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RESULTS AND IMPLICATIONS OF THE EBR-II
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ABSTRACT

On April 3, 1986 two milestone tests were conducted in EBR-II. The first test was a loss of flow without scram and the second was a loss of heat sink without scram. Both tests were initiated from 100% power and in both tests the reactor was shut down by natural processes, principally thermal expansion, without automatic scram, operator intervention or the help of special in-core devices. The temperature transients during the tests were mild, as predicted, and there was no damage to the core or reactor plant structures.

In a general sense, therefore, the tests plus supporting analysis demonstrated the feasibility of inherent passive shutdown for undercooling accidents in metal-fueled LMRs. The results provide a technical basis for future experiments in EBR-II to demonstrate inherent safety for overpower accidents and provide data for validation of computer codes used for design and safety analysis of inherently safe reactor plants.

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I INTRODUCTION

Experimental Breeder Reactor II has conducted a testing program during the last two years to investigate the capabilities of liquid metal reactors (LMRs) to perform vital safety functions inherently. In particular the tests have examined post shutdown decay heat removal by natural circulation and passive shutdown of reactor power after accidents which lead to undercooling of the reactor. The undercooling accidents have been divided into two categories - the loss of flow without scram (LOFWS) (a family of accidents involving a loss of forced flow through the reactor) and the loss of heat sink without scram (LOHSWS) (a family of accidents involving a loss of the ability to transfer reactor heat to down stream components which generate steam and electricity.

The testing program was completed on April 3, 1986 when LOFWS and LOHSWS tests were conducted sequentially, both from full power. The results of these tests demonstrate that LMR's can be extraordinarily and inherently safe. The purpose of this paper is to describe the tests, how they were conducted, the results, and some of their implications.

II REACTOR PLANT DESCRIPTION

EBR-II is a sodium cooled fast breeder reactor plant. Its design power is 62.5 MW thermal at which it generates about 20 MW of electrical power. The reactor, primary coolant pumps IHX and connecting piping are immersed in a large pool of sodium held in a large double-walled tank. The two centrifugal primary coolant pumps take a suction of sodium from the pool and deliver a combined flow to the reactor inlet. Each pump is driven by a variable-speed motor-generator set with control from 20% to 100% flow. A battery backed auxiliary electromagnetic pump is located in the pipe between the reactor and the IHX. This pump augments natural convection flow for post shutdown cooling. Sodium circulated by an electromagnetic pump in the secondary loop takes heat from the IHX and transports it to the steam generator where superheated steam is generator for the turbine generator.

The reactor is fueled with a uranium alloy. The fuel is a pin 0.3302 cm in diameter encased in type 316 stainless steel cladding with a thermal bond of sodium between the fuel and cladding. A regular fuel assembly is composed of 91 fuel pins. Control or instrumented assemblies are composed of 61 pins. The reactor is made up of these hexagonal assemblies, 5.817 cm across flats, located on 5.893 cm centers. The fueled (driver) region extends out through row 6 or 7, reflector assemblies extend through row 10 and the blanket region extends to row 16.

III TESTING BACKGROUND

Since 1974 EBR-II has conducted a sequence of thermal-hydraulic testing programs. The initial test programs were planned to support continued safe, flexible operation of EBR-II. More recently the tests have been planned to also prove feasible of inherent safety in advanced liquid metal reactors. The approach has been to closely integrate the development and validation of thermal-hydraulic-neutronic computer codes with plant testing and then analytically and experimentally investigate an ever widening variety of plant upset conditions. The success of the analytical/experimental programs has been due in large part to a series of instrumented, calibrated in-core fueled assemblies. These assemblies XX07, XX08 and XX09 measure in-core sodium temperatures and flow and thereby allow very good control of the experiments and timely, precise comparisons of measurements to calculations.

As described by Golden et al. [1], the initial test program studied steady state natural circulation phenomena. The natural convective flow rate and core temperature rise were measured as a function of power level and secondary flow rate. NATDEMO [2], the EBR-II system dynamic simulation code, was validated for steady state low flow conditions.

The next series of tests explored the transition to natural circulation from various reduced power conditions. The measured results of these tests were used to validate NATDEMO for transient natural circulation calculations. They were also used to develop the HOTCHAN code for calculating the detailed fuel assembly flows and temperatures during the transition to natural circulation.

