder reactor located near Idaho Falls, Idaho.*

Of the IFR's many virtues inherent plant safety is of particular interest since IFRs are endowed with the remarkable ability to protect themselves under extreme undercooling conditions. A convincing demonstration of this ability was carried out at EBR-II on April 3, 1986. With the reactor operating at full power and with all flow-related automatic shutdown systems intentionally disabled all cooling systems were turned off. In effect the

*EBR-II is operated by Argonne National Laboratory, Argonne, Illinois under a contract with the USDOE Chicago Operations Office and the University of Chicago.
reactor couldn't be cooled and it couldn't be shut down. But nothing happened. The plant quietly and uneventfully shut itself off and it did so without the assistance of any electrical, mechanical or operator action. Within a matter of minutes power generation ceased and temperatures returned to normal.

The results of the tests are destined to revolutionize concepts of nuclear plant safety. At the present time the safe operation of all nuclear plants relies on the infallible response of manmade safety systems. Although these are highly reliable the accidents at Three Mile Island and Chernobyl are telling us that neither man nor his carefully engineered safety systems are truly infallible. In the Idaho tests, however, the fallibility of man was never a factor. In all of the tests the mechanisms responsible for shutdown action were those that self-initiated in accordance with principles and laws of nature such as gravity, heat conductivity, thermal expansion, etc. Nothing in the form of manmade safety systems or human intervention was needed to protect the plant under conditions that could lead to the devastation of conventional nuclear plants.

Another important feature of the IFR consists of the inaccessibility of its fuel material. All fuel reprocessing activities are carried out remotely behind massive shielding in windowless and doorless cells. Since the fuel is never entirely purified of fission products the reprocessed material is much too radioactive for physical access even if a way could be found to penetrate the cells.

As promising as the IFR concept appears to be there may be spinoff technologies of equal importance. In the course of research and development concerned with the IFR fuel cycle Argonne engineers may be opening the door to new technologies that may prove useful in solving some of the problems concerned with the disposition of fuel discharged from LWRs (light water reactors). Simply put, the IFR fuel cycle is based on two major procedures: The electrochemical separation of metallic uranium and plutonium
from fission products and the pyrochemical enrichment of plutonium from IFR blanket material. In principle, at least, these procedures may be modified for electrolytically reducing the uranium and plutonium content of LWR oxidic fuel material to the metallic state. The resulting metallic material could then enter the IFR fuel stream at the same stage entered by discharged IFR blanket material.

Equally interesting is the possibility of using IFR fuel cycle technologies to separate long-lived alpha-emitting actinides from spent fuel material (both IFR and LWR). The separated actinides in metallic form would then be incorporated into recycled IFR fuel elements where they would fission (incinerate) under the fast neutron environment of IFRs. If this can be done, and thermodynamic data and the results of bench scale tests indicate that it can, waste storage procedures could be significantly simplified.

These and other matters along with an update on the status of the IFR will be addressed.
THE INTEGRAL FAST REACTOR (IFR)
AND ITS ROLE IN A NEW GENERATION OF NUCLEAR POWER PLANTS

INTRODUCTION

The nuclear option as it exists in the United States today is in trouble. In 1975, the high point of nuclear enthusiasm, 236 plants were either operating, under construction, on order or in various stages of planning (Ref 1). Today, mid 1986, the number of committed plants has slipped to 130. Of these 101 are either operating or in startup testing; 27 are under construction; and two are still in various planning stages. Eight years have passed since the last plant was ordered and, during this time, orders or plans for 75 plants have either been cancelled or indefinitely deferred. Clearly a de facto moratorium on new plant construction has descended on the nuclear industry.

Of the 101 plants either in operation or in startup testing several are beginning to age. If the average useful lifetime of these plants is limited to 35 years and if the de facto moratorium on new plant construction becomes permanent the nuclear component in the American energy mix will peak in the early 1990s and will, from that time on, inexorably decrease. As the nuclear component dwindles to zero and as oil and natural gas are phased out as power plant fuels the American generating capacity will come to rely almost exclusively on the burning of coal, a practice considered by many as an environmental threat. To avoid an uncomfortable reliance on a single fuel, i.e., coal, the nuclear option must prevail. But will the nuclear option prevail? Hopefully, yes, but standing squarely in the way of a meaningful resurgence of interest in nuclear power is public opinion. Scarcely a decade ago a substantial majority of the American public endorsed the use of nuclear power. Since then public sentiment has shifted from outright approval to suspicion and more recently to opposition and even open hostility. The extent of the public's disenchantment with nuclear power is apparent in the results of a Washington Post – ABC public opinion poll (Ref 2). Of over 1500 respondents 78% opposed the construction of additional plants and over 40% were in favor of phasing out those that are currently operating.
In a practical sense the future of the nuclear option in the United States rests with a public that, rightly or wrongly, perceives nuclear power as dangerous, environmentally threatening, unduly expensive and as a pathway to nuclear terrorism. Reversing convictions such as these will not be easy but one possible move in this direction may be the development of a new generation of plants that address these concerns. Such plants must be inherently safe, environmentally clean, virtually immune to the malevolent actions of unfriendly forces and economically competitive with existing energy options. Implicit in this approach to revitalizing the nuclear option is the belief that if such plants can be developed and convincingly demonstrated the pendulum of public opinion may return to a more moderate position.

Prominent among those who are advocating innovative reactor concepts is Alvin M. Weinberg, Director of the Institute for Energy Analysis at Oak Ridge, Tennessee (Ref 3). In a guest editorial in Science - 1982, Weinberg called for "reinventing nuclear power." He went on to say, "I therefore propose that we use the current pause in deploying new nuclear plants to launch a second nuclear era that would rely on a new nuclear power system that would be more acceptable to the public and a better investment for the utilities...I believe we may need wholly new designs for reactors, designs that eliminate or minimize the vulnerabilities of today's reactors."

In a follow on article in Science, Weinberg discussed the progress that is being made in designing reactors that are inherently safe and "whose safety depends not on the intervention of humans or on electromechanical devices but instead depends on immutable and well-understood laws of physics and chemistry" (Ref 4). Singled out for particular attention in this regard were two concepts: the Process Inherent Ultimately Safe (PIUS) reactor being developed in Sweden by ASEA/Atom (Ref 5) and the modular High Temperature Gas (HTG) reactor proposed by General Atomics (Ref 6) in the USA and by Interatom (Ref 7) a subsidiary of KWU in Germany. Weinberg concluded that inherently safe reactors are "surprisingly reasonable engineering devices" and he asks, "Are there other equally clever schemes for inherently safe reactors and can similar principles or better ones be adopted to the design of breeder reactors?"
The answer to these important questions is an emphatic "yes". Recent reconsiderations of earlier technologies based on the use of metal fuels in EBR-II* are leading to the development of an attractive advanced reactor concept, the IFR (Integral Fast Reactor), a sodium-cooled, pool-type metal-fueled breeder that operates in concert with colocated fuel reprocessing and refabrication facilities.

As discussed in the text that follows the IFR has all of the virtues of conventional breeders plus those that make an inherently safe system (the breeder) even safer, closes the fuel cycle in colocated (contiguous) reprocessing facilities, virtually eliminates the threat of fuel diversion and promises cost-savings through simplifications in plant design.

The ability of the IFR to breed when breeding becomes necessary is an additional virtue. Of all proven or technologically feasible energy systems only breeder reactors can promise energy in essentially inexhaustible amounts. All other established options, i.e., coal, gas, oil, hydro and nuclear-steam (from conventional nuclear plants) are resource limited. No other energy option responds as well to the growing practice of electrification and the national goals of energy conservation since the breeder has the unusual ability of transforming essentially inexhaustible amounts of an otherwise worthless material, i.e. U-238, into electricity which can, in turn, be used to avoid the burning of equally prodigious amounts of coal, oil and natural gas.

If the breeder's virtues were limited to these it would still be a leading contender for development and deployment in a "Second Nuclear Era". But in addition to the virtues of inherent safety, a diversion-resistant fuel cycle and an unlimited fuel supply the IFR is simple to operate, maintain and inspect; it poses less of a threat to the environment and public health and

* EBR-II is operated by Argonne National Laboratory, Argonne, Illinois, for the USDOE under a contract with the USDOE Chicago Operations Office and the University of Chicago.
safety than all other major energy options; and it is well down the path of
development. No technological breakthrough is needed. It would be ironic,
indeed, if breeders often maligned as unsafe and unneeded prevail as a means
of restoring credibility to the nuclear image and supplying an energy hungry
world with essentially inexhaustible amounts of affordable electrical power.
2.0 THE INTEGRAL FAST REACTOR (IFR)

In simple terms the IFR is nothing more than a conventional sodium-cooled pool-type breeder reactor that is fueled with a metallic alloy and is operated in concert with colocated fuel reprocessing and refabrication facilities. The concepts of metallic fuel material and colocated fuel cycle facilities are really nothing new. EBR-II has been operating successfully for over 22 years with a metallic fuel alloy and was actually operated as an IFR during the period 1964-1969 when more than 40,000 spent fuel elements (five full-core loadings) were processed and refabricated in contiguous fuel cycle facilities.

For historical reasons and to serve as a basis for subsequent discussions brief descriptions of EBR-II and its colocated fuel cycle facilities are given in Appendices A and B, respectfully. Aside from some obvious differences, size, for example, any future IFR complex will closely resemble EBR-II and its earlier on-site fuel processing and refabrication facilities.

A typical IFR complex consists of four principal facilities: a reactor, a spent fuel reprocessing plant, fuel element refabrication facilities and radioactive waste handling and storage facilities. The reactor portion of the complex is nothing more than a conventional pool-type sodium-cooled system that differs little in concept from similar plants either in operation or under development in other industrialized nations. The IFR does differ, however, from sibling systems in two principal respects. Whereas all other similar systems are fueled with a mixture of uranium and plutonium oxides the IFR is fueled with a metallic alloy that has the nominal composition U, 70%; Pu, 15%; and Zr, 15%. The second feature that distinguishes the IFR from sibling systems is the use of electrochemical and pyrochemical techniques for reprocessing discharged fuel into rechargeable elements.
2.1 Reprocessing Fuel and Blanket Materials

The procedures being developed for the processing of metallic fuel and blanket alloys address two principal needs, viz, the substantial removal (99% or greater) of fission products from discharged fuel material and increasing the concentration of plutonium in blanket material so that the bred plutonium can be used to compensate for burnup losses in the fuel. The first of these needs, the removal of fission products from the fuel, is carried out electrochemically in a special electrolytic cell. Increasing the concentration of plutonium in discharged blanket material is achieved in a two phase molten metal-molten salt extraction process known as "halide slagging".

2.1.1 Electrochemical Refining

The electrochemical process for separating uranium and plutonium from fission products is carried out in a low voltage (approximately one volt d.c.) electrolytic cell. Although spent fuel and discharged blanket materials are treated separately the head-end procedures for both are essentially the same. Elements are mechanically chopped into small pieces (including the cladding) and mixed with molten cadmium in a low carbon steel pot that serves as the anode (positive terminal) of an electrolytic cell. Floating on the surface of the cadmium is a mixture of various molten chloride salts (principally lithium, barium and calcium chlorides). Immersed in the molten salt layer is a metallic molybdenum rod or plate that serves as a cathode. Alternative forms of cathodes are under consideration.

Although the solubilities of uranium and plutonium in the cadmium anode are low what does dissolve diffuses throughout the cadmium layer and makes physical contact with the molten salt. When the cell is energized dissolved uranium and plutonium at the cadmium-salt interface are anodically oxidized to the respective cations (positive ions) which migrate under the electric field through the salt phase to the cathode where they are electrochemically reduced back to the metallic form. Deposits on the cathode are periodically removed by scraping and sent to neighboring in-cell stations for further treatment and eventual refabrication into recyclable elements.
2.1.2. **Halide Slagging**

Blanket material (a metallic U-Zr alloy) is dicharged when the plutonium content reaches approximately 5 weight-percent. This material, too, is electrochemically refined and the plutonium content is used to replace that burned in the core. Because fuel material has a much higher plutonium content than that of the blanket material (15% versus approximately 5%) some means of plutonium enrichment is needed. The process under development is based on a two-phase molten metal-molten salt extraction process where the metallic phase consists of molten blanket material and the salt phase consists of molten calcium and barium dichlorides. Specific amounts of uranium trichloride are added to the salt to serve as an oxidizing agent for converting metallic plutonium to the trichloride form.

