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ABSTRACT

The Experimental Breeder Reactor-II (EBR-II) is a sodium cooled power reactor supplying about 20 MWe to the Idaho National Engineering Laboratory (INEL) grid and, in addition, is the key component in the development of the Integral Fast Reactor (IFR). EBR-II's testing capability is extensive and has seen four major phases: (1) demonstration of LMFBR power plant feasibility, (2) irradiation testing for fuel and material development, (3) testing the off-normal performance of fuel and plant systems and (4) operation as the IFR prototype, developing and demonstrating the IFR technology associated with fuel and plant design. Specific programs being carried out in support of the IFR include advanced fuels and materials development, advanced control system development, plant diagnostics development and component testing. This paper discusses EBR-II as the IFR prototype and the associated testing programs.

INTRODUCTION

In the initial phase of EBR-II's testing program, beginning in 1964, EBR-II's mission was to demonstrate the feasibility of an LMR as a complete power plant. That demonstration included reprocessing the metal uranium core in a facility built as an integral part of the plant. Reprocessing was done remotely, with fuel returned to EBR-II via an interconnecting tunnel through which the transfer casks traveled. The demonstration was successful for both plant performance and reprocessing.

The second phase of EBR-II's testing program began in 1968 when it was decided to emphasize oxide fuel. EBR-II's core was modified by installation of a radial reflector to enhance irradiation testing of fuels and materials. In addition, a number of sophisticated instrumented test facilities were designed and installed, replacing up to four of the 12 control rods. This program has also been very successful. The fuel for FFTF and CRBR was developed in EBR-II and many other irradiation testing programs continue to be supported.

The third phase of EBR-II's testing program began in 1978 and involved testing the response of fuel and the plant to off-normal conditions such as breached cladding, anticipated operational transients, and more severe events such as loss of offsite power with failure to scram. In April of 1986, two landmark tests were run on EBR-II to demonstrate

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passively safe response to loss-of-flow and loss-of-heat-sink, both without scram. Many modifications were made to support such testing. Fission product control was upgraded by the addition of a cover gas cleanup system. A computer controlled rod drive was added to allow shaped power transients to be conducted and the core and plant were qualified for transient testing through a combination of analysis and testing. Much work was directed to instrumented test facilities for measuring temperature and flow in the core. A major effort was also directed toward improving the dynamic simulation of EBR-II. This phase of testing has also been very successful. A major fuel-testing program is being conducted with PNC of Japan to address the run-beyond-clad-breach and transient overpower performance of oxide fuel.

This work has prepared EBR-II very well for its present mission of developing and demonstrating IFR technology as the prototype for the next plant. In fact, work on the three missions identified earlier has never stopped. Through this period, significant work was directed toward improving the performance of EBR-II metallic driver fuel. It's burnup has been greatly increased, its run-beyond-clad-breach performance was demonstrated, and its ability to accommodate power transients was proven. Characterization of plant performance for both steady state and transient operation was extensive. EBR-II is well prepared to take on the IFR prototype mission in addition to its role as a major irradiation testing facility for other programs.

EBR-II AS THE IFR PROTOTYPE

The Integral Fast Reactor (IFR) is derived from the EBR-II concept and demonstrated technology. The concept includes a uranium-plutonium-zirconium (U-Pu-Zr) metallic fuel, a pool type LMR, high breeding ratio core designs, core designs and plant design to enhance passive safety and an integral pyrometallurgical fuel processing facility with remote injection casting of reprocessed metal fuel. These features are incorporated or have been demonstrated in EBR-II and will continue to be developed to refine design concepts and demonstrate performance.

Use of EBR-II to develop and demonstrate the IFR technology has four main components as follows:

(1) Fuel and Fuel Cycle: To develop and demonstrate in conjunction with the ANL Fuel Cycle Facility, satisfactory performance of reprocessed U-Pu-Zr fuel.

(2) Safety: To demonstrate safety advantages and to set the precedent for safety documentation appropriate to an IFR.

(3) Operation: To demonstrate operational advantages of EBR-II/IFR, considering benign response to controller malfunctions and benefits of sodium for operation and maintenance.

(4) Regulation Compliance: To demonstrate that compliance with regulations involving operation of EBR-II/IFR is enhanced by virtue of characteristics inherent in design.