The last series of tests has lasted over a two year period from 1984 to 1986 and has included natural convection tests, plant perturbation tests, loss of flow without scram tests and loss of heat sink without scram tests. Nineteen natural circulation tests as discussed by Planchon et al [3] were conducted from various at-power and shutdown conditions including a complete loss of forced flow from full power along with a reactor scram. The measurements showed temperatures were mild, less than the temperature limit for an anticipated transient. The measurements also confirmed thermal hydraulic models in NATDEMO.

Primary flow and reactor inlet temperature perturbation tests were conducted to validate reactivity feedback coefficient data and validate feedback models in the NATDEMO model as reported by Mohr and Chang [4]. These tests were run by perturbing flow ±30% (from an initial at-power condition) and allowing power to respond to the reactivity feedbacks. As expected, the reactivity feedback tended to match power to flow and minimize variations in sodium temperature at the reactor outlet.

The reactor inlet temperature was perturbed, in parallel a set of tests, by controlled changes in secondary flow rate or controlled changes in the steam drum temperature. These tests were used to validate the reactivity feedback coefficients and transient models which govern the reactor response for the loss of heat sink tests [4].

IV LOSS-OF-FLOW-WITHOUT-SCRAM-TESTS

Nineteen loss-of-flow-without-scram tests were run under many different conditions, as shown in Table 1. The first 6 tests (tests 27 through 32) were run in May 1985 from 16.7% power and 19% flow. Chang et al. [5] describe the tests and results. The sodium temperatures measured in-core agreed very well with pretest predictions for all the tests. As shown in the table, the test matrix involved variation of the pump coastdown time, condition of the secondary pump, and condition of the auxiliary pump. The agreement between measurement and prediction provided a firm technical basis for proceeding to tests from higher powers. The results also confirmed previous calculations which showed that pump coastdown is critically important in determining the peak transient temperatures during the loss of flow without scram. With this experience and confidence in the computer analysis we planned the remaining tests (tests 33 through 45) with two goals as follows: 1) Demonstrate a loss of flow without scram to natural circulation from 100% power while keeping peak reactor temperatures within limits for an "anticipated transient". Tests 33 through test 39 and test 42 were planned to meet this goal, 2) Demonstrate a passive shutdown in EBR-II for a loss of all AC power (station blackout) without scram. Tests 40 and 41 and tests 43, 44, and 45 were planned to meet this goal.

To achieve the first goal we developed the capability to electronically control the pump rundown and simulate longer pump coastdowns attainable in new plant designs with pumps and motors with larger inertia. In order to conduct a station blackout test, the pump drive controls were upgraded so that the motor generator and pump stayed connected during a coastdown. The extra kinetic energy provided by the M-G set extended the pump coastdown to about 95 to 100 sec. The modifications to the pump drive are described by Messick et al [6].

It was also necessary to extended the fuel temperature limits beyond the normal 715°C for an "anticipated" event in order to conduct the station black test. A new time-at-temperature correlation was developed based on test data from a series of furnace tests of Mark IIA fuel as explained by Lahm et al [7]. A special "hot-lead-driver XY-22", in-reactor test was then conducted to confirm the correlation from the furnace test data. XY-22 was tested in EBR-II at elevated, constant temperature until failure occurred. The failure was a clad breach in the above core plenum gas region and resulted in a benign release of fission gas. The test showed the time-at-temperature correlation was conservative and that the planned tests (tests 44 and 45) in Table I could be run without expected failure of any fuel elements or shortening of the scheduled fuel irradiation time. The extended limits were further confirmed by actual testing.

The sequences of steps which led to the final loss of flow without scram tests (tests 45 and 39) are as follows:

1) Establish 100% Power

The test were run from a steady state full power condition. The results were not sensitive to the decay power or time at power.

2) Place Auxiliary Pump in Correct Lineup

For test 45 the AC power supply was disconnected, transferring the pump to a limited capacity battery. This simulated a station blackout. For test 39 the auxiliary pump was turned off to simulate a complete loss of flow.

3) Insert Special Scram Protection

A special overtemperature scram function, using temperatures measured with the instrumented assembly XX09, was inserted to protect against additional equipment failures or errors during the tests. This provided "safety grade protection" for the reactor. For tests in which the margin between expected and limiting temperatures was lower, the fast acting, computer operated, pump coastdown test monitor (CTM) circuit was designed inserted. Had the pump coastdown been too fast this circuit would have automatically dropped a control rod to terminate the experiment and limited losses to fuel lifetime.

4) Bypass Loss of Flow Scrams

The normal scram protection functions for a loss of flow (based on measured flow and measured subassembly outlet temperature) were bypassed to allow the experiment to be conducted.