The two layers are intimately mixed by continuous agitation. Because of thermodynamic considerations plutonium is preferentially oxidized to the trichloride form which accordingly remains with the salt phase. According to the results of preliminary thermodynamic analyses the plutonium-uranium ratio of 0.05 in the discharged blanket material is increased to 1.7 in the salt fraction. Since the plutonium-uranium ratio of the fuel alloy is approximately 0.20 the plutonium content of the salt mixture is entirely adequate for upgrading the plutonium content of the fuel. This is done by adding plutonium-rich salt from the halide slagging process to the salt mixture of the electrolytic cell.

2.2 **Refabrication of Fuel Elements**

The product of the electrochemical reduction of fuel material is a metallic mass that has been upgraded with respect to plutonium through the addition of plutonium-rich salt from halide slagging operations. At this stage the metallic mass contains substantial quantities of occluded halide salts. Separating the occluded salts from the fuel material is straightforward. The mass is melted and the salts, being less dense, float to the surface. The metallic fraction is simply drained from the crucible.
The first step in pin-casting operations consists of remelting the purified fuel material in a graphite crucible in an induction-heated vacuum furnace. Located in the furnace in a vertical attitude is a cluster of approximately 100 precision Vycor (quartz) molds. After the charge is melted the crucible is raised so that the lower ends of the molds are immersed below the surface of the melt. The furnace is then rapidly pressurized (to approximately two atmospheres) to drive the melt upward into the evacuated molds. Within a matter of a few seconds the melt freezes and the crucible is lowered. Following a carefully programmed cooling period the furnace is opened and the molds removed for subsequent crushing. At this point the pins are ready for refabrication into rechargeable elements.

Vacuum injection casting technologies are well established. As discussed in Appendix B such techniques were used to fabricate elements for the initial loading of EBR-II from radioactively cold fuel material. After plant startup (August, 1964) radioactive spent fuel was processed and refabricated into rechargeable elements for approximately five years. Since that time the vacuum injection casting method has been used to fabricate EBR-II fuel elements from "cold" fuel material. More recently hundreds of U-Pu-Zr IFR-type fuel elements needed for in-core irradiation studies have been fabricated in essentially the same way.

2.3 Waste Handling

There are four principal classes of radwastes. These are: fission product gases, high level wastes, intermediate level wastes and liquid wastes.

2.3.1 Fission Product Gases

These, primarily xenon and kryton, are released directly to the argon atmosphere during fuel disassembly and chopping operations. A small continuous purge of the argon atmosphere is continuously passed to cryogenic traps in which the rare gases are liquified and transferred to pressure-resistant cylinders which, in turn, are transferred for interim storage in a waste storage building.
Most of the tritium produced from ternary fission diffuses through the cladding into the primary coolant and is eventually removed in special cold traps. The remaining tritium is released during fuel disassembly and chopping operation directly to the cell atmosphere and is eventually recovered in the form of tritiated water vapor by molecular sieve dryers.

2.3.2 High Level Wastes (HLW)

These consist of fuel element scrap (plenums and cladding hulls) anode baskets, cadmium wastes, salt wastes, fume traps, crucibles, and broken molds. These are compacted in steel liners and placed in a stainless steel HLW container. Salt wastes in molten form are drained into the HLW container and allowed to freeze from the bottom up. When the HLW container is filled it is removed from the furnace, a cover is welded on and the filled container is transferred to the waste storage building.

Such waste contains essentially all of the fission products (except the rare gases), unrecovered actinides and TRU-contaminated scrap. The daily waste from a 1400 MWe complex of smaller modular units can be accommodated in a six-inch schedule 80 steel pipe 58 inches long. After a cooling period of seven years the sealed stainless steel HLW containers (pipes) are encapsulated in a thick layer of copper or lead alloy to isolate the wastes from the environment for a time span of at least 100,000 years. After final encapsulation the copper or lead-clad cylinders are ready for transport to a permanent HLW facility.

2.3.3 Intermediate Level Waste (ILW)

ILW wastes consist of blanket and fuel assembly hardware. This is loaded into a steel container which, in turn, is sealed and transferred to the waste storage building. Here the containers are reencapsulated for final storage in shallow land-fill facilities.
2.3.4 Liquid Wastes

HLW containers are water-washed in the shielding cask that is used to transfer the containers to the waste storage building. The wash water, accordingly, contains fission products, fuel material and actinide species. Waste water is processed via an evaporative process in the fuel cycle building. The resulting solids are then incorporated into the HLW stream.

All of the activities and procedures described above are subject to change as refinements are made to waste handling technologies. Many refinements are under consideration. One of these, actinide separation, is of particular interest. If, as described in Sect. 5.0, the actinides can be separated from HLW storage problems become much simpler since non-actinide HLWs do not require permanent storage.

2.4 Diversion-Resistant Features of the IFR Fuel Cycle

An important feature of the IFR concept consists of the inaccessibility of its fuel material. All fuel (and blanket) reprocessing and refabrication facilities are carried out remotely behind massive shielding in windowless and doorless cells. Since fuel material is never entirely free of fission products the reprocessed material is much too radioactive for physical access even if a way could be found to penetrate the cells. In this sense the fuel cycle may be considered highly diversion-resistant if not diversion-proof.

The possibility of diverting fissionable material is further reduced by a startup strategy that is based on fueling the initial loading with non-weapons grade enriched uranium. As the enriched uranium burns out plutonium grows in and after approximately 15 years essentially all of the original fuel will be gone and the plutonium content will have reached an equilibrium level that is easily adjustable to satisfy core needs. From this point on plutonium bred in the blankets serves as fuel material for the core. Since the generation rate of plutonium in the core and blankets can be "tuned" to burnup losses there is no need for on-site and off-site shipments of fuel and blanket material after the original shipments are made. This feature eliminates or at least considerably softens concerns that arise from over-the-road shipments of fuel material, e.g. accidents, spills, hijacking, crossing municipal and state boundaries, etc.
3.0 THE IFR AND INHERENT SAFETY

At the present time the safe operation of all nuclear plants depends on the infallible responses of man-made safety systems and plant operating personnel but, as the accidents at Three Mile Island and Chernobyl have clearly shown, neither man nor his engineered safety systems are truly infallible. Clearly needed are plants whose safety is guaranteed, not by the actions of man, but by immutable laws and principles of nature, e.g., gravity, heat conductivity, thermal expansion, etc.

Of particular interest in eliminating the fallibility of man as a factor in powerplant safety are plants that are inherently safe, i.e., those that are naturally endowed with features that self-actuate to protect the plant under all credible upset conditions. Such features must actuate in a fool-proof manner and they must function without the assistance of any electrical, mechanical or operator actions.

From the viewpoint of immunity to loss-of-coolant accidents EBR-II, the progenitor of the IFR, is an inherently safe plant. Proof of its inherent safety appears in the results of the tests summarized below.

3.1 Decay Heat Removal by Natural Circulation. The Rests of Tests in EBR-II.

Long a matter of concern in the design and operation of breeders is the need to remove and dissipate fission product decay heat after plant shutdown. Although details vary from one plant to another all breeders have been and still are equipped with active systems that maintain a limited flow of coolant through the core after plant shutdown. As the results of definitive shutdown heat removal tests in the United States, Scotland and Japan (Ref 8-11) such systems, pony motors for example, may no longer be needed. Without exception all of the tests have demonstrated the effectiveness of self-initiating natural circulatory flow as a means of core cooling in the absence of all forced coolant flow after plant shutdown. Such results strongly suggest that in the design of future breeders active shutdown heat removal systems may be eliminated.
Of the ten breeders operating in the world only one, EBR-II, has all of the features found in an IFR, viz, sodium-cooled, pool-type configuration and metal-fueled. Because of the similarities between IFRs and EBR-II the thermo-hydraulic behavior of an IFR immediately after a loss-of-flow-with-scram, but without active decay heat removal systems, may be reasonably inferred from the results of an intensive series of shutdown heat removal tests conducted on EBR-II in June, 1984 (Ref 12).

The most severe test of a series of nine involved a scram from full power and full flow with the simultaneous loss of electrical power to the main coolant pumps, the auxiliary pump and the secondary pump. As a result the only means of conveying heat from the core to the secondary system consisted of natural circulatory flow. A summary of the test results in graphic form has been given by Planchon et. al. (Ref 12) and, with appropriate permission, is repeated here as Fig 1. Actual measured data points, the circles, were those registered by a thermocouple located at the discharge side of a fueled instrumented subassembly.

As the data indicate a maximum coolant outlet temperature of 1150°F, an increase of 150°F, was measured approximately 70 seconds into the transient as the primary flow coasted to zero. From this point on the outlet temperature decreased as natural circulatory flow developed and as the effects of neutron power and decay heat decreased.

Beside demonstrating the easy transition into natural circulatory flow and the ability of natural circulation to keep core temperatures acceptably low the test also demonstrated the passive ability of the secondary sodium system to convey heat from the primary system to the steam generators via natural circulatory flow in the closed loop of the secondary sodium system. In all of the tests the ultimate heat sink was the generation of steam and its subsequent condensation in the turbine condenser.
3.2 Decay Heat Removal by Primary Tank Shutdown Cooling System

A case of hypothetical interest is one in which the following events happen more or less simultaneously: the two primary pumps trip, the reactor scrams and power to the auxiliary pump, secondary pump and feedwater pumps is lost. Under conditions such as these and in the absence of all means of heat transfer from the primary tank the temperature of the bulk sodium (in the primary tank) will increase monotonically. Two additional means of cooling, however, are available. These are the normal loss of heat from the primary tank to outlying structures and the transfer of heat from the bulk sodium to the atmosphere via an in-tank shutdown cooling system. These modes have, respectively, heat rejection capabilities of 135 and 270 KW.

The shutdown cooling system is remarkably simple. It consists, essentially, of two NaK-filled bayonet type convective loops which have hot legs in the primary tank and cold legs outside the reactor building. Both loops are continuously in operation. Neither requires an external power source. All flow proceeds through natural circulation. A system of louvers and dampers on the heat rejection side (outside the containment building) is used to regulate the rate of heat dissipation. Under normal operating conditions, with the dampers closed, 120 KW of heat are continuously transferred from the bulk sodium to the atmosphere. If, for some reason or other, the temperature of the bulk sodium should exceed 710°F, as it would if the secondary sodium system were unavailable, the dampers would automatically open and the heat transfer rate would increase to approximately 350 KW. Under normal operating conditions the louvers are kept closed by an air-actuated spring system. If full convective flow is needed, air from the pneumatic actuators is vented and the dampers open under spring action. A system of thermocouples in the primary tank supplies signals to the pneumatic actuators.

The effectiveness of the shutdown system to reject heat from the primary tank may be assessed for the following conditions: full fission product inventory in the core, reactor tripped from full operating power, and the secondary sodium system unavailable for heat rejection. Under such conditions the temperature of the primary sodium would increase and the dampers would
automatically open to increase the heat-dissipation rate to the atmosphere. Approximately seven hours after shutdown, the decay heating rate and the combined heat dissipation rate (130 KW for normal losses and 350 KW from the shutdown coolers) become equal. From this point on, if remedial action were not taken, the bulk sodium temperature would decrease.

With only one shutdown cooler operable (with a heat rejection capability of 175 KW) and taking credit for 130 KW of normal heat loss, the bulk sodium temperature would peak at 780°F, 36 hours after shutdown. With both shutdown coolers inoperable, and with 130 KW of normal heat loss the bulk sodium temperature would increase to a maximum of 1067°F, 9-10 days after shutdown. Not only would temperatures in this range be acceptable but the long heat-buildup period would permit effective remedial action, e.g., returning the secondary system to service, increasing the forced air supply to the shield cooling system, etc.

