Passive Safety Testing at the Fast Flux Test Facility Relevant to New LMR Designs

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Abstract. Significant cost and safety improvements can be realized in advanced liquid metal reactor designs by emphasizing inherent or passive safety through crediting the beneficial reactivity feedbacks associated with core and structural movement. This passive safety approach was adopted for the Fast Flux Test Facility (FFTF), and an experimental program was conducted to characterize the structural reactivity feedback. The FFTF passive safety testing program was developed to examine how specific design elements influenced dynamic reactivity feedback in response to a reactivity input and to demonstrate the scalability of reactivity feedback results to reactors of current interest. Benchmarks based on empirical data gathered during operation of the FFTF as well as design documents and post-irradiation examination will aid in the validation of software packages and the models and calculations they produce. Evaluation of these actual test data could provide insight to improve analytical methods which may be used to support future licensing applications for LMRs.

Key Words: FFTF; Liquid Metal Fast Reactor; passive safety; benchmarks.

1. Introduction

Advanced sodium cooled fast reactors (SFRs) are one of the most promising Generation-IV reactor concepts for providing a safe, sustainable energy source, based on their inherent safety properties and more than 400 accumulated reactor years of operating experience with SFRs worldwide. Many current SFR designs already incorporate features to reduce the likelihood or to mitigate the effects of severe accidents and include passive safety features such as passive shutdown systems and natural convection decay heat removal systems. However, further innovations that enhance safety, reduce capital cost, and improve efficiency, reliability, and operability can make SFRs even more attractive as an option for electricity production. The accident at TEPCO's Fukushima Dai-ichi nuclear power plant has served to emphasize the importance of design measures that can prevent or mitigate the effects of unlikely severe accidents and extreme external events. A few examples of SFR projects under development include France's conceptual design of the ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) power reactor [1], Japan's innovative JSFR (Japan Sodium-cooled Fast Reactor) concept [2], and South Korea's prototype SFR [3]. Because these new demonstration reactors are seen as industrial scale demonstrations of SFR safety and operations, extrapolation of the major technical options and safety performance to follow-on reactors is very important.

In the mid-1980s the U.S. DOE conducted a passive safety test program at the 400 MWt Fast Flux Test Facility (FFTF) that included static tests to measure reactivity feedback between selected states in order to separate fuel and structural feedback components, conservative dynamic tests to demonstrate the transient behavior of reactivity feedbacks, and direct demonstration of safety enhancement features and their impact on upset events [4,5]. To the degree that FFTF static and dynamic test data can be separated into component feedback mechanisms, verification of models used in system codes like SASSYS and improved estimates of associated model uncertainties can be obtained. This was a key objective of the FFTF Passive Safety Testing Program. A specific reactor's test data can be applied to other reactors through the validation of design codes. A test of a design computer program is whether it can predict a real reactor. Codes like SASSYS are being used to analyze advanced SFR designs such as PRISM to verify their passive safety characteristics. The credibility of these system models can be greatly enhanced if they are shown to make reasonably good predictions of SFR reactivity feedback effects. The rationale driving the FFTF program was to provide a complete set of accurate static and dynamic measurements of reactivity feedbacks so that SFR system analysis code models can be confirmed. In order to support design decisions, fundamental experimental information is required that demonstrates the effectiveness of passive safety features and that can be used to verify the operation of system safety codes.

2. FFTF Description

The FFTF was the most recent Liquid Metal Reactor (LMR) to operate in the United States. The FFTF is located on the U.S. Government's Department of Energy Hanford Site near Richland, Washington. Conceptual design of the FFTF began in 1965, followed by a period of construction and acceptance testing that ended with first cycle operations in 1982. FFTF operations extended for a decade until it was shutdown in 1992.