The designation of EBR-II as the IFR prototype follows from the demonstrated fuel behavior (>18 at.%)[1] passive safety features of the metallic fuel,[2] demonstrated thermal performance during upset conditions (loss-of-flow and loss-of-heat-sink without scram tests)[3] and soon to be demonstrated fuel reprocessing and the integral closed fuel cycle.[4] EBR-II is also the test facility designated to demonstrate self consumption of plutonium and the actinides[5] thereby greatly reducing the high level waste problem and reducing the time for monitored fuel storage by several orders of magnitude.

As mentioned above, the IFR will incorporate an integral fuel cycle where the spent fuel from EBR-II will be transferred to the fuel cycle facility for reprocessing and fabricated remotely into new assemblies and returned to the reactor. The U-Pu-Zr fuel is central to the process. The initial step of the processing occurs in the electrorefiner where the chopped fuel (cladding included) is dissolved in a halide salt solution. The fuel remains in the metallic state during the cycle and the cladding material and fission products are separated to the salt phase and the uranium and plutonium are collected at the cathode.

The cathode is processed and the resulting fuel and actinides are melted in a casting furnace and new fuel pins are made by an injection casting technique. A very similar fuel cycle was demonstrated in EBR-II during the first five years of reactor operations.[6] Over 700 irradiated assemblies of all types were processed. Of these, 560 were fuel bearing subassemblies, and these were processed to separate the fuel and most of the fission products. This operation produced 34,500 reprocessed fuel elements that were fabricated remotely into 418 subassemblies and returned to the reactor. EBR-II, along with the associated Fuel Cycle Facility, has come full circle and will serve as the prototype and test bed for IFR fuel cycle development and other IFR technologies.

EBR-II has another equally important role for the IFR. This derives from the passive safety features of the concept and promises substantial savings in operations, safety documentation preparation and maintenance, and environmental and radiological impacts. The EBR-II safety documentation is currently being revised to better support the operation as the IFR prototype and to reflect the advantages of the IFR in areas of safety, health and the environment. Also, work is ongoing to investigate and demonstrate the operational safety features of the IFR that result in savings in operations. This promises innovative approaches to such issues as number of operations personnel, training requirements for emergency response, automation of control and procedural functions, and reduced maintenance requirements due to fewer safety grade systems and components.

FUELS IRRADIATION PROGRAMS

EBR-II has been employed as a test bed for LMR fuel irradiations since the late 1960s. This use had included developments of the reference mixed-oxide fuel plus designs for the Fast Flux Test Facility (FFTF) and the Clinch River Breeder Reactor, as well as advanced fuels to promote breeding in the form of sodium-bonded and gas-bonded carbides and nitrides. When FFTF became available for steady-state testing in 1980, further development of oxide fuels at EBR-II took the form of Operational

Reliability Testing (ORT), in collaboration with PNC of Japan. This work comprises international RBCB tests and the in-site simulation of operational transients. Phase II of the ORT program is now in progress, following the successful conclusion of Phase I tests.[7,8] The improvements that have been steadily made in the performance of the sodium-bonded U-metal driver fuel of EBR-II over more than 25 years of operation have also laid the basis for the fuel design employed in the Integral Fast Reactor concept, which is now the focus for LMR development in the US.

IFR Irradiation Program

Although irradiation tests of a number of U-Pu-Zr alloys with various cladding materials in EBR-II in the late 1960s established the principle of the sodium-bonded LMR metal fuel system, only a few fuel pins reached burnup of 6 at.% before the program was terminated. Thus, at the beginning of the IFR program in 1984, the higher priority fuels activity was to begin to build a data base on the U-Pu-Zr fuels. After an aggressive fuel fabrication campaign, the irradiation test program has been ambitious with a comprehensive range of design and operating parameters. The U-Pu-Zr fuel composition has been varied from no plutonium to 28 wt % plutonium with zirconium variation from 2 to 14 wt %. Three cladding materials have been used, which include the two austenitic claddings, AISI type 316 and D9, and the ferritic cladding, HT9. In addition, the plenum-to-fuel ratio, smear density, linear power, and fuel and cladding temperatures have covered a wide range. The maximum burnup achieved to date is 18.4 at.%, which far exceeds that achieved earlier in the 1960s and is certainly comparable to other LMR fuel types. In general, the steady-state irradiation performance has been excellent, as summarized in parallel papers.[1,9]

The irradiation test program and out-of-reactor tests and analyses also include an important component aimed at off-normal performance of the fuel. RBCB behavior after cladding breach, as well as fuel performance during reactor transients and beyond-design-basis events, require in-depth understanding. So far RBCB performance has been shown to be excellent, with exposure times of up to 200 days.