The simultaneous loss of the shutdown cooling and secondary cooling system is highly unlikely but even if such should happen the thermal capacitance of the primary sodium inventory would limit the consequences. Nothing in this unlikely series of events suggests thermal damage to components in the primary tank.

3.3 Loss of Cooling Capabilities Without Scram

A major milestone in the development of inherently safe nuclear plants was reached on April 3, 1986 at EBR-II when the results of a rigorous series of tests confirmed the ability of sodium-cooled, pool-type metal-fueled systems to protect themselves under extreme undercooling conditions. (Ref 13) The results of the tests clearly validated the concept of inherent plant safety and unambiguously established the sodium-cooled, pool-type, metal-fueled reactor as an outstanding example of an inherently safe plant.
Two tests took place. In both, all flow-related trip circuits were intentionally disabled. The first test took place during the morning of April 3, 1986. With the reactor operating at full power (nominally 60 MWt) power to all cooling systems (primary, secondary and auxiliary) were turned off while the reactor continued to run. In an equally meaningful test conducted during the afternoon of the same day power to the secondary pump was turned off while the reactor continued to run.

In neither case did anything serious happen. Core and coolant temperatures increased but, as they did, self-actuating temperature-sensitive countervailing effects took over and shut the reactor down. No electrical, mechanical or operator action was taken. Increases in core and coolant temperatures were the sole driving forces for shutdown action. More specific details of the tests follow below.

3.2.1 Loss-of-Flow-Without-Scram (LOFWS)

Conditions immediately prior to the tests were: normal full power (60.0 MWt); normal primary flow (9218 gpm); normal secondary flow (5750 gpm); reactor inlet temperature, 650°F (reduced from a normal value of 700°F); reactor outlet temperature, 828°F (50°F lower than normal); flow and temperature related trips, intentionally disabled; and operator action, none taken during the test.

In this test power to the auxiliary and secondary pumps was turned off and the primary pumps were tripped. Since the flow and temperature-sensitive automatic shutdown system had been rendered inoperative the reactor continued to run. To a very large extent this particular test simulated a hypothetical situation in which all station power is lost, all emergency diesel power is lost and, for one reason or another, the reactor cannot be shut down. Such a sequence, although highly unlikely in the real world of reactor operations, has long been envisioned as one of the worst (if not the worst) accidents that could hypothetically happen to a sodium-cooled breeder.
As a result of the tests this worry no longer exists. As evident from the results summarized in Fig. 2 the reactor "rode" through the test with no evidence of any problem. Of particular interest is the behavior of reactor power as shown in Fig. 2. Whereas the pump rundown was over in approximately 100 seconds power continued to decrease over a period of roughly six minutes. This led to an adverse mismatch between power and primary coolant flow and an attendant increase in core and coolant outlet temperatures as flow continued to decrease faster than power. Approximately two minutes into the test the primary pumps stopped. Since power continued to decrease after the pumps had stopped the power-to-flow ratio began to decrease. This led to decreasing outlet temperatures. After approximately ten minutes into the test the coolant outlet temperature returned to its original (pre-test) value.

The information given in Fig. 2 demonstrates the importance of temperature-sensitive feedback effects in shutting the reactor down. As the coolant flow coasted down core coolant temperatures increased and, as they did, negative reactivity was fed back to the system. Such action led to a power decrease which, in turn, caused a decrease in the core outlet temperature. In effect, two countervailing phenomena were active: one that tried to drive the outlet temperature up (the results of a temporary power-to-flow mismatch during the flow coastdown), and one that tried to drive the outlet temperature down (the results of the strong decrease in reactor power). The contest between the two competing physical processes ended when the heat generated in the core triggered enough feedback reactivity to drive the power nearly to zero. Again, increasing core and coolant temperatures were the only driving forces needed to trigger the reactor shutdown.

3.2.2 Loss-of-Heat Sink - Without Scram (LOHSWS)

Conditions immediately prior to the second test, i.e., the LOHSWS, were essentially the same as those for the first. The test was initiated by interrupting power to the single a.c. electromagnetic secondary pump. The two primary pumps and single auxiliary pump were left running. As evident from Fig 3, the secondary flow rate decreased within a matter of seconds to a
value less than 0.5% of normal flow. During this time the reactor continued to run and, since the normal heat transfer path from primary to secondary system no longer existed, heat was directly dumped to the primary sodium coolant. This led to an increase in the temperature of the inlet coolant and this action, in turn, initiated temperature-sensitive negative feedback effects that triggered a continuous reduction of power.

Since full primary flow continued throughout the test power decreases were sensed as decreases in the temperature of the outlet coolant. The three principal effects, i.e., an increasing inlet temperature, decreasing reactor power and decreasing outlet temperature are evident in the data of Fig. 3.

Approximately 18 minutes after time zero the power had dropped essentially to zero and the inlet and outlet temperature of the core were approaching the same asymptotic value of 725°F. Not only did the reactor shut itself off without the assistance of man-made shutdown systems or operator action but the outlet temperatures actually decreased by approximately 100°F. The information given in Fig. 3 again confirms the existence of inherent plant safety and established the pool-type sodium-cooled metal-fueled breeder (such as EBR-II) as an inherently safe nuclear plant.

3.3 The Quenching Temperature

The remarkable behavior of the reactor outlet temperature as illustrated in Fig. 3 may be understood in terms of two fundamental and easily measurable reactor parameters, viz, the PRD (power reactivity decrement) and the ITC (isothermal temperature coefficient. In all breeder reactors the net power coefficient is negative. Reactivity must be added to take the reactor from hot critical to operating power. This amount of reactivity is, by definition, the PRD.

The ITC is also negative; it is defined as the amount of reactivity lost from a uniform heatup of one degree in the core, blanket and structure. Under loss-of-heat sink condition such as those described above the primary or bulk sodium temperature must increase and as it does reactivity is taken from the
system. It follows, then, that when the reactivity taken from the system through an increase in the bulk sodium temperature is equal to the PRD the reactor power level must go to zero.

The temperature at which the negative reactivity fed back by bulk coolant temperature increases equals the PRD is defined as the quenching temperature, $T_q$. Any increase in the bulk sodium temperature beyond this point will cause the reactor to go subcritical. A mathematical statement of this fact is given by the following simple expression:

$$\text{Quenching temperature} = T_q = T_b + \text{PRD/ITC}$$  \hspace{1cm} (1)

where $T_b$ is the bulk coolant temperature.

The expression for $T_q$ is not an approximation; it is mathematically exact. In the case of EBR-II the PRD and ITC are known to be $30\xi$ and $0.4\xi/°F$, respectively. From the expression for $T_q$ given above the quenching temperature under LOHSWS conditions is:

$$T_q = 650°F + 30\xi/0.4\xi/°F$$  \hspace{1cm} (2)

As evident from an inspection of the data of Fig. 3 the calculated value of $725°F$ is in exact agreement (perhaps fortuitously) with the asymptotic value of $725°F$ measured for the inlet and outlet coolant temperatures during the LOHSWS test.

3.3.1 The Importance of the Grid Plate Under LOHSWS Conditions

As useful as Eq.1 is in describing the self-protective nature of LMRs it fails to address the concern that the quenching temperature could be so high that in-tank components could be thermally damaged under unprotected loss-of-flow conditions. Clearly, the margin of safety is enhanced by keeping the quenching temperature low. How high the quenching temperature of a given breeder will be depends strongly on the amplitude and sign of the various components that affect the PRD and the ITC.
All of the feedback components that affect the PRD are sensitive, in one way or another, to power and temperature. All of the feedback components that affect the ITC, on the other hand, are sensitive to bulk coolant temperature only. Of particular importance in this distinction is the grid plate. Under normal operating conditions, the grid plate is continuously bathed by the bulk coolant which remains at a constant temperature. Power increases, accordingly, have no effect whatsoever on the temperature of the grid plate. As a result reactivity feedback effects from grid plate expansion are completely missing in the definition of the PRD. The reactivity effects of grid plate expansion are, however, a part of the ITC since changes in the bulk coolant temperature will be reflected by grid plate expansion and an attendant decrease in reactivity balance. This means that grid plate expansion effects under loss of heat sink conditions have the effect of decreasing the quenching temperature. This is evident from an inspection of Eq.1 since the term PRD/ITC is made smaller through the inclusion of grid plate expansion effects in the ITC. The role that grid expansion effects play in keeping the quenching temperature acceptably low is extremely important. Without grid plate expansion effects, the quenching temperature could be so high that thermal damage could result from unprotected loss-of-flow sequences.

3.3.2. The Effects of Fuel Type on the Quenching Temperature

One of the reasons why EBR-II (and by inference the IFR) is relatively immune to loss-of-heat-sink situations may be inferred from Eq.1. Included in the PRD is the reactivity vested in the Doppler temperature effect. Although a large Doppler temperature coefficient of reactivity acts on the side of safety during plant startups and under inadvertent reactivity insertion conditions, it acts detrimentally during temperature-induced shutdowns.*

* A temperature-induced shutdown is one in which the negative reactivity needed to cancel the reactivity added during startup (the PRD) comes from increases in the temperature of the bulk coolant.
During a temperature-induced shutdown essentially all of the reactivity vested in the Doppler effect will return as positive. In order to cancel the positive reactivity the bulk coolant temperature must be driven high enough to provide an offsetting amount of negative reactivity from other feedback processes such as grid plate expansion, structural expansion, etc. Depending on the amount of reactivity vested in the Doppler effect the quenching temperature may be so high that thermal damage could be a problem.

The amount of reactivity vested in the Doppler effect depends on amplitude (strength) of the Doppler temperature coefficient of reactivity and the temperature of the fuel. Because of their low thermal conductivities mixed oxide fuels must operate at very high temperatures in order to drive heat from the fuel pins into the sodium coolant. As a result the temperature at the center of mixed oxide elements may run as high as 5000°F. A substantial amount of reactivity must, accordingly, be added to overcome the large negative Doppler effect in going to power. If the plant undergoes a temperature-induced shutdown, most of the Doppler-vested reactivity will reappear as positive and will, accordingly, work against the negative feedbacks that are trying to shut the plant down.

The behavior of metallic fuel under similar circumstances is far more conducive to lower quenching temperatures. In the first place the magnitude of the Doppler temperature coefficient in metallic fuel loadings is smaller than it is in mixed oxide loadings (a factor of roughly two). In the second place the heat conductivities of metal fuels are very much higher than those of a mixed oxide fuels. The space-average temperatures of metallic fuels are accordingly lower than those of mixed oxides. The cooldown of metallic fuel from its operating temperature to the quenching temperature during a temperature-induced shutdown would be much less than in the case of a similar cooldown of a mixed oxide fuel. The amount of reactivity added during the cooldown of metallic fuel would, of course, be smaller than for a mixed oxide fuel and the quenching temperature of the metallic loading would be lower.
4.0 OTHER VIRTUES OF SODIUM-COOLED BREEDERS

The breeder, even as it currently exists, is remarkably well endowed with features that guarantee plant safety under all credible undercooling conditions. Simplifying the development of the breeder as an inherently safe plant is an established technological base that is being additionally refined in every major industrial nation. On a worldwide basis eight breeders have already been built, tested, shut down and decommissioned; ten are currently operational; five are under construction and plans are in progress for the construction of an additional two.

Although the breeder is very likely the most promising energy option it is, unfortunately, the most poorly understood. Primarily responsible for this unfortunate situation is the stereotypical portrayal of the breeder as a plant whose only virtue is its ability to generate more fuel than it consumes. As attractive as this virtue may eventually become there are other less heralded virtues that make the breeder, in general, and the IFR, in particular, highly attractive for development, not necessarily as a breeder, but as a power plant that responds to the concerns of the utilities, the regulatory agencies and, most of all, the general public. A brief summary of the many virtues associated with pool-type, sodium-cooled metal-fueled plants (patterned after the IFR concept) is given below.