The primary mission of the FFTF was to test full-size nuclear fuels and components typical of those to be found in a commercial liquid metal reactor. The fundamental objectives were that the reactor plant technology would support the liquid metal reactor industry by developing fuel assemblies, control rods, and other core components whose lifespans could be proven to be economical in commercial power-generating applications, and the reliability of the FFTF would be proven by matching or exceeding the operational performance of commercial light water plants. The FFTF did not have steam generators but included dump heat exchangers, and provided a prototypic test bed with respect to temperature, neutron flux level, and gamma ray spectra for fast reactor fuels and materials testing. It was the most extensively instrumented fast spectrum test reactor in the world, with proximity instrumentation of temperature and flow rate for each core component as well as contact instrumentation and gas and electrical connections for special test positions.

3. FFTF Passive Safety Testing Program

Prior to startup, the U.S. NRC reviewed the FFTF Final Safety Analysis Report (FSAR) but required tests to demonstrate the transition to natural convection circulation. These tests were performed at startup in 1980. With the reactor at 100% power and flow, the pumps were turned off and the control rods were scrammed. Special instrumented fuel open test assemblies (FOTA) were used to provide direct real-time measurements of temperatures of individual fuel pins at several axial levels to verify the natural circulation decay heat removal.

In the mid-1980's the FFTF conducted a passive safety test program that is summarized in TABLE I. This program included static tests to measure reactivity feedback between selected states in order to separate fuel and structural feedback components, conservative dynamic tests

to demonstrate transient behavior of reactivity feedback, and direct demonstration of safety enhancement features and their impact on upset events.

TABLE I: FFTF PASSIVE SAFETY TESTS

Passive Safety Test	Objective
Steady State Reactivity Feedback Tests	Separation of fuel and structural reactivity feedbacks
Delayed Pony Motor Trip Test	Verify transition to natural circulation performance and performance of fast response thermocouples in two assemblies
Steady State Natural Circulation tests	Demonstrate natural circulation performance
Controlled Flow Transient Test	Decrease in flow rate with no CR movement to confirm dynamic reactivity feedback models under loss of flow conditions
LOFWOS Tests with GEMs	50% Power and 100% Flow to Pony Motor Flow 50% Power and 100% Flow to Natural Circulation Flow
Inadvertent Pump Start with GEMs	Investigate potential accident scenario
Core Demonstration Experiment Tests	Characterize feedbacks with different fuel type

3.1.Phase 1 Reactivity Feedback Component Tests

During February and March of 1986 (Cycle 8A), a series of static tests was used to measure reactivity feedback between selected states in order to separate fuel and structural feedback components. These Phase 1 tests featured an extensive maneuvering through static conditions of prescribed changes in reactor power level, coolant flow, and coolant inlet temperature. A total of 198 static state conditions were measured by control rod position as the reactor power was varied between 10% and 100%, coolant flow rate was varied between 67% and 100%, and core inlet temperature was varied between 303 °C to 360 °C. Different feedback mechanisms were emphasized by changing core power, flow, and inlet temperature. The resulting changes in fuel and structure temperatures altered core reactivity, which was measured by compensating movement of a calibrated control rod. The magnitude of the associated reactivity between test states was determined by converting rod movements to reactivity. The test series was carefully designed to excite the different reactivity feedback modes, e.g. Doppler, axial fuel expansion, core radial expansion, duct bowing, sodium density, and differential control rod driveline expansion. All reactor plant conditions during the test series remained within reactor operational limits. State-point changes were grouped into seven types, depending on the combination of reactor operating parameters that were varied, as described in TABLE II.

In Type 1 tests, inlet temperature was controlled by adjustments to the heat removal rates of the secondary coolant loops. The outlet temperature was maintained by keeping the power-to-flow ratio constant. To first order, the temperatures of the coolant and structural materials did not change, and observed reactivity changes were attributed to changes in the temperature of the fuel material.

In type 2 tests the temperature of the fuel was held constant by keeping both the reactor power and axially averaged coolant temperature constant while both the flow rate and the core inlet temperature were increased. This emphasized subassembly bowing while minimizing temperature feedbacks.