5) Turn Off the Pumps

The electrical power to the primary and secondary coolant pumps was turned off in test 45 thus allowing the pumps to coast down to a stop in about 95 sec. This simulated a station blackout. In test 39 the pump speed was automatically decreased on a speed vs. time curve simulating a hyperbolic coastdown with a pump stop time of 300 sec.

V LOSS OF HEAT SINK WITHOUT SCRAM TESTS

Loss of heat sink tests were run from 50% power and from 100% power. The loss of heat sink tests were designed to experimentally investigate the response of EBR-II to a large category of accidents that involve a loss of the normal means of transferring heat from the sodium pool to the balance of plant where electricity is generated. These accidents include failures of secondary loop pumps or their controllers; feedwater pumps, valves or their controllers; or other parts of the steam plant. Traditionally reactor plants are protected by automatic scrams and automatic controls against the overtemperature and overpressure that accompany a loss of heat sink. The tests were designed to show that these automatic protection systems were unnecessary for EBR-II.

The pretest analysis [8] showed that the loss of heat sink transient was very mild and slow compared to the loss of flow transient. Therefore, the tests were simple to conduct. Neither hardware changes to control the tests nor extensive bootstrapping to work up to the full power test was necessary.

The pretest analysis also showed that a simple worst-case experiment could be conducted to umbrella the many different loss of heat sink transients that could result from various types of equipment failures. The worst-case test was conducted by stopping secondary flow thus essentially blocking heat transfer from the primary pool to the balance of plant.

The operational steps to initiate the loss of heat sink tests were as follow:

1) Establish 100% Power

   The test was run from 100% power steady state condition. The results were not sensitive to the power history (decay power).

2) Block Automatic Initiation of the Shutdown Coolers

This prevented the normal opening of the airside dampers to the shutdown coolers when the temperature of the sodium pool exceeded 332°C.

3) Stop Flow in the Secondary Loop

The electromagnetic pump in the secondary loop was deenergized. Voltage to the pump was reversed to develop a reverse head to balance the thermal head and effectively stop flow in the secondary loop.

VI REACTOR SAFETY DURING TESTING

Reactor safety and the quality of the tests were assured by 1) detailed analysis of the tests and of accidents that could occur during testing, 2) deliberate testing operations carried out by a trained staff in strict adherence to proven step-by-step procedures. A bootstrapping analysis based on previous computer calculations and test results was carried out prior to conducting each succeeding test, and 3) engineered features to control critical aspects of testing and protect the reactor plant against overtemperature in the event of equipment failure or errors made during testing.

A. Analysis

The analysis for a test series generally started with a projected core loading an which a neutronic analysis was performed with the DOT-IV [9] code. The power for each assembly was computed from the fission and gamma heating rates produced by DOT. The power was used in the EBRFLOW [10] code together with flow pressure drop characteristic for each assembly to calculate steady-state temperatures and flows.

A transient analysis was done with NATDEMO [2] and HOTCHAN using assembly power and flow input from EBRFLOW. The analysis identified the hottest assembly in steady state and its peak transient temperature for a representative test transient and for upset transients during testing. Generally the assembly with the highest steady state power in each row was the row-wide hottest in steady-state and was also the hottest assembly for transients. This is because the flows for the drivers in a row are about the same. Transient analysis was necessary to pick among rows for the core-wide hottest assembly. Reactivity feedbacks were assumed not to vary outside a prescribed uncertainty band from previous measured values determined by Mohr and Chang [4]. This assumption was checked by test prior to the test window.

At this point each test in the sequence was planned and analyzed in detail. NATDEMO was used to calculate the temperatures, flows and power on a plant wide basis and HOTCHAN was used to predict transient driver, blanket and instrumented subassembly temperatures. The analysis showed that all the tests could be conducted well within limits for the most frequent transients--the anticipated event limit--except for LOFWS tests 44 and 45. These tests were conducted within the extended time-at-temperature limits previously discussed.

Considerable analysis was done to make sure the reactor was protected if there were equipment failures or mistakes during testing. We analyzed for the effect of the design basis loss of flow and transient overpower events occurring at the worst time during each test.

For the loss of heat sink tests there were no safety problems different from those during normal operation. This is because once the transient is initiated, power and peak reactor temperatures monotonically decrease. Thus, in a general sense, the reactor is always further from its thermal limits during the LOHSWS than it is at full power.