4.1 Inherent Safety See Section 3.0.

4.2 On-Site Fuel Reprocessing See App. B.
4.3 Unlimited Fuel Supply

Breeders effectively "burn" uranium-238, a relatively abundant material that is essentially worthless for all other purposes. The energy content of our uranium reserves when used in breeders is equivalent to the energy available from 30 trillion barrels of oil, an amount 75 times greater than the oil reserves of the entire middle east. Put in another perspective the energy available from our proven uranium reserves could satisfy all of our electrical needs at today's consumption rates for 10,000 years. Without breeders, however, our uranium-based energy reserves will decrease by a factor of at least 100.

4.4 Conservation and Electrification

An important factor in our future energy mix is the direct use of electrical energy to do work that would otherwise be done by the burning of fossil fuels (coal, oil and gas). Electrification is the only practical long-term option that offers a permanent solution to our energy problems. But electrification will work if and only if electricity is generated from either renewable or inexpensive inexhaustible resources. Whenever renewable energy sources (solar, wind, tide, etc) are used to generate electricity which is used in place of fossil fuels, our supplies of these valuable materials are effectively stretched. Essentially the same logic applies whenever waste products or worthless materials (straw, corn stalks, uranium-238) etc. are used to generate electricity. Biological wastes, unfortunately, are not available in unlimited quantities. Uranium-238 essentially is. Its use in breeder reactors can generate prodigious quantities of electricity that can displace the need for equally prodigious quantities of valuable fossil fuel. Development of the breeder as a machine that can convert inexhaustible amounts of a worthless material into electricity is clearly consistent with the loftiest goals of electrification and energy conservation.
4.5 Environmental Impact

Of all practical power-generating options (including coal and commercial reactors) breeders have the smallest impact on public health, public safety and the environment. Although coal-burning has been, is, and will continue to be the "heavyweight" in our generating mix it does suffer from significant environmental disadvantages. Principal impacts on public health and safety are the environmental results of mining, crushing, washing, transporting and burning of the coal. The effects of these activities are reflected by accidental death, occupational diseases and the causal relationship between powerplant emissions and pulmonary and cardiovascular diseases. Other negative effects associated with the coal burning cycle include the impact of strip mining on fragile ecosystems and the possible involvement with the production of acid-rain and the threat of climatic changes caused by the so-called greenhouse effect.

In contrast, the fissioning of uranium suffers from none of these hindrances. The nuclear fuel cycle does lead to releases of radioactive materials to the environment, mainly from mining and milling activities, fuel manufacturing, plant operation and maintenance, fuel processing and the handling and storage of nuclear wastes. Of these, mining and milling activities contribute disproportionately to the radiation-induced cancer-related death rate through the release of radiobiologically dangerous alpha-emitting decay products of uranium and thorium daughter species. Although not widely appreciated the same dangerous alpha-emitting species are also emitted during the mining and burning of coal. In fact the results of studies by the Environmental Protection Agency led to the conclusion that modern coal-fired plants lead to public radiation exposures at least equal to those of comparably sized conventional nuclear plants. (Ref. 14)
4.6 Longer Plant Lifetime

Breeders are essentially immune to the effects of radiation-embrittlement and radioactive corrosion product buildup, phenomena that limit the effective lifetimes of commercial nuclear plants. There is no technological reason why breeders cannot be operated for 75 years or more and then shut down, not for decommissioning, but refurbishment.

4.7 High Thermal Efficiency

The thermal efficiency of an IFR (and other sodium-cooled breeders) is approximately the same as that of a fossil-fueled plant and approximately 20% higher than the thermal efficiency of a LWR (40% versus 33%).

4.8 Sodium as a Coolant

Molten sodium is a remarkably effective coolant. It has excellent heat transfer properties and its high boiling point (1650°F) eliminates the need for high pressure boundaries (pressure vessels, pipes, pumps, etc). Depressurization phenomena originating from leaks, sticky relief valves, etc. in the primary system are impossible. In the TMI-2 accident and very likely at Chernobyl depressurization was a key factor in fuel melting. Molten sodium is essentially non-corrosive in breeder systems. After 20 years of operation chalk marks on the sodium side of an EBR II steam generator were still clearly visible. (Ref. 15)
4.9 **Favorable Iodine Source Term**

Any iodine released from damaged fuel elements in an IFR will react immediately with the sodium coolant and be retained in the primary system as a non-volatile inorganic iodide.

4.10 **Low Occupational Radiation Exposure**

Of the three principal types of nuclear plants (boiling water reactors, pressurized water reactors and breeders) the breeder is by far the best plant from the viewpoint of occupational radiation exposure. Averaging over the past ten years of operation the annual radiation exposure rate at BWRs was approximately 800 Man-Rem and at PWRs, approximately, 450 Man-Rem. (Ref. 16) At EBR-II, however, the annual radiation exposure was only 19.0 Man-Rem (Ref. 17) and at Phenix (France) the dose rate was even lower, approximately 7.0 Man-Rem. (Ref. 18)

4.11 **Pool-Type Configuration** See information summarized for EBR-II, App. A.

4.12 **Unavailability of Hydrogen**

The generation of hydrogen through coolant-cladding (water-zirconium) and coolant-moderator (water-graphite) reactions is impossible in breeders. Water, the source of hydrogen in the TMI-2 accident and a contributing source in the Chernobyl accident is completely absent. Hydrogen released in the Chernobyl accident originated, very likely in part, from the reaction of water with graphite. The absence of water, zirconium and graphite in a breeder eliminates concerns over hydrogen explosions.
4.13 Simplicity in Maintenance

Breeders are simple to maintain. All major in-tank components are removable, repairable and replaceable. Primary components are much less radioactive than their commercial reactor counterparts. Radiation doses to maintenance personnel are, accordingly, much smaller (See Section 4.10). Furthermore, unlike the situation for commercial nuclear plants the containment building of a breeder is completely accessible during reactor operation.

4.14 Economics

Although far from clear the economic picture is encouraging. Increasing the safety margin through the use of metallic fuels should logically lead to the elimination or simplification of unduly complicated safety systems. Specifications and standards for steam-generation systems, building structures, components, etc. can likely be relaxed at considerable cost savings. Reprocessing the fuel in colocated facilities and handling and storing radioactive wastes in life-of-plant on-site facilities should lower fuel cycle costs.

Other cost saving strategies applicable to the IFR include the following.

- Standardization — choosing an acceptable design and staying with it.

- Modularization — factory fabrication of standard models under conditions of high quality control.

- Clustering — grouping several standardized modules at a single site and around a common fuel reprocessing facility.

- Licensing — simplifying licensing procedures by using the completed plant as a test-bed for demonstrating plant worthiness.
Even if breeders are never commissioned to breed other virtues, such as those described above, can justify the development of advanced breeder systems like the IFR as a highly attractive power-generating system that could, if necessary, supplement or supplant the role that is currently filled by existing commercial plants.
5.0 SOLVING LWR FUEL CYCLE PROBLEMS WITH IFR TECHNOLOGIES

One of the most serious institutional and technological problems facing the domestic nuclear industry is concerned with the temporary, interim and permanent storage of high level radioactive wastes. At the present time the disposition of spent LWR fuel is based on a throwaway strategy. Discharged fuel is stored on site in water-filled pools that are rapidly becoming overloaded. Plans for releasing the pressure on on-site storage facilities call for shipping the unprocessed fuel to centrally located facilities that provide monitored retrievable storage (referred to as MRS). Such facilities are little more than "attics" on the way to the dump. Spent fuel stored at such facilities must eventually be processed and packaged into a suitable form for permanent storage in geologically stable formations.

Earlier thoughts on the disposition of spent fuel focused on the separation of residual U-235 and plutonium from the fuel in commercial reprocessing facilities. The U-235 component, presumably, would be returned to uranium enrichment plants for upgrading into commercial LWR fuel. The plutonium fraction, as the thinking went, could be used as fuel material in a strategy known as "plutonium recycle".

But several things happened. Some were socio-political and some were economic. Concern over the separation of plutonium from spent fuel and its use for illicit purposes led to a ban on commercial reprocessing activities during the final days of the Ford administration. This policy continued throughout the Carter years and was eventually changed by President Reagan in his Nuclear Policy Statement of October 8, 1981.

During the past ten years the economics of fuel reprocessing have changed. Earlier predictions of a greatly accelerated electrical demand never materialized and as nuclear power plant orders slowed and eventually stopped
the demand for uranium decreased, all this at a time when substantial additions were being made to domestic uranium reserves. Improvements in fuel utilization also had an effect. By running the fuel to higher burnups less U-235 is being left in the discharged fuel. At the same time longer fuel residence times lead to an increase in the non-fissionable Pu-240 component relative to the fissionable Pu-239. Both effects lessen the incentives for separating and salvaging for recycle the fissionable species, U-235 and Pu-239.

The poor economics of fuel processing are particularly well illustrated by some statistical information. At the end of 1985 there were 13,000 metric tons of discharged fuel located in on-site storage pools. (Ref.19) Of this approximately 0.85% or 110 metric tons is U-235. Since the U-235 component of natural uranium is 0.72% there is little or no incentive to process discharged fuel for its U-235 content, particularly in view of increasing inventories of natural uranium and falling prices.

Salvaging the plutonium content of discharged fuel for recycle is even less economically attractive. Of the 13,000 metric tons of discharged fuel only 0.54% or 70 metric tons is fissionable plutonium but this is made less valuable by a heavy contamination of non-fissionable Pu-240.

5.1 Permanent Storage of High Level Radioactive Wastes: The Actinide Problem

Essentially all economic incentives for reprocessing spent fuel have vanished and it is unlikely conditions will change at least for the foreseeable future. To reduce pressure on on-site storage facilities there is only one practical solution, viz, shipping spent fuel to MRS facilities for interim storage and eventual processing and packaging into permanent storage form.

Complicating the processing and storage of commercial nuclear wastes is the presence of long-lived alpha-emitting radioactive species such as Pu-239,
Pu-240, Am$^{241}$, various Cm species, etc. Such species are often referred to as actinide elements or TRU which is an acronym for transuranium elements. Since the half-lives of many actinide species are long (the half-life of Pu-239, for example, is 24,000 years) and since alpha-emitting radioactive species (like the actinides) are radiobiologically threatening permanent entombment is considered the only practical solution.

By far the bulk of the radioactivity of spent fuel is associated with the beta-particle gamma-ray emitting fission products. It is these and not the actinides that determine the need for radiation shielding. But fission products are, in the long run, much easier to store and the reason for this is rooted in their much shorter half-lives. Whereas the half-lives of actinides species may be tens of thousands of years, the half-lives of nearly all fission products range from a few seconds up to a few tens of years. Such matters have a profound impact on storage times. After 500 years the radioactivity associated with fission products would be essentially the same as that of the ore from which the original uranium came. This means that after 500 years of storage waste cannisters of fission product wastes could be removed for disposal elsewhere and the facilities could be reused for fresh incoming wastes. Facilities for storing TRU wastes, on the other hand, would not be reusable. Incoming cannisters would require the construction of additional facilities. Construction, accordingly, would go on forever and costs would scale linearly with the volume of incoming wastes.

The best possible solution to the reprocessing-disposal dilemma would be the development of technologies that can economically justify the reprocessing of spent LWR fuel and drastically reduce the amounts of high level TRU (transuranium) wastes that require expensive permanent storage. As so often happens spinoffs from one technology may lead to the solution of other problems. So it may be with the IFR since the technologies under development for its fuel cycle may conceivably be modified for economically reprocessing spent LWR fuel and reducing the amounts of high level TRU wastes.
IFR fuel cycle operations are based on two principal technologies, viz the electrolytic purification of the uranium and plutonium contents of spent IFR fuel and blanket material and the recovery of plutonium from IFR blanket material through the use of a two-phase molten metal-molten salt contact process referred to as "halide slagging." These two technologies can, in principle, be used to develop the following strategy.