In Type 3 tests, the flow rate was increased or decreased while the core outlet temperature was held constant. The fuel temperature was held constant by adjusting the reactor power and the

core outlet temperature was held constant to eliminate control rod driveline expansion and radial expansion at the top of the core.

In Type 4 tests, the inlet temperature was varied with constant power level and flow rate. All components in the reactor experience a uniform temperature increase. The major feedbacks come from uniform radial expansion.

In Type 5 tests, the coolant flow rate was varied while the reactor power and inlet temperature were held constant.

In Type 6 tests, static configurations were used to simulate the dynamic core response from a rapid reduction in coolant flow without any control rod movements. Fuel temperatures were changed to compensate for reactivity changes from coolant temperature changes.

In Type 7 tests the reactor power coefficient was measured for each series of state-point measurements as an overall indicator of reactivity feedback.

TABLE II: SEVEN TYPES OF STATIC FEEDBACK TESTS

Test	Parameters Held Constant	Parameters Varied	Major Contributor
1. Fuel Effects	Inlet Temp Outlet Temp Power to Flow	Power Flow	Fuel Temp
2. Structural Effects	Average Coolant Temp Power	Flow Power to Flow Inlet Temp	All Bowing
3. Structural Effects	Outlet Temp	Flow Inlet Temp Power	Core Support Expansion Bowing
4. Temp Coefficient	Power Flow	Inlet Temp	Radial Expansion
5. Flow Coefficient	Power Inlet Temp	Flow	Bowing
6. Controlled Transient	Control Rod Movement	Flow Power	All
7. Power Coefficient	Inlet Temp Flow	Power	Fuel Temp

FIG. 1 illustrates two of the types of tests, where both measurements effectively eliminate the fuel temperature feedbacks, but only the Type 2 measurement contains significant contributions from assembly bowing. These static tests confirmed the magnitude of the Doppler effect and better quantified the axial fuel expansion.

In June of 1986 a flow transient test was performed at FFTF to investigate dynamic reactivity feedback from a rapid decrease in core flow rate. This test confirmed the magnitude of the structural reactivity feedbacks under dynamic conditions. Control rods were not adjusted and reactivity changes due to the flow rate perturbations were compensated by changes in reactor power level and core inlet temperature. This test represented a mini low-of-flow transient. End-point results of the flow transient test were very consistent with the static reactivity data. Although this test represented a small perturbation of the reactor, it illustrates the basic understanding of LMRs.

