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ABSTRACT

Liquid metal cooled reactors (LMRs) have already been demonstrated to be robust machines. Many reactor designers now believe that it is possible to include in this technology sufficient passive safety that LMRs would be able to survive loss of flow, loss of heat sink, and transient overpower events, even if the plant protective system fails completely--and do so without damage to the core. Early whole-core testing in Rapsodie, EBR-II, and FFTF indicate such designs may be possible. The operational safety testing program in EBR-II is demonstrating benign response of the reactor to a full range of controls failures. But additional testing is needed if transient core structural response under major accident conditions is to be properly understood. The proposed international Phase IIB passive safety tests in FFTF, being designed with a particular emphasis on providing data to understand core bowing extremes, and further tests planned in EBR-II with processed IFR fuel should provide a substantial and unique database for validating the computer codes being used to simulate postulated accident conditions.

1. INTRODUCTION

The concept of employing passive safety for advanced nuclear power plants has advanced considerably within the past decade. Whereas reactor safety has always received particular scrutiny, the accidents at Three Mile Island and Chernobyl provided considerable incentive for reactor designers to come up with new systems that do not require the activation of any safety device to mitigate offnormal events. The goal of being able to rely on completely intrinsic, or passive, mechanisms to terminate any conceivable accident condition has clear appeal.

For this reason, essentially all of the present projects and future concepts discussed at this conference incorporate passive safety features to a greater degree than earlier designs. The liquid metal cooled reactor, however, is particularly well suited to exploit the concept of passive safety. The fundamental basis for this position is that sodium has an exceptionally high boiling point--such that liquid metal reactors (LMRs) normally operate at low (essentially atmospheric) pressure and with coolant temperatures several hundred degrees below the boiling point. This large coolant temperature differential between steady-state operation and the onset of coolant boiling provides a large thermal margin, which allows sufficient time for inherent negative reactivity feedback mechanisms to become manifest and counteract the upset condition. Such mechanisms include the following: (1) core radial and axial expansion, (2) bowing, (3) Doppler broadening, (4) sodium density changes, (5) control rod driveline expansions, and (6) longer term effects such as core support plate and vessel expansion.

In addition to exploiting inherent mechanisms to provide neutronic shutdown, it is also necessary to ensure the removal of decay heat by some intrinsic means, if passive safety is to be attained. This paper, however, is focused on the neutronic aspects (i.e., the reactivity feedback effects outlined above). The broader issues are dealt with elsewhere.[1,2]

2. EARLY EXPERIMENTAL PROGRAMS IN PASSIVE SAFETY

Among other configurations for which passive safety features are relevant, the unprotected loss-of-flow (ULOF) scenario has been the object of some experimental verification. The potential for safe response to ULOF events was first identified for the French Phenix reactor. The earliest direct experimental evidence to support the neutronic robustness of the LMR was provided in 1983 with the Rapsodie reactor at Cadarache in France.[3] There, the normal plant protective system was intentionally disabled and the reactor was subjected to an unprotected loss of flow accident from 50% of full power (namely, 21 MWt). Outlet coolant temperatures abruptly rose during the transient, but were automatically reversed as the core heated up (mainly because of the strong negative sodium temperature coefficient).

A series of experiments was conducted in the EBR-II reactor at Idaho Falls (USA) in 1986, culminating in intentional ULOF and loss of heat sink events from full power (60 MWt).[4] A temperature transient somewhat similar to the Rapsodie experience was observed, although the time scales were shorter because of the smaller Doppler effect (which introduces positive reactivity during a cooldown transient), and peak temperatures were lower (EBR-II tests at 50% power resulted in peak temperatures consistently lower).

Better use of the inherent characteristics of reactors to ensure safety in response to undercooling transients is simply good design practice. The difficulty is often associated with adequately predicting the behavior of systems in response to upsets, particularly when complicated thermohydraulic, mechanical interactions and reactivity feedback are involved. This is where the value of test programs in the LMR facilities is most apparent.