- Electrolytically converting the uranium and plutonium contents of LWR fuel to the metallic form.
- Routing the resulting metallic material to the blanket reprocessing stream of the IFR fuel cycle.
- Using a combination of halide slagging and electrolytic purification techniques to separate actinide species from fuel cycle wastes.
- Incorporating actinide species into the fuel of rechargeable IFR elements.
- Incinerating actinide species in the hard neutron spectrum of IFRs.

5.3 The Conversion of Uranium and Plutonium in LWR Fuel to the Metallic State

The fuel discharged from all LWRs consists of fission products and a mixture of uranium dioxide and plutonium dioxide in the approximate ratio of 200 to 1. If the uranium and plutonium content of the fuel could be converted to the metallic state the resulting material could be fed into the IFR fuel
cycle at the same stage as that for blanket elements. At this point the LWR uranium and plutonium lose their identities. From here on they effectively follow the same chemical and physical paths as the uranium and plutonium in discharged IFR fuel and blanket materials. In effect, the two fuel cycles merge into one.

There are two potentially promising possibilities of converting LWR fuel into metallic form. One of these is the simple chemical reduction of uranium and plutonium dioxides by magnesium in the form of a Cd-Mg alloy. This method, unfortunately, suffers from the disadvantage of dealing with large messy quantities of radioactively contaminated oxidic wastes.

There may be a better way. If the uranium and plutonium content of LWR spent fuel could be electrolytically reduced to the metallic form there would be no such wastes. But can the mixed uranium and plutonium content of spent LWR fuel be electrolytically reduced? Probably. One possible way would be to dissolve, or at least partially dissolve, the mixed oxides in a molten salt mixture that contains an oxidizing agent such as uranium tetrofluoride (UF₄). Uranium and plutonium cations (positively charged ions) so formed could then be electrolytically reduced to the metallic state in electrochemical cells similar to those being developed for the IFR fuel cycle.

5.4 Separating Actinides from the Fuel Reprocessing Stream

Radioactive waste materials containing long-lived alpha emitters (half-lives of longer than 20 years) with an activity level of 100 nanoCuries per gram or greater are classified as TRU (transuranium) wastes. This means, essentially, that such wastes must be processed and packaged for permanent storage. Non TRU radioactive wastes, on the other hand, do not require permanent storage. As discussed in Section 5.1 such wastes are simpler to
process and package and can be stored in less expensive reusable facilities. There are, accordingly, strong incentives for separating actinide species from fuel cycle wastes and handling the concentrated actinide fraction separately.

In principle, a concentrated actinide fraction can be handled in either of two ways: processing and packaging it in a suitable form for permanent storage or routing it in the metallic form to the IFR fuel stream for alloying with the metallic U-Pu-Zr fuel. Either alternative would be acceptable. In the first case the number of canisters of TRU wastes would be greatly reduced. And in the second case the actinides would eventually disappear through neutron fission under the fast neutron conditions of the IFR core.

The important question at this time is, "Can the actinides be separated from the various waste forms of the IFR fuel cycle?" Although a great deal of work remains to be done to answer this question definitively the results of thermodynamic calculations, backed up by some laboratory data, are encouraging.

There are five principal actinide species of interest. These are neptunium (Np), uranium (U), plutonium (Pu), americium (Am) and curium (Cm). Of these uranium and neptunium have half-lives so long (4.4x10^9 years and 2.1x10^6 years, respectively) that their specific activities are so low that considerable concentrations can be tolerated in the non-TRU waste stream. The remaining actinides Pu, Am and Cm have much shorter half-lives (2.4x10^4, 432 years and 8500 years, respectively) and therefore much higher specific activities. It is these that must be almost quantitatively removed from fuel cycle wastes if a non-TRU classification is needed.

Most of the actinide separation from spent fuel occurs during the initial step of the IFR fuel cycle i.e. the electrolytic deposition of U and Pu on the cathode of an electrolytic cell. Some Np and Cm also plate out on the cathode and are automatically routed into the normal fuel stream for return
to the reactor and eventual incineration. Americium on the other hand, is not electrolytically reduced; it remains in the molten salt phase.

The electrolytic process is periodically stopped and the cathodes, containing the bulk of the U and Pu along with most of the neptunium and curium, are removed. But substantial quantities of actinides remain behind in the salt phase, the cadmium anode and the cladding hulls. Metallic actinides are oxidized to their respective chlorides by adding cadmium chloride to the salt and metallic cadmium phase of the cell. The actinides, now in their chloride form, enter the salt phase.

At this point there are two waste streams: one, the salt electrolyte, and the other, anode baskets, some cadmium and the cladding hulls. Both streams contain residual quantities of Np, U, Pu, Am and Cm. Of these Pu, Am and Cm must be separated out. Although there are no proven procedures at this point for separating out residual quantities of actinides personnel at Argonne National Laboratory find the following operations promising.

A portion of the salt phase, which contains virtually all of the residual actinides in their respective chloride forms, is periodically removed. Cd-Li alloy is added and the actinide chlorides are reduced to their respective metals which dissolve in the alloy. Thermodynamic considerations are such that the reduction to the metallic form is essentially complete. The salt waste, accordingly, contains such small quantities of residual actinide that it should qualify as a non-TRU waste. Cement and fly ash is added to the salt waste to produce a high level, leach-resistant non-TRU waste for appropriate burial.

The other waste stream, i.e., the metallic, contains residual quantities of actinides that "hide out" in the cladding hulls in the anode basket. To decontaminate the hulls from the actinides a CdCl₂ oxidant is added. The actinide chlorides so formed enter the salt phase leaving the cladding hulls
and basket, presumably, as non-TRU wastes.

The actinide chlorides contained in the salt phase after the oxidation-flushing step are reduced to the metallic form and enter the Cd-Li phase. The two metallic Cd-Li fractions, which contain the dissolved metallic actinides (one from the salt waste stream and the other from the metallic) are combined and fed to a still. Cadmium, with a relatively low boiling point, is boiled and the actinide metals are recovered as still bottoms. These can be added in several different ways to the normal IFR fuel stream for their eventual incineration in the reactor.

In practice small quantities of actinides will contaminate items of equipment such as broken crucibles, pin-casting molds, fume traps, etc. Such items must be processed and packaged as TRU wastes but the important point is this, by far the bulk of the high level radioactive wastes should qualify as non-TRU waste.
6.0 SUMMARY

Of the various concepts under consideration for development as an advanced nuclear powerplants the sodium-cooled pool-type breeder is one of the most attractive. Among its many virtues are the following:

- **Inherent Plant Safety.** All sodium-cooled breeder reactors (both pool and loop-type) tend strongly to protect themselves under all credible undercooling conditions.

- **Essentially Unlimited Fuel Supply.** Breeder reactors effectively "burn" the heavier isotope of uranium, a relatively abundant material that is essentially worthless. The energy content of American uranium reserves when used in breeders is equivalent to the energy available from 30 trillion barrels of oil, an amount 200 times greater than the reserves of the entire Middle East.

- **Conservation & Electrification.** Sodium-cooled breeders respond perfectly to the goals of conservation and the growing practice of electrification by using a worthless and abundant material, U-238, to generate electricity which can be used to do work that would otherwise be done by the burning of enormous quantities of valuable fossil fuels.

- **Environmental Impact.** Sodium-cooled breeders have the smallest impact of all practical power-generating options (including coal and LWRs) on public health, public safety and the environment.

- **Extended Plant Lifetime.** Sodium-cooled breeders are essentially immune to the effects of radiation-embrittlement and corrosion product buildup, phenomena that limit the effective lifetime of LWRs. There is no technological reason why LMRs cannot be operated for 75 years or more.

- **High Thermal Efficiency.** The thermal efficiency of a breeder is approximately the same as that of fossil-fueled plants and approximately 20% higher than the thermal efficiency of LWRs (40% versus 33%).
Sodium as a Coolant. Molten sodium is a remarkably effective coolant. It has excellent heat transfer properties, it is non-corrosive in breeder systems, and its high boiling point eliminates the need for high pressure boundaries (pressure vessels, piping, pumps, etc.). Depressurization phenomena originating from leaks in the primary system are, accordingly, impossible.

Impossibility of Coolant-Cladding Interactions. The complete absence of water, zirconium, and graphite in breeder reactors makes the generation of hydrogen gas impossible.

Simplicity in Maintenance. Breeder reactors are simple to maintain. All major in-reactor components (for both loop and pool-type configurations) are removable, repairable, and replaceable. Furthermore, unlike LWRs the containment building of a breeder is completely accessible during reactor operation.

Simplicity in Operation. Breeders relative to LWRs have less need for shim control. There are, accordingly, fewer control and safety rods. Furthermore, phenomena such as flux tilting, power-peaking, xenon poisoning, etc. are completely absent.

Even if breeders lacked the ability for fuel breeding they would still be attractive as plants that could, if necessary, supplement or supplant the role that is currently filled by commercial LWRs. Attractive as the breeder appears to be improvements are nevertheless possible. The IFR (Integral Fast Reactor) is a conventional sodium-cooled pool-type breeder that is fueled with a plutonium-based metallic alloy. It has all of the virtues of the conventional breeder plus a few more. Among these are:

Making an Inherently Safe System Such as the LMR Even Safer. The physical process that shuts down a breeder under unprotected (no scram) loss-of-flow and loss-of-heat sink conditions is the heatup of the
core coolant. When the amount of negative reactivity fed back through coolant heatup equals the amount of reactivity added to take the plant to power the reactor will shut itself off. Because the Doppler temperature effect in metal-fueled systems is small (relative to that of oxide-fueled systems) less reactivity is needed to take the plant to power. The equalizing amount of feedback reactivity, accordingly, can be achieved at lower core coolant temperatures.

- Closing the Fuel Cycle in a Way That Virtually Eliminates the Threat of Fuel Diversion. Discharged fuel and blanket material are reprocessed pyrochemically and electrochemically in remotely operated heavily shielded colocated facilities. Enough fission products are carried over with the reprocessed alloy to deny even the slightest physical contact with the fuel.

- Improving the Economic Promise of LMRs. Increasing the safety margin through the use of metallic fuels should logically lead to the elimination or simplification of unduly diverse and redundant engineered safety features. Specifications and standards for reactor containment, balance of plant components, in-tank systems, etc. can be relaxed. Reprocessing the fuel in contiguous facilities and handling and storing radwastes in life-of-plant on-site facilities should also work in the direction of lower capital and operating costs.

As promising as the IFR concept appears to be there may be spinoff technologies of equal importance. In the course of research and development concerned with the IFR fuel cycle Argonne engineers may be opening the door to new technologies that may prove useful in solving some of the problems concerned with the disposition of fuel discharged from LWRs (light water reactor). Simply put the IFR fuel cycle is based on two major procedures:
the electrochemical separation of metallic uranium and plutonium from fission products and the pyrochemical enrichment of plutonium from IFR blanket material. In principle, at least, these procedures may be modified for electrolytically reducing the uranium and plutonium content of LWR oxidic fuel material to the metallic state. The resulting metallic material could then enter the IFR fuel stream at the same stage entered by discharged IFR blanket material.

Equally interesting is the possibility of using IFR fuel cycle technologies to separate long-lived alpha-emitting actinides from spent fuel material (both IFR and LWR). The separated actinides in metallic form would then be incorporated into recycled IFR fuel elements where they would fission (incinerate) under the fast neutron environment of IFRs.

The benefits of developing and implementing a common IFR-LWR fuel cycle would be many, among them the following:

- Eventually alleviating on-site fuel storage problems.
- Reducing expensive permanent storage needs of fuel cycle wastes by electrochemically and pyrochemically separating actinide (TRU) species from fuel cycle wastes and incinerating the separated actinides in fast flux reactors such as the IFR.
- Providing the necessary plutonium startup feed for breeder reactors.
- Eliminating problems associated with the handling and storage of high level liquid radioactive wastes.