The analysis of the LOFWS tests showed that a reactivity insertion from a rod runout during the tests could lead to unacceptable over-temperatures. This is because the control rod could easily overpower the reactivity feedbacks which tend to inherently shut down the reactor. The normal protection for rod runout (the overpower scram and the subassembly outlet temperature scram) would not have been effective for a rod runout starting at low power and low flow. The high temperature scram based on XX09 measurements could have provided protection; however, it was decided to simplify test and eliminate possibility of a rod runout by deenergizing the control rod drive motors. Deenergizing the control rods did not disable the rapid control rod insertion following a scram.

The possibility and effects of rapid pump coastdowns were examined. A failure mode analysis showed that there was no likely common mode failure that would cause simultaneous rapid coastdown of both pumps except the loss of site power. The NATDEMO analysis of the rapid loss of flow showed that no scram was necessary for the low power tests. The reactor feedbacks reduced power to keep temperatures within limits. For the higher power tests, a reactor scram based on XX09 measured temperatures was necessary to keep temperatures within limits. The NATDEMO and HOTCHAN analysis was used to determine the setpoints for the XX09 scram.

Immediately prior to a testing period the DOT, EBRFLOW, NATDEMO, and HOTCHAN analysis were repeated for the actual core loading. A final set of pretest predictions for each test was then produced.

In summary, a detailed nuclear-thermal-hydraulic analysis was used to plan and predict the tests and show that safety limits would not be exceeded even if there were equipment failures or mistakes during the tests.

B. Testing Operations

The tests were conducted in a way to further assure plant safety during testing and to provide quality data in an efficient way. This was accomplished by 1) bootstrapping from less severe to more severe tests and 2) conducting tests with detailed procedures.

As previously discussed, testing progressed from natural circulation testing and perturbation testing to the LOFWS and LOHSWS tests. This allowed confirmation of models and data to support the subsequent high-power, high-risk tests. Just before testing started for the final LOFWS and LOHSWS, a set of tests was conducted to confirm key plant data that had been used in he pretest predictions. These included:

1) Power Reactivity Decrement (PRD) Test. During the plant startup, reactivity as a function of power was measured. The measurements were used to confirm that actual integral reactivity feedbacks from zero power to full power agreed with the data in NATDEMO which was used in the pretest predictions.
2) Primary Flow Perturbation Test. The reactor power and temperature response to a perturbation of primary flow was measured. The measured data was used to check that the actual reactivity feedbacks which are proportional to flow agreed with data in NATDEMO that had been used in the pretest predictions.
3) Inlet Temperature Perturbation Test. The reactor power and, reactor outlet temperature responses to inlet sodium tempera-

ture perturbation measured. The measurement was used to verify the actual reactivity feedbacks which make up the integral inlet temperature coefficient agree with data in NATDEMO that had been used in the pretest predictions.

4) Pump Coastdowns. Coastdown tests were run prior to the reactor startup for each LOFWS test. This verified the pump and controller characteristics.

After each test a comparison of measured and predicted temperatures was made. In some cases additional NATDEMO analyses were performed to assure that any differences were understood before proceeding to the next test.

All plant operations during the tests were conducted in strict conformance with comprehensive detailed test procedures. The procedures were written by analysis and operations personnel and reviewed and approved by all disciplines in the EBR-II organization. The procedures covered the normal steps involved in operating plant equipment as well as administrative controls that were included to assure that the test setup was consistent with plans and analysis and that the special instrumentation and controls were set up as intended. The procedures clearly delineated the test limits and the operator action should any test limits be exceeded. The procedures were proof-tested before the tests with in-plant walk throughs.

C. Engineered Safety and Test Control Features

A number of modifications were made to facilitate the tests and to protect the plant during testing [6] With respect to plant protection the philosophy was to provide as far as reasonably possible independent, automatic and/or inherent protection for equipment failures, analysis errors or operator mistakes. At the same time the protective equipment and procedures were kept as simple as possible to avoid introducing equipment failure or operator mistakes. The XX09 overtemperature scram function and inherent feedbacks provided this protection for the loss of flow tests. The XX09 trip was configured as a two-out-of-four trip that bypassed the normal loss of flow and subassembly outlet temperature trip functions and was activated by a key switch. The key switch assured that the normal loss of flow scrams would not be activated during normal operation. The XX09 trip was considered "safety grade" and met all our safety requirements. The normal subassembly outlet temperature, over-power and low-flow scram functions and reactor feedbacks provided protection during the loss-of-heat-sink tests. The reactor feedbacks reduced power and reactor temperatures so promptly during the LOHSWS tests that were at the normal scrams had to be bypassed.