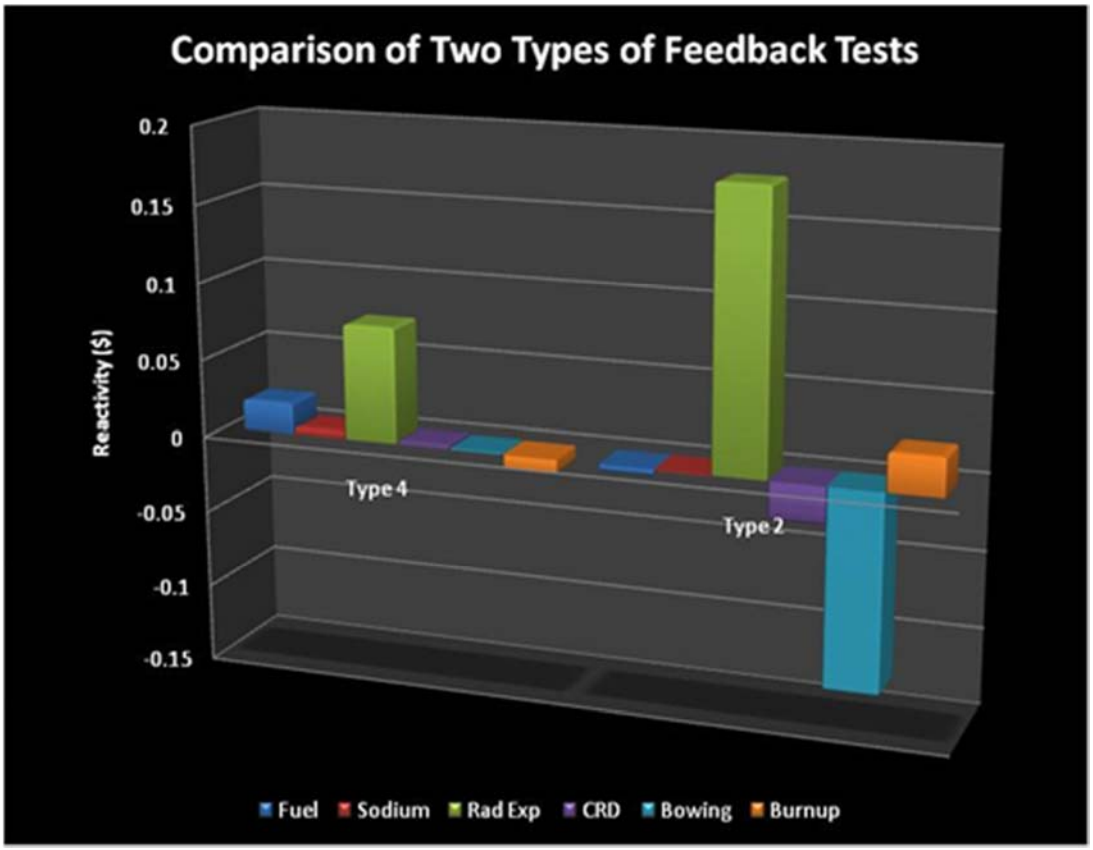

FIG. 1. Comparison of Two Types of Feedback Tests

3.2.Phase 2 Natural Circulation Cooling Tests

Two natural circulation cooling tests were performed to demonstrate inherent core cooling capability from a refueling condition where there is no thermal driving head and in steady state operating conditions. These tests were performed to verify adequate natural circulation performance at high heat load and to demonstrate the performance of special instrumentation. These tests provided data that can be used directly to validate systems analysis computer codes.

The Delayed Pony Motor Trip (DPMT) test demonstrated the ability of an LMR to transition to natural circulation from nearly isothermal conditions without experiencing excessive core temperatures. Before the test, there was some concern about the ability to predict the development of natural circulation driving head and flow from initially isothermal conditions. Two core positions, in Row 2 and Row 6, were provided with a special fast response thermocouple package. The thermocouple tips (five per subassembly) protruded into the sodium stream and provided a time constant of approximately three seconds.

The Steady State Natural Circulation Test established natural circulation through the core on decay heat and then the reactor went to power with only natural circulation cooling of the core. Such conditions generate different core temperature profiles than those encountered during normal operation because the flow in each subassembly is dictated by the heat being generated in it rather than the mechanical flow orificing at the core inlet. There was a concern whether the effects of these temperature distributions on the structural reactivity feedbacks were adequately predicted. The goal was to better understand natural circulation performance at high heat loads, since this is a potential final condition for a loss of flow without scram from high power. The cool-down from the highest power level was performed significantly faster than the

heat-up, resulting in a larger effect from the vessel. During this test, the reactivity feedback from fuel temperature changes was quite small. Thus the data confirm that nothing unusual can be expected from the structural feedbacks under natural circulation.

The major conclusions from these tests were:

- Measured overall system response generally agreed with predictions for both tests
- Modeling of radial heat transfer is required to adequately predict the response of the outer row of fuel assemblies
- Reactor feedback correlations developed from data near normal operating conditions reasonably predict the core nuclear response under natural circulation conditions
- There is a significant time lag in the thermal response of the outer regions of the reactor vessel. This effects the expansion of the reactor vessel and its associated reactivity feedback effect.
- Fast response analytic models were confirmed as acceptable.