The breeder has long been maligned as unsafe and unneeded. It would be ironic, indeed, if the IFR, an advanced breeder, prevails as a means of restoring luster to the nuclear image and supplying an energy hungry world with essentially inexhaustible amounts of affordable electrical power.
7.0 REFERENCE


17. L. Witbeck, Argonne National Laboratory, personal communication (March 1984).


APPENDIX A.

EBR-II (EXPERIMENTAL BREEDER REACTOR-II)

Since the IFR concept and the EBR-II plant have many features in common a description of EBR-II and its earlier on-site fuel cycle should serve as a useful introduction to IFR technology.

A.1 Location

The EBR-II complex is located approximately 35 miles west of Idaho Falls, Idaho, at the southeastern corner of the Idaho National Engineering Laboratory (INEL)*. L (formerly the National Reactor Testing Station) was selected in 1949 as a national site for centralizing the construction, operation, and testing of a wide variety of reactor concepts. Since the early 1950's, 52 individual reactor systems have been built and tested at the INEL. Of these, 15 are still operable, the remaining 37 have been phased out upon the completion of their missions.

EBR-II forms the hub of a complex that is dedicated to research and developmental activities under the national breeder reactor program. Other principal facilities at the Argonne-INEL site include the ZPPR (Zero Power Plutonium Reactor); TREAT (Transient Reactor Test Facility); and HFEF (Hot Fuel Examination Facility).

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*EBR-II is operated by Argonne National Laboratory, Argonne, Illinois, for the USDOE under a contract with the USDOE Chicago Operations Office and the University of Chicago.
A.2 History

Born as a concept during the war years and nurtured under the "Atoms for Peace" era of the 1950's and 1960's, EBR-II was originally designed to establish the feasibility of metal-fueled sodium-cooled breeder reactors for powerplant service and to demonstrate the feasibility of on-site fuel reprocessing techniques. By the late 1960's all original objectives were met. As interest in fuel breeding developed, EBR-II became the nation's lead facility for the irradiation testing of fuels, materials, and instruments of interest from the viewpoint of more advanced systems.

Site preparation began in October, 1957; construction began in June, 1958 and was completed in December, 1962; criticality was reached during November, 1963; and on July 16, 1964, EBR-II began its ascent to power. After 22 years of highly successful operation EBR-II continues to serve the nation as a key facility under the USDOE's Civilian Reactor Development Program.

A.3 Pool Type Concept

A principal feature in the design of EBR-II is the pool-type concept which is based on its complete submersion under molten sodium (700°F) of the reactor core, reflector, blanket, neutron shield, primary piping, primary to secondary heat exchanger and in-tank fuel handling equipment. Features that affected the decision to utilize the pool-type concept (EBR-II was the first) include the following:

- 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 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.

Loss of coolant is a virtual impossibility. Surrounding the primary tank is a backup guard vessel and surrounding this is the inner cylindrical edge of the biological shield.

A.4 Plant Layout

The EBR-II plant consists of three principal facilities which house the reactor, steam generation system, and power plant. Heat generated in the core is transferred to the primary sodium coolant which is pumped through the core, into a primary-to-secondary heat exchanger, and discharged directly back to the sodium-filled primary tank. The heated secondary coolant is pumped from the heat exchanger to the sodium-boiler building where steam is produced and superheated. The superheated steam, in turn, is used to drive a turbine-generator located in the power plant building. Condensate along with makeup water is returned to the sodium boiler building for steam generation. After losing its heat to the steam generation system, secondary coolant is pumped back to the heat exchanger in the reactor tank to complete the cycle. In effect, the reactor is a 62.5-MW source of heat which is used to provide steam for electrical power generation.

A central control room, located on the upper floor of the power plant building serves as a "nerve center" for monitoring and controlling the reactor plant and its ancillary systems. Additional facilities consist of HFEF/S and HFEF/N where discharged fuel and irradiation experiments are processed for ultimate disposition. A more functional description of the EBR-II plant is
given below. For convenience the description begins with the reactor itself. Subsequent descriptions follow a format in which components and systems are addressed in the approximate order of their locations with respect to the core.

A.5 Reactor Vessel Top Cover

The top cover consists of two concentric sections: an outer section in the form of a hollow doughnut-shaped shell filled with shielding, and an inner section, equipped with penetrations for 12 control-rod drive shafts. Attached to the underside of the cover is a cylindrical flow baffle. The complete unit is raised during fuel handling to permit access for fuel handling equipment. In its lowermost position, during reactor operation, the cover and flow baffle nest against the reactor vessel to form an upper coolant plenum. From here the heated coolant flows to the intermediate heat exchanger.

A.6 Reactor Vessel

The reactor vessel assembly consists of two concentric shells. The outer shell, 91-in. in diameter and 90-in. high, is made of 3/4-in.-thick type 304 stainless steel. A 14-in. hole in the upper portion is fitted with an outlet nozzle with an inside diameter of 13 in.

The inner shell consists of a cylinder approximately 67 in. in diameter and 71 in. high. The two shells form a cylindrical annulus which is filled with stainless-steel-clad graphite bricks. The bricks serve as a moderator for leakage neutrons, which are effectively captured outside the reactor vessel in a borated-graphite radial shield. The shielding greatly reduces the activation of components located in the primary tank.

A.7 Reactor Core

The basic loading unit is a hexagonally shaped subassembly, 2.290 in.
across external flats and approximately 92 in. long. A subassembly consists of three basic components: an upper adapter, a central hexagonal tube section and a lower adapter (pole piece). In its simplest form a subassembly is nothing more than a hollow container which is used to package fuel, blanket, or other materials for irradiation in the core.

The lower grid plate provides 637 loading positions. Of the 637 positions, 127 are in the high pressure plenum. Any of these may be used for driver subassemblies or irradiation test vehicles. Sixty-six of the 127 positions, in rows 6 and 7, may also be occupied by inner blanket subassemblies. Rows 8 through 16 (at the outer boundary) are in the low pressure plenum. Rows 8-11 are filled with stainless steel reflector subassemblies; rows 12-16 contain depleted uranium.

Of the 127 positions in the high pressure plenum, two in row 3 are occupied by safety rods, eight in row 5 by control rods, and four in row 5 by a reactivity drop rod and three instrumented facilities.

Reactivity is controlled with the eight fueled control rods (with boron-loaded followers) and two fueled safety rods. Any one of the eight control rods may be used for fine reactivity control; all are discharged from the core under rapid shutdown conditions. The two safety rods are always fully inserted; their principal function is to provide shutdown capability in the fuel-handling mode.

A.8 Driver Fuel

The driver fuel (Mark-II consists of 0.130-in. diameter metallic pins 13.50 in. long. The composition of the fuel is 95 wt % uranium metal enriched to 67 wt % of 235U and 5 wt % of a mixture of metallic Mo, Ru, Rh, Pd, Zr, and Nb. The fuel is sodium-bonded in annealed Type 316 stainless steel jackets, 24 in. long. A standard subassembly consists of 91 elements arranged on a hexagonal pitch. Stainless steel sections below and above the elements
serve two purposes: to reduce the neutron fluence on structural components below and above the core and to reflect leakage neutrons back to the core.

Various changes in fuel element design and burnup limits have been made since initial power operation in 1964. In the earlier loadings, fueled with elements of the Mark-I design, fuel burnup was limited to 1.0 at. % (with respect to heavy atoms). Basic changes were made in 1966. The fuel column was shortened, the U-235 enrichment was increased, the gas plenum was enlarged and the pin restrainers were modified. At that time, burnup limits were increased to 1.2 at. %. Subsequent increases were made in the burnup limits, to 1.5 at. % to 1.8 at. % in 1969, and finally to 2.6 at. % in 1975. During the 1970s research and development efforts led to breakthroughs in fuel element design. As a result burnup limits have been increased to 8.0 at. % and plans are being made to increase this to 10.0 at. %. Details that relate to improved fuel element design are given App. B.

A.9 Heat Removal

Essentially all power production takes place in the core and inner blanket region. A small fraction of the total power, approximately 5%, is generated in the outer blanket. Although blanket power tends to increase with operating time, as plutonium accumulates, the periodic replacement of spent blanket subassemblies tends to keep the core/blanket power ratio constant. Each region is cooled separately by upward parallel flows of primary coolant. The two streams merge and mix in a plenum immediately above the reactor and beneath the reactor vessel cover. From this point on, heat dissipation proceeds via three principal system: The primary sodium system, the secondary sodium system, and the steam system.
A.10 Primary Sodium System

Two centrifugal pumps, each rated at 4500 gpm, take suction from the bulk sodium in the primary tank and discharge sodium to the lower plenum. Flow from the pump outlets is split into two streams: one which enters the high pressure plenum at approximately 50 psi, and the other which enters the low pressure plenum at approximately 17 psi. Throttle valves in the two low pressure delivery systems are used for pressure reduction.

Sodium flows upward through the core and blanket to the upper plenum, where it leaves the reactor through a single 13-in.-ID outlet pipe and flows through an auxiliary dc electromagnetic pump. From here the stream flows to the intermediate heat exchanger, where the heat from the thermally hot and radioactive primary system is transferred to the secondary sodium system. Primary coolant enters the heat exchanger at 883°F and is discharged directly to the bulk sodium at 695°F. Principal radioactive species in the primary sodium systems are Na-24 and Na-22. Under sustained operating conditions the activity levels of these species approach 3.0 mCi/g and 0.13 uCi/\(\text{g}\), respectively. The secondary sodium, on the other hand, is essentially radioactively inert. Na-22 is missing and the Na-24 component may approach a maximum activity level of 40 nanocuries per gram.

The auxiliary pump operates continuously. it serves solely as a means of removing decay heat from the core in the event of a primary pump coastdown. The pump, rated at 500 gpm, operates on a rectifier-battery system which guarantees flow even if all electrical power should be lost.

A.11 Reactor Containment Building

The reactor containment building is a gas-tight shell which houses the reactor, primary tank, shielding, and all ancillary primary components. The shell, usually referred to as the reactor containment building, consists of a carbon steel cylinder, 80 ft. in diameter, up to one inch thick, with a
hemispherical top and a semi-ellipsoidal bottom. Reinforced concrete lines the inside of the cylindrical portion of the shell to a depth of 12 in. Five inches of reinforced concrete line the inside of the hemispherical dome.

The shell was designed to withstand an internal pressure of 24 psig and a maximum leak rate of 1000 ft$^3$/day under a pressure differential of 20 psig. Three airlocks provide the means for moving equipment and personnel into and out of the reactor building without violating building containment. All areas of the reactor building are accessible during normal reactor operation.

A.12 Secondary Sodium System

The nonradioactive secondary sodium system is isolated from the radioactive primary sodium at the intermediate heat exchanger (IHX). Secondary sodium under pumped flow of 5000 gpm enters the IHX at 588°F and leaves at 875°F. It then flows to the sodium boiler building where it flows upward through two superheaters and downward through eight water evaporators. The streams from the lower portions of the evaporators then flow to a surge tank located at the highest point in the system. Sodium is pumped from the surge tank back to the IHX to complete the loop. All portions of the secondary sodium system are located at elevations higher than the level of primary sodium. This feature prevents the entrance of primary sodium into the secondary system in the event of an IHX leak between the two systems and provides the capability for natural circulation in the secondary system.

The secondary sodium system includes a subsystem designed to relieve overpressures in the event of a leak between the sodium and steam-water systems. A tank, isolated from the secondary sodium system by means of rupture discs, provides containment for reaction products in the event of a steam-water leak into the secondary sodium.
A drain tank and related piping and pumps provide the means for draining the secondary system. The inventory of sodium in the secondary system is about 12,000 gal.

A.13 Steam System

Superheated steam at 1250 psi and 820°F is piped from the superheaters in the sodium boiler building to a 20-MWe turbine generator located in the power plant building. After passage through the turbine, the discharged steam is condensed and recycled through a condensate and feedwater system. Low grade heat in the condenser cooling water is dissipated to the atmosphere via a forced draft evaporative cooling tower.