The main question associated with off-normal performance is the temperature at which a liquid phase forms between fuel and cladding, and the rate at which cladding is penetrated once a liquid phase forms. This question is being studied in detail by several techniques. Sections of irradiated fuel and cladding are heated at various times and temperatures to determine the onset of melting and the penetration kinetics over a range of variables, including different fuel and cladding combinations and steady-state irradiation temperature. These studies are complemented by tests where entire irradiated fuel pins will be heated in a furnace above the temperatures where a liquid phase forms between fuel and cladding.[10] Initial results with both types of tests have been extremely encouraging, showing no incompatibility at temperatures up to 800°C.

Phase II ORT Program

The principal objective of the Phase II ORT tests is to examine the performance capability of long-lived advanced cladding LMR pins under steady-state and operational transient conditions. The mixed-oxide pins will be clad with advanced austenitic steel and high strength ferritic steel, and reference size fuel with standard austenitic steel claddings will also be tested. The performance of the pins will be evaluated for slow transient overpower and transient overheating conditions for burnups up to 15 at.%.

The Phase II RBCB tests will build on the encouraging results of the Phase I program with reference size fuel pins and provide the requisite data to show that limited RBCB operation will be safe and feasible in LMRs with mixed-oxide pins in advanced cladding. Specific objectives will be to establish safe RBCB operating procedures based on reliable diagnostic information, excluding fuel center melting and failure propagation, and minimizing contamination of the primary circuit.

Source subassemblies are being used to preirradiate a variety of 7.5 mm pins to burnups of 5 at.% by early 1991. Variables include linear power, fuel stoichiometry, and fuel smear density. These pins will be predefected in the plenum region and reirradiated to establish the impact of RBCB operation on pin thermal and mechanical performance, including fuel center melting. Tests have already been performed to establish impact at zero burnup; it was found to be small. Another test is under irradiation to simulate the RBCB behavior of a pin bundle at high burnup with these large diameter pins.

These source pins then will be employed in transient overpower tests to be performed in 1992 and 1994 in EBR-II, and in whole pin transient overheating tests in a hot cell (beginning 1992). Both tests series will also contain reference size pins from the Phase I program TOP-4 tests which have been taken to 15 at.% burnup in a test extension in EBR-II.

Other tests will probe the characteristics of Delayed Neutron (DN) release, fuel loss mechanisms from breached pins, and breach propagation potential of advanced claddings. The fuel loss tests involve development of a special contamination capsule, and tests of a special triple-station DN monitoring system now being proved out of reactor.

DIAGNOSTIC SYSTEMS DEVELOPMENT: ENHANCING OPERABILITY

Current technology development, both at Argonne and elsewhere, includes efforts relating to the use of artificial intelligence, sensor/signal validation in many forms, pattern recognition, optimal control technologies, etc. The EBR-II effort is to identify needs, develop and/or adopt promising technologies, and integrate them into an operating power plant for proof of value. After they have proven useful at EBR-II, it is expected that they can be incorporated into advanced designs such as IFR and/or included in backfit activities.

Several projects are currently underway at EBR-II to assist the plant operators during normal operation and off-normal events. These include an expert system-based program called DISYS for plant and system diagnostics and control, advanced graphics displays as operator aids based on depiction of plant states as a Rankine cycle, and a pattern recognition system called the System State Analyzer (SSA).

DISYS and the System State Analyzer

DISYS[11] is an expert system developed in conjunction with Westinghouse and Pennsylvania State University which, on a real-time basis, evaluates sensor data, combines this data to determine component symptoms which in turn are evaluated to assess the state of components. The "health" of these components is then used to assess the "health" of subsystems, systems, and so on, up to the plant level. This sensor-driven logical hierarchy is seen to be the most appropriate structure upon which to provide real-time diagnostics and control guidance information. It is also very consistent with the engineering design, and compatible with the human conceptual structure of a complex plant.[12]

Because of the inherent uncertainties in the diagnostic process, and also because degrees of faultedness exist, fuzzy logic[13] is used to pass information up the hierarchical model. Each rule in the model has multiple input pairs and one output pair, each pair consisting of a numerical value for "degree of faultedness" and "credibility of diagnosis." In general, diagnosis can only take place within the context of a known system configuration. For example, the diagnosis of a valve in the "closed" position will differ from its diagnosis in the "open" position. For this reason the plant control signals are used in DISYS to determine plant configuration down to the component level. In addition to diagnostics, DISYS is also capable of recommending appropriate control actions to move the plant from an unsafe state to a functionally equivalent safe state. This feature has not yet been implemented at EBR-II. The DISYS system is being developed and tested on the EBR-II argon cooling system for fuel handling operations and the EBR-II steam plant.[14] A graphics based human interface system is also being developed and tested.