The EBR-II program of testing evolved over the course of 12 years, with an early emphasis on characterizing the behavior of natural convection flow of sodium under steady state conditions.[5,6] The tests yielded a great deal of information and were conducted under actual decay heat conditions and at low reactor power levels. The most important issue addressed was the flow behavior in parallel channels, each with a different heat rating and flow resistance. Structural subassemblies were contained within the core in addition to fuel and blanket subassemblies. This complicated configuration represented a major challenge for analysis but also provided a "rich" experimental mix with which to conduct testing. The testing and subsequent analysis quantified for the first time the strong intersubassembly flow redistribution that occurs during natural circulation. Good instrumentation is a necessary element in such a program and considerable effort was directed toward providing good in-core instrumentation. Fueled and non-fueled subassemblies instrumented for coolant flow and temperature were installed in selected control-rod positions.[5] It was found that it was also necessary to carefully consider coolant flow patterns within the subassembly to obtain reliable results for temperature measurements. Mixing of sodium within the subassembly can be an important factor for temperatures actually obtained. It was this early work of careful characterization of flow and temperature patterns in the core that laid the basis for the more aggressive tests that followed. The data generated from these tests validated codes that supported natural circulation testing in FFTF and evaluation of Clinch River Breeder Reactor Project (CRBRP) natural circulation. Codes that were validated using data from these tests include the whole core/assembly code COBRA-WC and the hot channel code FORE-2M.

Fortunately, as stated earlier, in conditions of natural convection cooling, the core tends to become more isothermal across any given elevation and modeling becomes more straightforward. Individual subassembly flows adjust according to the thermal "head", with those producing less heat developing less flow. Widely differing subassemblies tend to develop similar temperature rises. Also, temperature distributions within subassemblies tend to flatten, with flow and mixing profiles changing significantly from full forced flow conditions. Peaking factors are less. The extensive testing and model development at EBR-II for steady-state conditions set the stage for the next challenge, which was modeling the transition from forced to natural convective flow.

Uncertainties in the thermal-hydraulic behavior of sodium in the transition from forced to natural convective cooling prompted the EBR-II designers to install a small electromagnetic pump on the reactor outlet pipe to facilitate the transition. During the transition, flow patterns change in the core and it was postulated that some conditions could result in flow stagnation or even reversal in isolated regions of the core producing little heat.[7] To test the hypothesis, an instrumented structural subassembly was included in the core (along with an instrumented fueled subassembly) and a series of tests was conducted. It was confirmed that under some conditions, flow reversal would occur in the radial blanket/reflector region during a natural circulation transient. The data from these tests were important for validating the three-dimensional code COMMIX-1A. The most important result of these tests was that the transition to natural convective cooling was smooth and predictable.

The stage was then set for testing the response to actual loss-of-flow events. To adequately predict the response to loss-of-forced flow without scram, it was necessary to add one more ingredient; namely, detailed modeling of the change in reactivity feedback coefficients. The physics of reactivity feedback response is relatively straightforward. What is more complicated is the thermal-mechanical response of the core that gives rise to reactivity changes. Negative feedback effects in EBR-II are dominated by expansion of the sodium coolant. Also important, however, is expansion of the core grid, expansion of control rod drives, expansion of the fuel element and effects from subassembly bowing. EBR-II has a relatively small Doppler coefficient, a consequence of low fuel centerline temperatures with metal fuel. This is a major advantage in designing a core to accommodate loss-of-flow without scram, since there is considerably less positive reactivity to overcome as the reactor power and fuel centerline temperatures drop.

Characterization of thermal feedback effects under a variety of flow and temperature conditions was, therefore, a major part of the preparations for full-scale testing. This was accomplished in a series of tests conducted involving changes in power and flow to characterize individual contributors to reactivity feedback.[8] This series of tests has also been beneficial to developing techniques for monitoring reactivity feedbacks in the course of operation. Principally, these involve small changes in reactor inlet temperature, power-to-flow ratio and power at constant power-to-flow ratio.