Two modes of generator control are available. The mode that is usually used consists of controlling steam pressure in the main steam header with the turbine-generator synchronized to the power grid. The other mode limits the output of the generator with a speed governor control system. Excess steam is dumped to the condenser and the heat is dissipated to the atmosphere via the cooling tower and condenser cooling system in the second mode.

A.14 Fuel Handling

All fuel handling operations in the primary tank are carried out with remotely operated equipment under submerged sodium conditions. Principal items in the fuel handling chain are the following: gripper, subassembly holddown, fuel transfer arm, fuel storage basket, and a system of rotating plugs. The subassembly holddown and core gripper are physically mounted on the small rotating plug which, in turn, is eccentrically mounted in the large rotating plug. Through the precision rotation and indexing of the two plugs, the holddown mechanism and subassembly gripper can be placed over any of the 637 core and blanket subassemblies.
The fuel handling system is specifically designed to permit two modes of operation. In the unrestricted fuel handling mode, the reactor is shut down and subassemblies may be inserted into or removed from the core. In the restricted mode, subassemblies may be transferred to or from the primary tank with the reactor operating.

The unrestricted fuel handling mode involves the following sequence of events. The reactor is shut down with all control rods driven to their lowermost configurations, the control rod drives are disconnected and raised, and the reactor cover is raised to its uppermost position. The two rotating plugs are then rotated to place the holddown mechanism and subassembly gripper over a specific core position. The holddown mechanism and core gripper perform the following respective functions: to secure the six neighboring subassemblies and to engage the lifting adapter of the subassembly of choice. The gripper, along with its engaged subassembly, is raised to the elevation of the transfer arm. The subassembly, now completely clear of the core, is transferred to the transfer arm. At this point the storage basket is raised, rotated to its proper position, and the subassembly is placed in a specific location of the storage basket. The insertion of a subassembly into a core position follows the reverse order.

The restricted fuel handling mode involves the transfer of a subassembly from the storage basket to the transfer arm or the reverse. With the subassembly firmly engaged, the transfer arm is rotated to a position under the fuel transfer port. Located above the fuel transfer port is a track-mounted shielded cask called the FUM (fuel unloading machine). A gripper in the FUM is lowered to engage the lifting adapter. When free of the transfer arm, the subassembly is lifted into the cask with the FUM gripper. The FUM is then moved along a set of rails to a position which permits the subassembly to be lowered into a portable shielded cask called the IBC (interbuilding coffin). Following this transfer the FUM is backed away from the IBC and the IBC is lifted with a polar crane and lowered through an air lock onto a railed cart. The air lock is closed, and the cart is driven to a second air lock which
opens into a corridor in the HFEF/S Complex. From here the IBC with the discharged subassembly may be moved to facilities in either HFEF/S or HFEF/N. In general, spent fuel assemblies are handled in the air cell of HFEF/S, whereas irradiation subassemblies are processed in HFEF/N.

Irradiation subassemblies containing irradiated materials may be reinserted in the core via the reverse route. Fresh fuel subassemblies, on the other hand, may be loaded directly into the FUM. Since initial reactor operation in 1964, over 6000 round-trip transfers have been made.

A.15 Important Subsystems

Many of the EBR-II subsystems that are operating today were designed with the technologies of the early 1950's. Despite the unavailability of important design information at that time, the systems have performed remarkably well. Some of the more important systems that were designed on a first-of-a-kind basis are discussed below.

A.16 Steam Generator

The building of reliable steam generators for sodium-cooled reactors is a matter of worldwide interest. Experience with the EBR-II steam generator has shown that reliability and efficiency are the logical consequences of sound design and construction practices. The EBR-II steam generators have operated reliability since 1964. Since that time, the only difficulty consisted of a faulty but easily repairable weld.

The steam generator system consists of eight evaporators connected in parallel, a steam drum, and two superheaters also connected in parallel. The evaporators are supplied with water from the steam drum, which is located at the high point in the system. Water flows from the steam drum through the evaporators. The water-steam mixture returns to the steam drum under natural circulation.
Construction details of the superheaters and evaporators are similar; both are vertical, straight-tube, counter-flow heat exchangers. The tubes in each superheater and evaporator consist of two bonded concentric tubes. Sodium flows on the shell side of the exchangers. Steam and water, on the other hand, flow in the tube side of the superheaters and evaporators.

The bonds between the concentric tubes consist of two types. The tubes in four evaporators and one superheater are mechanically bonded by drawing the concentric tubes through a die and over a mandrel and then expanding the inner diameter with a subsequent drawing operation. This procedure leaves the tubes in a prestressed condition with the outer tube in tension and the inner tube in compression. The tubes in the remaining four evaporators and one superheater are metallurgically bonded by the predrawing treatment of mating surfaces with a nickel, nickel-phosphorus alloy.

A.17 Sodium Pumps

The two primary sodium pumps are single stage centrifugal pumps with hydrostatic sodium lower bearings. The pump motors are sealed to the pumps and the motors are filled with argon gas. Aside from early difficulties, operational experience with the pumps has been excellent. On three occasions, twice in 1963 and once in 1971, pumps were removed to correct problems caused by shaft bowing and galling.

The secondary system utilized a single ac three-phase linear electromagnetic pump for the main system loop. The pump not only provides normal flow, but is also used to restrict natural convective flow in the system when the reactor is not operating. Reversing the pumping direction and "bucking" the convective flow allows control of sodium flow to within 1/10 of 1% in the secondary system during shutdown. Only one repair, the welding of a crack in the pump tube, has been necessary since 1964.
Pump operation experience in EBR-II has shown that properly designed pumps can operate satisfactorily in a sodium environment.

A.18 Intermediate Heat Exchanger (IHX)

The primary sodium to secondary sodium heat exchanger (IHX) is another major component at EBR-II that has demonstrated the high operational reliability of components submerged under sodium.

The IHX is a counter-flow heat exchanger that has 3026 stainless steel tubes, each of which is 5/8 in. in diameter. Secondary sodium flows through the tubes and primary sodium flows around the tubes on the shell side. Except for a broken pipe clamp on a drain tube inside the IHX, no operational difficulty has been experienced.

A.19 The Use of Metallic Fuels in EBR-II

EBR-II is the only operating FBR (Fast Breeder Reactor) in the world with metallic driver fuel. All but one of the others, either planned, under construction, or operating are fueled with a mixture of PuO$_2$ and UO$_2$.*

Several considerations affected the initial design of the EBR-II driver fuel during the early and mid-1950's. These were: the relative ease of fabrication, the success of four metallic fueled loadings in EBR-I, the excellent heat transfer properties of metals, the superior breeding characteristics of metallic fuels, and the inherent promise of a simple and rapid on-site turnaround of discharged fuel.

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*BOB-60, in the USSR, is an exception: BOR-60 is fueled with enriched UO$_2$. 
From the start, operation with metallic driver fuel was successful. As operating experience accumulated, burnup limits were incrementally increased from the original value of 1.0 at.% (established in 1961) to 1.2 at.% in 1966, to 1.8 at.% in 1969, and to 2.6 at.% in 1975. Beyond 3.0 at.%, fuel swelling became a problem. Pressures exerted at the fuel cladding interface were so large that cladding strain and subsequent rupture seemed likely. Such limitations were always evident. Apparent, even in the early days of design, were the inevitable penalties: short operating cycles and inefficient fuel utilization. However, the impact of the penalties was softened by the rapid on-site reprocessing of discharged fuel. On certain occasions discharged fuel was reprocessed and returned to the core within 29 days.

Despite the successful implementation of short turnaround reprocessing techniques, considerable effort was devoted to the development of fuels which could be operated to much higher burnup levels. One concept that seemed particularly promising in the early days of operation was the permissive swelling principle. Basically, this concept was premised on the contention that if fuel material is permitted to swell in an unconstrained manner, a point will be reached (at approximately 30% V/V when pores in the fuel matrix become interconnected, thus permitting the release of trapped fission product gases to the fuel element plenum. The fuel, in a weakened condition, tends to deform into the pores rather than straining the cladding.

The proof of this principle appeared in the form of an improved fuel element design (Mark-II). The cladding thickness was increased from 9 to 12 mils to strengthen the cladding, the pin diameter was reduced from 144 to 130 mils, the Na-filled annulus was increased from 6 to 10 mils, and the gas plenum (above the fuel) was increased to accommodate the increased release of fission product gases, the $^{235}$U enrichment was increased to 67 wt % and the cladding was changed to annealed Type 316 stainless steel. Intensive irradiation-surveillance studies on test specimens were conducted and, as a
result, it was shown that fuel material swelled rapidly and made contact with the cladding at approximately 2 at.% burnup. At this point, the fuel lattice becomes sufficiently porous to permit the relatively free flow of fission product gases to the fuel element plenum. Beyond this point, fuel material remains in contact with the inner wall of the cladding but, because the fuel is now porous and "weak", and because the driving force for diametral swelling is low, the cladding strain rate is low. As a consequence, the lower stress on the jacket permits much higher burnups to be realized.

At the present time, the burnup limit for Mark-II fuel is 8.0 at.%. The results of a continuing irradiation-surveillance program imply that even higher levels are possible, since endurance tests in EBR-II, up to 18.9 at.%, have been successfully completed. Of considerable interest are the following statistics: Of approximately 130,000 fuel elements used during the period 1964-1985 only two suffered cladding rupture. Performance statistics such as these are as good as those expected for fuel elements containing mixed Pu and U oxides.

A.20 Operating Schedule

The EBR-II plant is operated on a 24-hour 7-day-per-week basis. Four crews work normal 8-hour shifts according to a rotating schedule. Run lengths are nominally 3450 MWd or 55 full-power days. Two factors determine the run length: the need to discharge irradiation experiments at scheduled intervals and the need to replace spent fuel with fresh fuel. Approximately 7 days are needed between runs to accommodate refueling operations and to perform minor maintenance activities that cannot be carried out with the plant running. Each year the plant is shut down for 4-6 weeks to carry out more comprehensive modification, maintenance, and inspection activities.
A.21 Operational Stability

The operation of EBR-II has always been kinetically stable. A prompt negative power coefficient of reactivity from the expansion of coolant and fuel effectively damps the effects of small reactivity changes caused by inlet temperature variations, control rod motion, etc. The amplitude of the prompt power coefficient component is monitored on a run-to-run basis. Reactivity is rapidly withdrawn from the system by dropping a special rod and analyzing the shape of the power decay curve with digital computing equipment.

A.22 Component Performance

Aside from problems of a relatively trivial nature, all major systems and their subsystems have performed well. The following systems have been particularly trouble free: primary sodium, secondary sodium, steam, power-plant, and plant instrumentation (flow, temperature, pressure, etc.). Many difficulties have been experienced with the fuel handling and rotating plug systems, but even these were correctable by relatively simple modifications.

A.23 Power Output

EBR-II is normally operated at its full power rating of 62.5 MWe. Under full power conditions, 19.5 MW of electrical energy is generated. Approximately 5.5 MW is used to satisfy the ANL-W demand; the remaining 14 MW is fed through a 13.8 kv loop to the INEL distribution grid. In addition, approximately 12,000 lbs. of saturated steam per hour is utilized for local space heating. The use of plant steam results in an annual savings of approximately 400,000 gal of oil. The savings of electricity and fuel oil costs and revenues from the sale of electricity amounts to approximately $1,000,000 /year.
A.24 Plant Availability

During the period 1976-1985, EBR-II operated with plant availability factors greater than 70%. A peak value of 76.9% was reached during 1976. Plant availability factors in this range compare favorably with those for commercial nuclear and conventional fossil-fueled power plants. If downtime required by the experimental program is discounted, the actual plant availability factor during the 1976-1985 period exceeded 80%.