The SSA,[15] in different forms, is in various stages of development, testing, and application at EBR-II. The SSA has been developed specifically to be tolerant of instrumentation faults by being able to identify faulted instruments and provide a reasonably accurate estimated value for that instrument reading. This estimated value is based on previously learned pattern relationships with readings from other instruments that are similarly related to and define the process being monitored. The SSA has demonstrated a high degree of tolerance to sensor faults during plant tests and during normal plant operational surveillance. The SSA was also developed to be able to detect a change in the operational state of the process and provide information about the characteristics of the new state and what caused the state change. This can be a change to a different valid operational state or to a faulted state.

The SSA process begins with identification of system operational states over a learning period of system operation. The SSA then compares new observed data with the learned state patterns. The SSA then establishes an estimated state based on the similarities with previously learned states. The estimated state is "built" using a weighted combination of learned states, the weighting value being determined by the degree of pattern overlap with each learned state. The estimated state contains a new estimated value for every parameter being monitored, including estimated values for sensors that have degraded or failed, assuming sensor data was available during the learning process. Because the estimated signal values are based on actual established relationships with the values of all the sensor signals in the signal group representing the process, the failure or degradation of a sensor has an insignificant effect on the SSA estimated value for that sensor signal.

The SSA has been applied to EBR-II plant surveillance during startup, steady-state, and transient operation and as a monitor of core outlet temperature during steady-state operation.[16,17]

Display System Development

More effective display of plant data in nuclear plants has been of considerable concern, especially since the TMI accident. With the advent of the computer, CRTs have been proposed and used for display of system information. Advanced reactor concepts envision extensive use of CRTs. To date, display designs have usually not taken advantage of the capability of the computer. Significant strides in information presentation have been made at EBR-II based on computer concepts. The result is the first real-time model based display incorporating the complete plant process as a Rankine cycle. This promising work will result in the development and testing of real-time plant displays that incorporate thermodynamic models of the plant operation. Rather than merely using the computer as a data collection and display device at EBR-II, the computer is used to collect data, transform it into a real-time model of plant performance, and then to display it. This plant information display system has been fully tested and is discussed in Reference 18.

In addition, the coupling of display information with diagnostics is being pursued. It is likely that the alarm systems, as now used in nuclear plants, will disappear through integration of real-time diagnostics with the advanced graphical displays.

BENIGN RESPONSE TO CONTROLLER FAILURE TESTING

Even though the unprotected loss-of-flow and loss-of-heat sink tests have both been successfully run in EBR-II from 100% initial power,[19,20,21] there remains the question of the unprotected transient overpower (TOP). There are really three subcategories of TOP events for an LMR; the familiar control rod withdrawal, the primary pump run-up, and sudden increase in power demand in the balance of plant. Focusing on the rod withdrawal (rod insertion in EBR-II, with its fuel-bearing control rods), it is known that only about half of the power reactivity decrement (reactivity addition needed to go from hot critical to full power) could be inserted from initial full-power conditions without taking the driver fuel above currently approved EBR-II safety limit temperatures. This is only about 1/5 of the allowed worth of one control rod. As increasing amounts of reactivity would be added, there would be an increasing level of fuel damage.

The solution to this problem in an IFR is to limit the total worth of control rods by controlling power largely by some other means. This is the substance of the Plant Inherent Control Tests (PICT) to be discussed below.

Controlling power by other means requires the development of one or more control strategies. That is, the ability to conduct meaningful (limiting) rod withdrawal tests, as well as tests in the other two subcategories of unprotected TOPs, requires the development of a compatible control strategy.

There are two other critically important reasons for work on a control strategy. First, control must be carefully designed not to override inherent safety characteristics of a plant. We have encountered this problem in utilizing our automatic control rod drive system for EBR-II. Second, the control system must be designed to accommodate passively the malfunction of automatic controllers. Thus, preparing to run an unprotected TOP in EBR-II is a broad-based activity.