Full scale testing began when this series of characterization of feedbacks from the core were complete. Tests in preparation of the loss-of-flow without scram and loss-of-heat-sink without scram tests had spanned a period of more than 12 years. Over 40 individual tests were conducted starting in 1984, leading to full power tests conducted on April 3, 1986.

For the loss-of-flow without scram test, an additional reactor trip was added, measuring in-core temperatures and set to trip the reactor if predicted values were exceeded. Full power was established, the normal trip parameters for loss-of-flow were bypassed, and power to the primary pumps was removed. Reactor core temperatures rose by about 200 °C and then returned to near normal as the reactivity feedbacks reduced power to match the level of natural convective flow.[9,10] Response of the reactor was as predicted and demonstrated that an LMR can accommodate an anticipated transient without scram (ATWS) event without core damage. There was no damage to the core and the fuel subsequently operated normally without cladding breach or loss-of-lifetime. As evidenced by the approximately 40 tests that were conducted before those conducted from full power, the range of upsets that can be accommodated passively is surprisingly large. The EBR-II testing program has systematically addressed failure of each of the major controllers (e.g., pump speeds increasing to maximum, the turbine admission valve full open, etc.). It has been demonstrated that the plant "self protects" against any single controlled failure and most double failures. These are very encouraging results and hold significant promise for follow-on LMR designs.

These results are probably best reflected in the PRA conducted for EBR-II. The probability for core melt is extremely low. Perhaps just as important, the potential for operator actions causing damage to the core, either as acts of omission or commission, is extremely low. This is reflected in the relative simplicity of emergency actions and training required for the operators.

If these advantages can be realized in large LMRs, as analysis indicates can be achieved, many aspects of plant operation can be simplified. Basically, it comes down to the fact that while it is simpler for the designer to engineer systems to provide active systems for protection, the results of effort spent refining characterization of plant response through development of analytic models and testing pays significant dividends.

Although testing at the Rapsodie and EBR-II reactors had demonstrated the beneficial effect of reactivity feedback mechanisms in a liquid metal fast reactor in a holistic sense, little experimental information existed concerning how specific design elements influenced dynamic reactivity feedback in response to a reactivity input. Additionally, the scalability of reactivity feedback results from smaller cores like Rapsodie and EBR-II to reactor cores that were more prototypic in scale to reactors of current interest had not been demonstrated. The Fast Flux Test Facility (FFTF) passive safety testing (PST) program was developed to examine these two areas.

Phase I[11] of the FFTF passive safety tests was conducted at the Hanford Site (Richland, Washington) in 1986, with the most aggressive unprotected loss of flow test being conducted from 50% of rated power (i.e., 200 MWt). For these tests, however, a group of special passive safety devices called gas expansion modules (GEMs) were employed to provide rapid negative reactivity feedback upon loss of pump pressure.[12] Additionally, an extensive set of static experiments was conducted wherein power, coolant flow, and inlet coolant temperature were independently varied–in an attempt to separate reactivity feedback effects resulting from changes in fuel or coolant temperatures.

Measuring FFTF core reactivity effects at power conditions caused by changes in fuel temperature and core geometry required precision difference measurements. Core conditions had to be carefully controlled and reactivity measurements had to be made to an accuracy of 0.1¢ or 3×10^{-4} $\Delta k/k$. Additionally, corrections for burnup depletion had to be used when comparing reactivity states that were measured more than a few hours apart. The basic premise for the feedback measurement test program was that simple manipulation of key reactor observables (power, flow, inlet temperature) could be done in a way that would maximize the effect of one feedback driver, such as fuel temperature, while minimizing the contribution of the remaining drivers. Fig. 1 contains the results of the most aggressive of the passive safety tests from the three reactors just discussed. Note that in each case, the outlet coolant temperature was well below the sodium boiling point.