A.25 Refueling

Refueling time between runs is not a serious constraint. The capability of interim in-tank storage permits the preshutdown transfer of fresh fuel subassemblies to the storage basket. Approximately 4 hr after shutdown, spent subassemblies may be transferred from the core to the storage basket and replaced with fresh subassemblies. The turnaround time per subassembly amounts to approximately 1 hr. In the absence of problems, the time required for end-of-run refueling operations amounts to approximately 14-48 hr. The interim fuel storage feature is beneficial in another important respect. After fulfilling minimum cooling requirements, spent subassemblies may be transferred into and out of the storage basket while the reactor is running.

A.26 Fission Product Releases

Routine operations are occasionally interrupted by the release of gaseous fission products from a failed fuel element. The majority of the releases are the inevitable results of endurance tests in which experimental fuel elements are intentionally irradiated to failure and beyond. On other occasions, the failure may be the result of birth defects in the element (e.g. a faulty weld) or by premature failure of cladding. More recently, the effects of sustained operation under breached cladding conditions have been investigated. Fuel elements which have failed under irradiation are permitted to remain in the core for periods up to a few weeks in order to evaluate the effects of
sustained operation on the fuel element and the consequences of releasing fission products to the primary coolant and cover-gas systems.

A portion of the program is concerned with the effects of fission products on the primary sodium and cover-gas systems. Species released to the coolant are removed, in part, by means of an externally located cold-trap. Gaseous fission products released to the cover-gas system are removed by a combination of cryogenic trapping and absorption by liquid-nitrogen-cooled charcoal beds.

A.27 Maintenance Experience

Many years of operating experience have demonstrated the feasibility of conducting routine and special maintenance on FBR 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. Experience has shown that prudent procedures coupled with ingenuity form a maintenance technology that can be additionally developed for more advanced FBR systems.

Maintenance activities conducted on EBR-II and its peripheral plant systems may be assigned to three general categories that reflect accessibility and hazards concerned with radioactivity, viz, work on conventional, readily accessible and nonradioactive components; work on unconventional, readily accessible but moderately radioactive components; and work on conventional, relatively inaccessible, and highly radioactive components.

Activities that fall under the first category are classified as preventive maintenance. Typical systems subject to preventive maintenance are those located outside the primary system. Preventive maintenance consists essentially of periodic inspection, repair, modification, and replacement, if necessary. Depending on the component and its function, preventive maintenance activities may or may not be conducted with the plant in operation.
If the plant must be shut down, work activities are scheduled to coincide with the next scheduled shutdown.

Considerable maintenance experience with unconventional, relatively accessible, and moderately radioactive components has been achieved. A typical component falling in this category is the transfer arm (part of the fuel handling system). Precautionary measures associated with such activities are those needed to maintain an argon atmosphere around the component, to prevent the inleakage of air to the cover-gas system, and to prevent the leakage of cover gas to the reactor building. All such activities are based on pulling the component into a rubberized-nylon bag which is sealed by a flange to a penetration in the primary containment tank.

The most difficult components to maintain are those that are unconventional, relatively inaccessible, intrinsically radioactive, and radioactively contaminated. Examples of such components are control-rod drives, the main core gripper, and the subassembly holddown fixture. Such components penetrate the reactor vessel cover and, as a result, become highly radioactive through neutron activation. The problems of intrinsic radioactivity, sodium contamination, and fission-product plateout make direct repair virtually impossible. In situations such as these, e.g., a malfunctioning control rod drive, the entire unit is removed and replaced. When components in this category are being removed, the system must be "cooled" for approximately 5 days to permit the decay of $^{24}\text{Na}$. The component is then pulled into a shielded pipe that is handled by the building crane. As for all activities that involve access to the primary tank, precautionary measures must be taken to prevent the inleakage of air to the cover gas system and the leakage of cover gas to the reactor building.

As a result of over 22 years of successful experience with removing, cleaning, maintaining, repairing, and disposing of primary tank components, it is reasonable to conclude that with ingenuity and forethought, FBR's can be designed to permit maintenance-repair activities on in-tank components.
APPENDIX B.

ON-SITE FUEL REPROCESSING ACTIVITIES AT EBR-II, 1964-1969

One of the principal features of the IFR will be a remotely operated on-site facility for reprocessing spent fuel and refabricating the upgraded product into rechargeable fuel elements.

Many of the necessary technologies are already in place; they were developed and implemented during the period 1964-1969 when EBR-II was actually operated as an IFR. Fortunately, the original facility known as the FCF (Fuel Cycle Facility) is still in place. It has been decontaminated and, if existing plans materialize, it will be used to demonstrate various aspects of IFR technology under pilot plant conditions. Because of the many similarities that exist in the earlier processing of EBR-II fuel and that envisioned for the IFR a description of the FCF and its earlier operation is considered worthwhile.

B.1 Background Information

During the early design stages of EBR-II, circa 1953, considerable effort was devoted to the selection of driver fuel. Many factors soon narrowed the selection of fuel to metallic alloys since these satisfied the following criteria: high fissile density, excellent heat transfer properties, ease of fabrication, low neutron moderation, high breeding ratio, and high specific power density. Oxide fuels were considered and rejected because of their lower breeding ratios and the absence of an established technology for fuel fabrication. An additional consideration that influenced the early selection of a metallic fuel loading was the limited but satisfactory experience achieved with metallic fuels in EBR-I.

Prevalent at that time was the belief that metallic fuels would suffer from a serious disadvantage, viz, their tendency to swell and strain the cladding at relatively low levels of fuel burnup. Obvious, even then, was the adverse impact of low fuel burnup on run length and fuel utilization efficiency
Equally obvious was the need to develop a technology for the rapid "turnaround" of spent fuel in the reprocessing cycle. Since fuel material in the storage-reprocessing-refabrication cycle must be charged to total fuel inventory, a rapid turnaround of spent fuel was considered an economic necessity.

Clearly needed was a process that satisfied the following criteria:

- Short "turn-around" for recycled fuel
- Low fuel inventory
- Low capital plant costs
- Capability for handling "short-cooled" fuels
- On-site reprocessing
- Minimum solid wastes

The process that satisfied these criteria and which was implemented from the beginning was based on a pyrometallurgical technique in which spent fuel was melted, separated from bulk fission products, and recast into fuel pins for return to the reactor.

B.2 The FCF

The first loading of EBR-II fuel was manufactured from "cold" material, using prototypal fuel fabrication equipment under "cold" laboratory conditions. All subsequent reprocessing operations were carried out in a complex called the FCF (Fuel Cycle Facility). In its simplest form the FCF consisted of two heavily shielded and remotely operated facilities: one, the air cell, in which spent fuel subassemblies were disassembled and fresh fuel elements reassembled into subassembly form; and the other, the argon cell, in which all fuel element reprocessing and fabrication operations were performed. All operations in both the air cell and the argon cell were performed with the aid of bridge cranes, electromagnetic bridge manipulators, and master-slave manipulators. Specialized equipment contained in the cell was actuated
from the various operating corridors via electrical, electromechanical, and pneumatic means. All in-cell equipment was designed for either remote repair or replacement on a modular basis.

The first reprocessing of irradiated fuel began in September, 1964 and continued until April, 1969. During this period approximately 6000 kg of irradiated fuel material was processed. This amount was equivalent to 353 subassemblies or approximately five full core loadings. Turnaround times for fuel during this period averaged approximately 45 days after arrival at the FCF. On certain occasions, however, irradiated fuel material was returned to the reactor within 29 days.

Until the FCF shutdown in April, 1969, over 40,000 fuel elements (including the initial cold charge) were used to fuel the reactor. Of these, only one failed in service. The average availability of remote handling and process equipment during the active lifetime of the FCF was approximately 90%.

B.3 Fuel Flow

The first phase of fuel recycle operations began in the reactor. Subassemblies that had reached their target burnups (1.2 at.% at that time) were transferred from the core to the in-tank storage basket. Vacancies left in the core were filled by transferring fresh fuel subassemblies stored in the basket.

After a cooling period of 14 days, spent fuel subassemblies were ready for transfer from the reactor tank to the FCF. Transfer operations were relatively simple. By means of a system of shielded casks, airlocks (between the reactor building and the FCF), and cranes spent fuel subassemblies were placed in the air cell for disassembly operations which consisted, essentially, of slitting the wrapper can and removing individual elements from the fuel bundle. Fuel elements, in turn, were loaded into magazines and moved
to the argon cell via an interconnecting lock.

Initial operations in the argon cell consisted of the following: shearing upper and lower ends from the fuel elements, removing spacer wires, removing the cladding, and chopping the spent fuel pins into 1-1/2-in. lengths. Chopped fuel, in amounts of 10-12 kg, along with precalculated amounts of enriched uranium were melted in CaO-coated zirconia crucibles and held for 3 hr at a temperature of approximately 1400°C. During this period, fission products with low vapor pressures, viz., Xe, Kr, I, Br, Cs and Cd, left the melt through volatilization. Less volatile electropositive fission products such as Y, Ba, Sr and the rare earths reacted with the zirconia to form an oxide which remained with the crucible after pouring operations. Other less reactive fission products, principally those with atomic numbers 40 through 46, remained with the melt.

Pouring operations were never 100% efficient. Inevitably, a relatively small but consistent amount of fuel, approximately 6-8% remained in the crucible along with the dross in a configuration that resembled a skull. The skulls as they soon became known, were set aside in hot storage for ultimate uranium recovery.

The loss of fuel material through skull formation was not necessarily serious and, in one important respect, was actually beneficial. The less reactive fission products, viz., Zr, Ru, Rh, Pd, and Nb, which remained with the fuel enhanced the metallurgical and radiation-resistant properties of the fuel when present to a collective extent of approximately 5 wt %. Since these species built up during the irradiation portion of the cycle and were removed from the stream with the skulls, an equilibrium concentration would have eventually been reached. To achieve an earlier equilibrium, carefully calculated amounts of the noble metals (atomic numbers 40-46) were added to the original fuel materials. This material, known as fissium, was present in the original fuel material in the following amounts: Zr, 0.1%; Mo, 2.4%; Ru, 1.9%; Rh, 0.3%; Pd, 0.2%; and Nb, 0.01% (all weight-percentage values).
Had there been no loss through skull formation, or "dragout" as the process was called, the concentration of the noble metals would have increased through repeated cycling to a point where their effects on reactivity and fuel performance would have been noticeable.

The first step in pin-casting operations consisted of remelting a melt-refined ingot in a thoria-coated graphite crucible in an induction-heated vacuum furnace. Also located in the furnace in a vertical attitude above the crucible was a cluster of approximately 100 Vycor molds. After the charge was melted, the crucible was raised to a position that immersed the lower end of the mold cluster to a depth of 1-1/2-in. The furnace was then rapidly pressurized to 1.7 atmospheres to drive the melt upward into the evacuated molds. After a few seconds, the melt froze and the crucible was lowered to its original position. Following a 4-hr programmed cooling period, the furnace was opened and the molds were removed. Unused fuel material in the crucible was broken up and returned to the fuel stream.

The first step in pin-processing operations consisted of breaking the Vycor molds away from the castings with a pneumatically actuated crushing system. Pins, collected in trays, were fed by gravity to a shear that cropped the pins to required lengths. Following cropping operations, the pins were subjected to nondestructive tests which consisted principally of measurements for weight, length, diameter, and porosity. Rejects were fed back to the melt refining operation.

Acceptable pins were loaded into stainless steel tubes containing solid sodium wire (approximately 0.65 to 0.85 g). The sodium was melted and the fuel pins settled by gravity. End-plugs were inserted into the jackets and peripherally welded to the jackets with a remotely operated capacitor-discharge welder. To ensure void-free sodium bonds, the finished elements, clustered 50 to a magazine, were heated in a furnace for one hour at 500°C. The elements were then subjected to approximately 100 vertical impacts (under 500°C conditions) at the rate of 100 impacts per minute. The finished elements
were then inspected for bond-flaws and sodium level. The final step consisted of mounting the elements on a grid-system and incorporating the resultant fuel bundle in a fresh hexagonal wrapper can for ultimate return to the reactor. After nearly five years of highly successful operation, the FCF was shut down (1969). Subsequent core charges were made either by vendors in the commercial sector or by on-site personnel in a companion "cold" line facility.