Quasi-Static Control Tests

The ability to quasi-statically control reactor power with changes in primary flow, and/or in a turbine load-following mode, were shown in the EBR-II 1987 tests.[22] Subsequent analysis has shown that power can be controlled with a control rod with limited reactivity in such a way to preserve the capability to passively shut down for a loss of flow without scram.

The reactor power change during the tests can be explained by considering changes in reactivity ($\delta\rho$), power (δP) power/flow ratio ($\delta(P/F)$), and reactor inlet temperature (δT_i). Reactivity changes due to fissile atom depletion are neglected as is control rod reactivity. A quasi-static (linearized) approximation for the reactivity perturbation can be expressed as:

$$\delta\rho = A\delta P + B\delta(P/F) + C\delta T_i \tag{1}$$

in which A is the power coefficient representing reactivity feedbacks proportional to power change alone, B is the coefficient representing the reactivity feedbacks proportional to the power-to-flow ratio (P/F) change, and C is the coefficient representing reactivity feedbacks related to reactor inlet temperature variation.

With the control-rod-drive mechanism deenergized, the net reactivity change from one-steady state to another would be zero ($\delta\rho=0$). For the tests in which reactor inlet temperature is kept constant, and where power is controlled with primary flow, then $\delta T_i=0$. By substituting $\delta T_i = \delta\rho = 0$ into Eq. (1), the relationship of P/F between two equilibrium states, 1 and 2, can be expressed as:

$$\frac{(P/F)_2}{(P/F)_1} = \frac{1 + (A/B)F_1}{1 + (A/B)F_2} \tag{2}$$

where subscripts 1 and 2 denote the steady-state conditions 1 and 2, respectively. The A/B ratio in EBR-II is estimated to be between 0.1 and 0.25 depending on reactor P/F, loading conditions, and the bowing reactivity components.

For the load following tests, the primary flow is kept constant (i.e., $F_1=F_2$), and the reactor power responds to changes in reactor inlet temperature. The relationship between power and reactor inlet temperature at two equilibrium states is:

$$P_2 - P_1 = C \times (T_{i1} - T_{i2})/(A + B) \tag{3}$$

where (A + B) is the approximate power reactivity decrement (PRD), i.e., the reactivity addition required to raise the power from zero power hot critical to 100% power at 100% flow. The PRD in EBR-II is about 0.28$ depending on loading conditions, and C is about 0.007$ per °C based on data gathered from reactor inlet temperature perturbation tests and LOHSWS tests in EBR-II. The final equilibrium conditions of the PICT tests can be estimated using Eqs. (1) to (3).

PICT 1 - Control of Reactor Power with Flow

The purpose of this test was to study the feasibility of controlling reactor power using primary flow. The primary pump speed was controlled to a prescribed speed by computer, and the secondary flow was regulated by the secondary EM pump through a secondary flow/tank temperature controller to maintain a constant reactor inlet temperature. The turbine admission valve controller was used to maintain a constant steam header pressure by adjusting the Turbine Admission Valve (TAV) position.

The initial reactor inlet temperature and turbine header pressure were controlled to the normal constant operating values of about 371°C and 8.7 MPa, respectively, and these values were controlled to remain essentially constant throughout the test. The reactor flow, the forcing function in this test, was reduced to about 42% at 1% per minute in three steps and then the flow was returned to 110% in a similar manner. The intermediate values were 77 and 58% flows. When the flow was reduced, the reactor temperature increased, which caused the reactor power to decrease due to negative reactivity feedback as explained by Eq. (2). The total change in power closely followed the flow decreasing to about 58% and then increasing to almost 100%.

In order to control the reactor inlet temperature, the secondary flow tended to follow the primary flow and power variation. It was noted during the transient that the reactor inlet temperature remained nearly constant as demanded, with a deviation of no greater than 2°C from the initial value. This deviation was somewhat reflected in the transient power response since the reactor power varies about 2.7% for every 1°C change. Although the reactor flow and the inlet temperature at the end of the test were very close to the initial conditions, the final power was about 3% lower than the initial power. This was caused by driver fuel burnup during the test period. After the test a calibrated control rod was moved to obtain the initial power and thereby to measure reactivity loss due to burnup. If the reactor power was controlled to be constant by varying the reactor flow, the burnup would be manifested as an increase in primary flow.