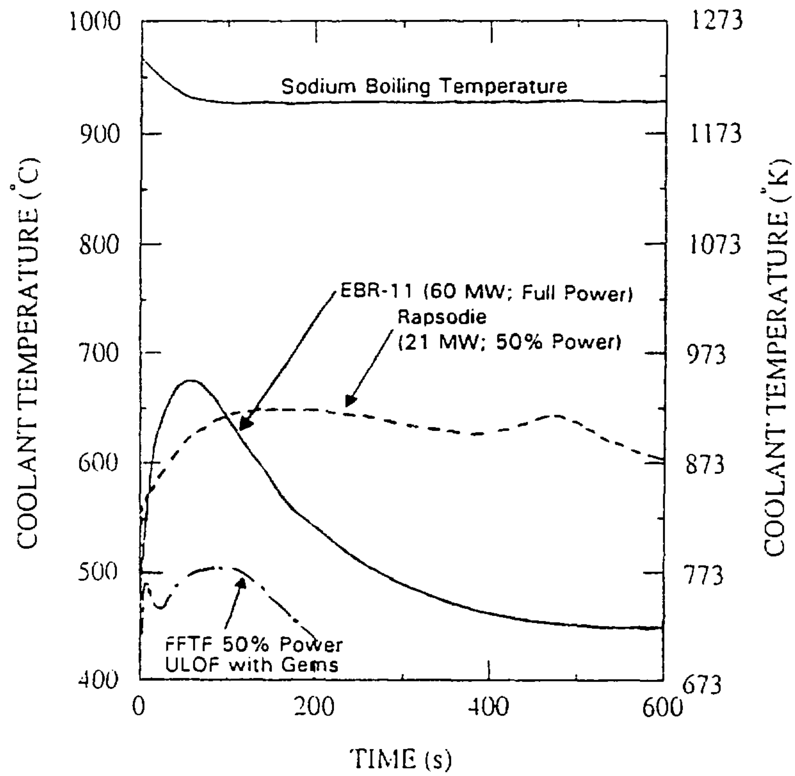

Fig. 1. Unprotected whole-core loss-of-flow transients–early reactor data.

The experimental results to date have been both impressive and encouraging. Despite the considerable design work done on a variety of reactor concepts, the LMR is the only reactor system that has been able to accumulate specific experimental data to demonstrate such robust transient performance. Nonetheless, it has become clear among the fast reactor safety community that more experimental data would be highly desirable if new LMR plants are to be licensed based on advantageous use of their passive safety characteristics. Principal reasons for this interest include the following: (1) knowing the margins available (to account for hot channel factors and other uncertainties), (2) reliability and reproducibility of the features that are being counted on for reactor shutdown, and (3) assurance that there are no situations in which such features could actually aggravate an accident condition. For the latter two reasons, the Phase IIA tests were recently conducted in FFTF.[13]

3. EXPERIMENTAL AND DESIGN INCENTIVES FOR PASSIVE SAFETY

Passive safety features must be predictable and reliable. A detailed knowledge of reactivity feedback effects for a broad spectrum of offnormal conditions is required to design a passively safe reactor. Feedback mechanisms must work together to mitigate the initiating transient and restore core reactivity to zero. The reactivity characteristics of a reactor core can change with refueling, burnup, and changes in structural components behavior caused by the buildup of neutron fluence. Because passive safety depends on a balance of fuel temperature reactivity feedback and core expansion reactivity feedback driven by coolant temperature, the time response of reactivity feedback is also an important factor in ensuring an effective passive safety response. Practical passively safe reactors can be realized only if designers can demonstrate that variability in reactivity feedback magnitude and dynamic response are within predictable limits and can be controlled. Licensing of such design features will require surveillance demonstration by analytic and/or experimental means. These are real challenges for reactor designers who wish to capture the safety margins and robustness of a passively safe reactor.

In the United States, recent LMR design efforts have been focused principally on relatively small, modular reactor configurations which incorporate metal fuel as the reference fuel system (with oxide retained as a backup). Such designs appear to incorporate a substantial margin of safety, even for low-probability events where it is postulated that the plant protective system fails completely. Calculations to date are impressive. It now appears that such reactors could sustain unprotected loss of flow, loss of heat sink, and overpower accidents without serious damage to the core. This is a level of safety not even dreamed of a decade ago. The results are so impressive that there is a legitimate question of whether classic containment is necessary.