A fast acting coastdown test monitor (CTM) was developed and used to supplement the XX09 temperature trip for loss of flow tests 40, 41, 43, 44, and 45. The CTM compared measured pump speed to desired pump speed during pump coastdowns and was programmed to terminate the test if a faster than desired coastdown had occurred. The CTM utilized the plant data computer and a control rod drop circuit which is normally used for rod-drop experiments. It was not considered to be safety grade protection. However, if the most probable equipment failures had occurred in one of the pumps during tests 44 or 45, the CTM would have sensed an off-normal pump coastdown, released a control rod and thereby limited fuel damage such that the tests could have been repeated.

A number of controllers including the pump/flow controllers were upgraded to support the plant tests. The tests could not have been done with the existing controllers or with manual control of pump coastdown. The upgraded flow controllers extended the pump coastdown, for station blackout to about 95 sec, to pump stop. This was done by keeping the pump and generator set connected during the coastdown and effectively utilizing the stored energy in the M-G set. The new controllers also allowed the control of pump speed to simulate a wide range of pump coastdowns in tests 33 through 39 and test 42.

VII TEST RESULTS

A. Loss of Flow Without Scram

The measured temperatures are shown plotted on predicted temperature curves in Figs. 1 and 2. Figure 1 is for test 39 (a LOFWS from 100% power with a 300 sec. coastdown time to pump stop and natural circulation). Figure 2 is for test 45 (a LOFWS from 95% power with a 100 sec coastdown time to pump stop and natural circulation with the auxiliary pump on battery). The maximum and minimum coolant temperatures were calculated considering nuclear, thermal and hydraulic uncertainties in XX09 as well as uncertainty in pump coastdown and reactivity feedback coefficients. The curve labeled "maximum hot driver clad" is the fuel-clad interface temperature of the hottest pin in the hottest fueled driver calculated with uncertainties. As shown the temperature measured near the top of the core in XX09 agrees well with the nominal predicted coolant temperature for XX09.

Further post-test analysis is necessary but as Mohr [11] discusses, there is reasonable confidence in the NATDEMO models used to predict the LOFWS. Figures 3 and 4 show measurements of temperature, reactivity, power and flow for tests 39 and 40 respectively. The measurements in these figures show the essentials of the passive shutdown. Flow coasts down resulting in an imbalance in the power to flow ratio. This causes the in-core sodium temperature transient. The difference between the P/F curve (the quotient of measured neutron flux and measured XX09 flow) and the ΔT ratio curve (the difference in sodium temperature at the top of core and core inlet as measured in XX09 normalized by their initial value) indicates the significance of heat capacitance, inter/intra assembly flow redistribution and heat transfer. Note the difference is smaller for the slower test 39 transient. The reactivity curve is the excess reactivity calculated from measured power with inverse neutron kinetics. It is normalized by the PRD. Note that the transient reactivity feedback is mostly proportional to the core ΔT.

The peak temperatures for the LOFWS transients were well below any safety limit imposed by the fuel or by sodium boiling. Comparing Figs. 1 and 2 further shows that lower peak transient temperatures can be obtained by lengthening the pump coastdown. The measured and predicted long term temperatures at the core exit tend to return to their original at-power values contrasted with approaching a high temperature asymptotic value. This is caused by the type of reactivity feedback in metal fuel which is largely proportional to the power to flow ratio or core sodium ΔT and weakly proportional to power alone. This proportionality is shown in Figs. 3 and 4. Analysis reported by Planchon et al [12] has shown this characteristic is typical of larger commercial sized LMRs that are fueled with metal fuel.

B. Loss of Heat Sink Without Scram

Temperature measurements and predictions for the LOFWS test from 100% power are shown in Fig. 5. A detailed discussion is given by Feldman et al [13]. The reactor inlet temperature rise from the sudden loss of heat rejection introduces negative reactivity from expansion of the grid support, lower reflector and other core materials. By the time the inlet temperature has risen about 42°C (75°F), the reactor power is shut down. The inlet temperature rise can be estimated with the quotient of the power reactivity decrement (0.30$) and the inlet temperature coefficient (0.0022$/°C (0.004$/°F)). It is significant that the temperature at the top of the core as measured in XX09 not overshoot -- it decreases from the full power value. This type of behavior - the low, long-term reactor outlet temperature and the lack of transient temperature overshoot are typical of metal fueled LMRs in which the high thermal conductivity of the fuel results in low Doppler reactivity feedback at power. Consequently the power-reactivity-decrement is low compared to the inlet temperature coefficient. As shown by Feldman [13] the reflector and blankets which normally operate at low power to flow ratio and hence at low sodium temperature rise can experience a small temperature increase from the inlet temperature increase.