3.3.Phase 3 LOFWOS Tests

The most dramatic series of passive safety tests were loss-of-flow-without-scram (LOFWOS) transients, starting from 10% to 50% power (40 MWt to 200 MWt), first to pony motor flow, and then with only natural circulation cooling. These tests demonstrated the effectiveness of special passive shutdown devices in mitigating LOFWOS events. The Gas Expansion Modules (GEM) were essentially hollow tubes sealed at the top, open on the bottom, and filled with sodium and cover gas, as shown in FIG. 2. During a loss-of-flow transient, the reduction in the inlet plenum pressure caused the cover gas to expand, driving the sodium down and out of the core region. This decreased the core reactivity by increasing radial neutron leakage and thus accelerated the decrease in reactor power. The result is that reactor power decreases more rapidly during a loss of flow transient with GEMs, as shown in FIG. 2.

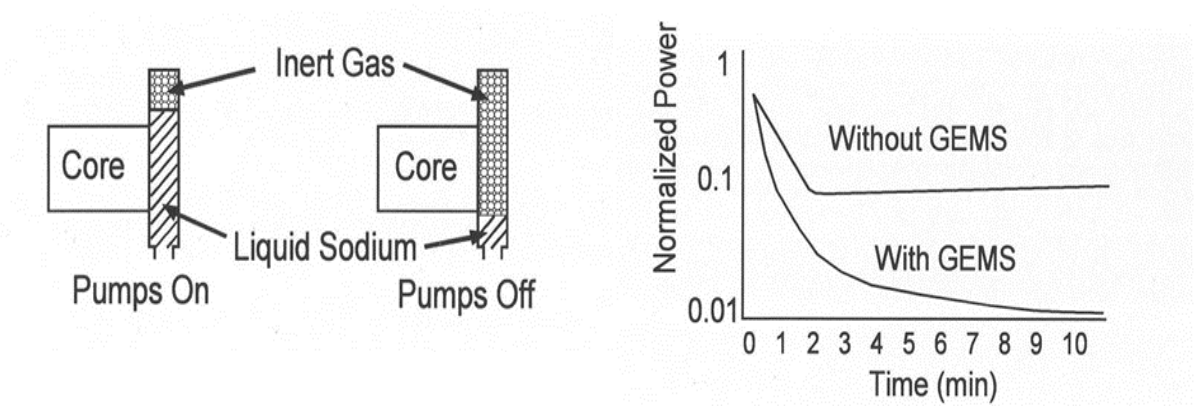

FIG. 2. Gas expansion module (GEM)

Estimates of the reactivity worth of these devices to justify their construction and subsequent tests were based on simulation in the Zero Power Plutonium Reactor (ZPPR). After they were installed in the FFTF, their worth was measured using the subcritical Modified Source Multiplication (MSM) technique. Measurements were made at two coolant temperatures because the volume of cover gas trapped in the top of the GEM devices (and therefore the initial starting position of the gas sodium interface) is determined by the temperature of the gas. A higher temperature lowers the interface and results in a more rapid reactivity insertion. Nine GEM devices were installed in the core periphery for the prototypic LOFWOS tests. Major

safety analysis and engineering packages were prepared to eliminate the automatic scram following primary pump trip, providing comparable levels of PPS protection as that specified in the FSAR for all test conditions. The first test series left the primary pump pony motors on throughout the transient so that the minimum flow reached was 9%. The reactor was taken to the target power level, the primary pump main motors were tripped, and the resulting thermal transient was observed for 15 minutes. The tests were run from 10, 20, 30, 40, and 50% of 400 MWt. The tests were then repeated with the same initial conditions, except the primary pony motors were left off, so that a transition was made directly to natural circulation flow in the primary system. Tests were run from 10, 20, 30, 40, and 50% power. A comparison was made of the peak power assembly outlet temperature for these two types of tests as measured by the fast response thermocouples. The first sharp peak about 10 seconds after the trip is associated with rapid initial flow coast down before GEM sodium level falls sufficiently to start inserting negative reactivity. Once the GEM starts inserting negative reactivity the power drops faster than flow and the core temperatures drop. As the GEM sodium level approaches the bottom of the core and the reactivity insertion slows, core temperatures begin to increase. The second broad peak is associated with the flow reaching a steady value while the power continues to slowly fall.