The results indicated that reactor power can be regulated using primary flow. However, if a precise transient reactor power profile is required, the secondary flow/tank temperature controller should be more precisely tuned if possible, such that reactor inlet temperature variation can be reduced during the power and flow maneuvers.

PICTs 3 and 4 - Load Following

Tests 3 and 4 demonstrated the slow reactor power change and load-following (reactor power follows the turbine-generator load demand) maneuvers involving reactor power, inlet temperature and turbine generator output demand changes. These plant disturbances, in turn, were controlled by the secondary pump and the turbine admission valve (TAV) position. The reactor power in these tests was maneuvered from 96% to about 50% and then back to about 96%. For both tests, the primary pump speed was controlled to maintain a constant reactor flow. In PICT 3, the demanded reactor inlet temperature setpoint was first set and the secondary flow/tank temperature controller thus responded by regulating the secondary flow. At the same time, the turbine admission valve controller was used to maintain a nearly constant steam header pressure by adjusting TAV position.

On the return to power, the TAV was controlled to attain desired electric output. The secondary pump was controlled to keep the steam header pressure constant. The results indicate that the power can be easily controlled by the reactor inlet temperature.

Future Tests

A series of tests are being planned or have been done to characterize EBR-II for control failures which could lead to overpower. Also, we are planning to dynamically test plant control methods which emphasize passive safety.

Lehto, Dean and Fryer[23] conducted primary pump run-up tests which showed that the increase in power due to increase in primary flow was acceptable. Primary flow was increased from 32% to 100% in 20 seconds from an initial power-to-flow ratio of 1.0. Power followed flow and leveled off at about 90%. Thus the final P/F ratio was less than 1.0 and core exit temperature was less than at the starting point. This was because of the negative fuel temperature coefficient due to the increased linear heat rating of the fuel. During the experiment, the secondary flow was conservatively controlled to keep the inlet temperature nearly constant. Lehto also showed by analysis that the power increase would be even less with a control strategy that allowed reactor inlet temperature to increase as a natural consequence of the increase in primary flow. Thus the transient overpower caused by primary pump runout has been shown by analysis and test to not be a safety problem for EBR-II. This conclusion is also true generally for metal fuel LMRs.

Tests are being planned that will examine how the plant limits the effects of large increases in the turbine load that could occur as a result of controller failures.

In the first of these, the Turbine Admission Valve (TAV) is assumed to fail wide open. The analysis shows that the steam pressure will decrease, the secondary cold leg sodium temperature will decrease, and ultimately this will cause a decrease in reactor inlet temperature and an increase in power. The capacity of the TAV and IHX however, limit the power increase to about 10%.

Another series of tests will investigate the passive safety characteristics of EBR-II which would limit reactor power during secondary flow runout event. The plant will be operated in the normal mode prior to the event -- the primary pumps providing constant flow, and the Turbine Admission Valve controlled to maintain constant steam pressure. The control rods will not be adjusted during the test. The secondary pump controller is assumed to fail and produce maximum flow. The pretest analysis indicates the plant "sees" the failure as an increased energy transport rate from the reactor to the steam generator. The secondary pump and heat transport system capacities limit the power increase to about 25%. The increase in temperature at the reactor exit is limited and not a problem because the transient is driven by a reactor inlet temperature decrease.

The last of the test series will address the traditional transient overpower caused by a control rod runout. The plant will be configured with the controllers in their normal lineup except the controlling rod will be operated with the automatic rod control rod drive system. Power maneuvers with the limited reactivity control rod will then be demonstrated. A failure of the control rod will be simulated by rapidly ramping the control rod to the end of its travel. The analysis shows the power will increase about 15% and stabilize if the other controllers act to remove the excess power. If other control schemes (such as steam load following) were used such that secondary flow did not increase to keep reactor inlet temperature constant, the power and temperature extreme would be even less.

Further testing of control methods which enhance passive safety are also being planned. In the near term, a dynamic version of the turbine load-following control scheme will be tested. In addition to dynamic validation of this control scheme, the test will also investigate the ability to passively compensate for fissile burnup with temperature adjustment and other operational aspects of this control scheme.

A modern control approach to EBR-II is being developed as a result of this testing and associated analysis. In the near term we are planning an "integral test" by applying the approach to control of the reactor inlet temperature. The problem is physically interesting since it involves the nonlinear behavior of the IHX and coolant stratification in the primary tank. On the other hand, it is simple enough to use and test control hardware and software interfaces in the plant environment.

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