VIII IMPLICATIONS AND CONCLUSIONS

The overriding implication of the tests is that a Liquid Metal Reactor plant can indeed be made to be inherently safe for severe undercooling accidents. The technical feasibility of passive shutdown and subsequent passive heat removal for LOFWS and LOHSWS was demonstrated. The passive shutdowns were achieved without automatic scram, operator intervention or special in-core devices.

Reactor plant features that are necessary for inherent safety were identified by the analysis and tests. The most important features are the reactivity feedbacks. It is important to have a PRD that is relatively small compared to the inlet temperature coefficient--this is the case in EBR-II. Thus a small reactor inlet temperature increase (less than 60°C) will reduce power to zero for loss of heat sink accidents. It is also important to have a PRD dominated by components proportional to the power-to-flow-ratio or core sodium ΔT as compared to power proportionality alone. Eighty five percent or more of EBR-II's PRD is proportional to P/F. As a consequence only a small increase in core ΔT is required to generate negative reactivity to balance the positive reactivity from power reduction. This results in low, long-term sodium temperatures for LOFWS.

These types of integral feedback coefficients are typical of metal-fueled reactors of all sizes, principally because of the high thermal conductivity, characteristic of metal fuel. This can result in low temperature fuel and a reactor with relatively small Doppler feedback. In contrast, the low thermal conductivity of oxide fuel can lead to higher fuel temperatures and significantly higher Doppler feedback--particularly in larger, reactors which have a softer neutron spectrum and consequently a large Doppler coefficient. As a result the PRD is large in these reactors and is dominated by terms proportional to power or fuel temperature. The temperature response to LOFWS and LOHSWS in a core fueled with uranium oxide therefore tends to be much less favorable than the response in a core fueled with metallic uranium.

The pump coastdown is a significant factor determining the peak temperatures for LOFWS. As shown by the EBR-II tests, a longer coastdown can be selected to keep peak temperatures lower than the temperature limits for even an anticipated transient. The pump coastdown is therefore a relatively independent, flexible, design parameter that can be selected bound peak full temperatures for LOFWS.

A previous objection to the inherent safety approach has been that one could not directly prove inherent safety by test for fear of serious damage to the reactor. The loss of flow and loss of heat sink tests showed that inherent safety capabilities can be demonstrated directly by testing. Further, the simple perturbation tests and simple analytical calculations were shown to be highly reliable indications of passive shutdown performance for the LOFWS and LOHSWS tests from full power. Thus the EBR-II tests indicate that perturbation tests can be easily conducted and that passive shutdown performance can be proved by test in new plant designs.

Finally, the tests also suggest the possibility of utilizing inherent feedbacks for passive control of reactor power and as a preferred method of protecting against expected equipment failures. Active scrams could then be used as a much simplified safety backup. This idea is explored by Planchon et al [12] and may lead to significantly simpler inherently safe reactor designs.

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TABLE 1. Summary of Loss of Flow Without Scram Tests

	Initial Power, % of Rated	Initial Primary Flow % of Rated	Primary Pump Coastdown Time (sec)	Secondary Pump	Auxiliary Pump	Predicted Peak Cladding Temperature of Fuel Assembly (with uncertainty) °
27	16.7	19	85	on	on	613
28	16.7	19	85	tripped	on	618
29	16.7	19	85	on	off	657
30	16.7	19	85	tripped	off	677
31	16.7	19	19	tripped	on	657
32	16.7	19	19	tripped	battery	679
33	50	100	300	400 s. coastdown	off	585
34	50	100	300	tripped	off	593
35	50	50	300	400 s. coastdown	off	625
36	50	50	300	tripped	off	625
37	100	100	600	tripped	off	604
38	100	100	300	400 s. coastdown	off	652
39	100	100	300	tripped	off	672
40	50	100	95	tripped	battery	635
41	50	100	95	tripped	off	622
42	50	100	200	tripped	off	676
43	70	100	95	tripped	battery	713
44	90	100	95	tripped	battery	774
45	100	100	95	tripped	battery	802