3.4.Phase 4 Core Demonstration Experiment Tests

Following these tests, in 1987 the FFTF core was reconfigured to the Core Demonstration Experiment (CDE), which consisted of replacing standard driver fuel assemblies with ten long-lived advanced fuel assemblies and six in-core blanket assemblies. The CDE driver assemblies were different from normal drivers in that they had axial blankets and the fuel pellets were not dished on the ends. Since the design of these fuel and blanket assemblies was significantly different from the standard driver fuel, with features such as larger diameter annular flat-end fuel pellets with axial blankets above and below the fuel column, the reactivity feedback response of the reactor was expected to change. Reactor tests to characterize the change in feedbacks were performed on the initial and subsequent startups. The power defect decreased significantly (12%) after the initial startup. Special steady state tests, like those performed for the previous passive safety test series, indicated a 19% decrease in the fuel temperature reactivity feedback. This was confirmed by a 20% decrease in the prompt reactivity feedbacks (fuel) obtained from MFBS tests. These test results are all consistent since the power defect includes contributions from reactivity feedback mechanisms other than fuel temperature. This could be explained if the axial expansion of the fuel was enhanced by the lack of fuel pellet dishes and then the expansion mechanism changed after the first major shock (shutdown) after full power was reached.

4. Potential Impact to the Design of New LMRs

Safety has always been paramount in LMR designs. Passive safety takes credit for intrinsic reactor materials and geometrical properties (neutronic, thermal, hydraulic, mechanical) which are completely passive in nature. The mechanisms leading to passive safety for initiating accident conditions are inherent in all sodium-cooled fast reactors, in that passive negative reactivity feedbacks like core axial and radial expansion, bowing, Doppler broadening, sodium density changes, control rod driveline expansions, and core support plate and vessel expansion will act to counter a rise in sodium temperature. The challenge is to incorporate passive safety into the design of advanced reactors such that the reactor will be brought to an acceptable power-to-flow match irrespective of the initiating accident condition. Plant economics can be improved by greatly reducing safety-grade active systems and plant safety is improved by relying on passive mechanisms instead of active engineered systems.

For example, a SFR can be designed so that during a loss-of-flow initiating event negative reactivity mechanisms such as enhanced neutron leakage can overcome positive reactivity associated with fuel cool-down to provide neutronic shutdown and a gradual transition to natural circulation at decay heat power levels. For transient overpower events, the available rod worth is limited to allow the negative reactivity feedback mechanisms to overcome the inserted reactivity and return the reactor to a power level matching the available heat sink. For a loss of heat sink initiator, the rising core inlet temperature triggers negative reactivity feedback (core expansion) to reduce the reactor power to decay heat levels. Passive decay heat removal systems can then use natural circulation to remove the decay heat.

Successful unprotected loss of flow tests were conducted from approximately half power (21 MWt) in the RAPSODIE reactor in 1983. Tests (both unprotected loss of flow and loss of heat sink) from full power (60 MWt) were successfully conducted in the EBR-II reactor in the spring of 1986. These experiments were the culmination of an extended series of tests, the SHRT series, begun in 1984. These were designed to evaluate the reactivity feedbacks and demonstrate shutdown heat removal capability and inherent safety characteristics of EBR-II and the IFR concept. Fundamental reactivity feedback tests were conducted in the FFTF in February 1986, and unprotected loss of flow tests up to 50% full power (200 MWt) were conducted in July 1986, which successfully verified the performance of GEM devices. The experimental results of the RAPSODIE, EBR-II, and FFTF unprotected loss of flow tests have been compared and show that whereas the primary coolant flow for the particular RAPSODIE and FFTF tests was allowed to drop to natural circulation conditions, flow for the EBR-II test was maintained at a minimum of about 4% (slightly higher than natural circulation) by battery power to the auxiliary pump. Running this test without the auxiliary pump was estimated to produce coolant temperatures about 28 °C higher. Other tests were conducted in EBR-II to natural circulation flow, but such tests employed longer flow coast-down times.

The French RAPSODIE reactor was powered by oxide fuel sufficiently enriched with ^{235}U that there was no appreciable Doppler effect. The principal negative reactivity feedback mechanism was sodium density change. The peak coolant temperature was turned around directly from negative reactivity feedback. The larger EBR-II reactor with metal fuel exhibited a similar transient response. In the EBR-II core, only passive feedbacks are acting to reduce the power and keep temperatures down to acceptable levels.

The much larger FFTF reactor with mixed oxide fuel demonstrated a qualitatively similar behavior from 50% power and full flow, but there was one major difference from the previous two reactors. The nine GEM devices provided the bulk of the initial negative reactivity feedback required to reduce the power. The first coolant temperature peak was due to the passive GEM reactivity insertion and the second peak was turned over by the onset of natural circulation, similar to the RAPSODIE and EBR-II experience. The result of all of these tests was that the peak coolant temperatures were several hundred degrees below the sodium boiling point. While the driver fuel for the FFTF passive safety tests was oxide, the structural reactivity feedbacks are independent of fuel type.

Part of the goal of the FFTF and EBR-II programs was to provide data to validate a code such as SASSYS over the range of interests for passive safety analysis. This equips designers with a high level of confidence in their predictions of safety margins for new designs.

Interest in using fast spectrum LMRs to effectively consume minor actinides introduces lower fuel melting temperatures, smaller delayed neutron fractions, and larger accident source terms, providing additional incentives to take advantage of passive reactivity feedbacks to reduce the consequences of accident scenarios. Incorporating passive characteristics into the design of new LMRs could reduce the requirement for safety grade (Class 1E) power to the point that it might

be supplied entirely from batteries. Continuous power is only required for the reactor protection system sensors, electronics, and monitoring displays, together with basic lighting and ventilation for the operators. Emphasizing passive safety in the early stages of LMR plant design, such as taking advantage of the inherent characteristics of LMR systems and including additional truly passive devices shifts the emphasis to accident prevention rather than accident accommodation. Designs incorporating such features appear practical and can potentially have a positive influence on design by simplifying the design, reducing costs, and increasing the safety of new advanced LMRs.

5. Conclusions

The results of the FFTF whole plant testing and demonstration of passive safety have broad implications for safety design and operation of advanced LMRs. For example, these tests have demonstrated the importance of negative feedback reactivity and magnitude of the Doppler coefficient, the importance of longer flow coast-down times of primary coolant pumps, and the need for detailed overall thermal-hydraulic design to enhance natural convective cooling. The historical FFTF passive safety testing program data supports the move towards greater reliance on passive safety in new LMR plant designs. The test data provides a very useful framework for testing advances in LMR safety technology and verifying plant safety codes, which should be of potential interest to the international fast reactor safety community.

Extrapolation of the major technical options and safety performance from test reactors and demonstration reactors to follow-on reactors is very important, and can be accomplished through the use of validated reactor safety codes. The data from the FFTF and EBR-II passive safety programs can be used to validate a code such as SASSYS over the range of interests for passive safety analysis. This can provide reactor designers with a high level of confidence in their predictions of safety margins for new designs. Economic gains can then potentially be realized through simplified designs and reduced uncertainties, which can translate to a reduction in vessel size for a specific power and reduction in design margins.

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