From a regulatory viewpoint, the question focuses on how accurately the computer models are able to simulate the transient responses that such low-probability accident conditions would cause to occur. Designers and regulators need to know what uncertainty levels exist in the models that provide the predictions for core response to such large-scale initiating events. Uncertainties in analysis can be reduced by confirmation testing in a prototypic plant, such as was done for natural circulation in FFTF.

Perhaps the most difficult reactivity feedback to understand (and properly model) that is important to passive safety is duct bowing. Early calculations for standard liquid metal fast breeder reactors (LMFBRs) indicated there may be negative reactivity feedback from bowing sufficient that these reactors could survive ULOF accident from full power without reactor damage. Further analysis, combined with limited experimental information, revealed that the bowing geometry under such conditions would yield considerably less negative feedback than originally envisioned. To have a fundamental mechanistic understanding of core expansion and bowing reactivity for core conditions that go beyond conditions of normal plant operations it is necessary to demonstrate to regulatory authorities that the core maintains the desired feedback characteristics, reload after reload, without having to resort to a frequent and costly safety test program that challenges design limits. To enable designers to make confident and licensable claims for passive safety in new designs, there is strong incentive to have analysts subject their codes to rigorous pretest analyses, and to see how well the models hold up once the experiments are actually conducted.

The concept of passive safety is gaining increasing attention at all international levels. In Japan, for example, the development of fast breeder reactors (FBR) has made steady progress. Following the operation of the experimental fast reactor, Joyo, and the prototype FBR, Monju, a stage has been reached where promotion of a demonstration plant (Demo-1 plant) will lead to the acquisition, improvement and demonstration of commercial plant technology and the viability of cost and performance without sacrificing the safety and reliability of FBRs. The reactor shutdown system (RSS) in the current Demo-1 plant design is composed of two fast-acting independent main and backup reactor shutdown systems. To exclude common-cause failure of the RSS, diversity has been introduced in the detectors for the following: (1) scram signals, (2) control rod insertion mechanisms, and (3) the types of control rods. For some control rods of the backup system, the adoption of the self-actuated shutdown system (SASS) and enhanced thermal expansion mechanism of control rod drivelines are being studied to ensure the diversity of the RSS by enhancing passive safety features.[14]

At the same time, it is important to establish long-term development programs for the practical use of FBRs. The Atomic Energy Commission of Japan has been drawing up the new version of the Long-Term Program for Development and Utilization of Nuclear Energy by revising the previous program published in 1987, based on the long-range implementation of the nuclear energy policy towards the 21st century.

The Nuclear Safety Research Association (NSRA) has organized a technical committee on FBR safety, which has studied and discussed the fundamental philosophy of safety design and evaluation of future commercialized FBRs.[15]

As a result of the technical discussions of the technical committee on FBR safety, enhancement of passive safety features (PSF) as preventive measures against core disruptive accidents was recommended to be given highest priority for the promotion of commercial FBRs. Steps recommended include the following:

- Reduction in the excess reactivity of control rods by achieving a core design with smaller burnup reactivity change.

- Reduction in the sodium void coefficient.

- Enhancement of negative reactivity feedbacks such as the following: (1) fuel axial expansion, (2) fuel subassembly bowing/flowering, and (3) thermal expansion of rod drivelines.

- Enhancement of the RSS with higher diversity by employing PSF such as the SASS.

- Enhancement of the decay heat removal systems (DHRS) with higher diversity by employing PSF such as natural circulation/convection.

In connection with the above discussions, the Power Reactor and Nuclear Fuel Development Corporation (PNC) and the Japan Atomic Power Company (JAPC) have been conducting design studies based on using passive safety characteristics of both the RSS and the DHRS.

In the design studies conducted by PNC and JAPC, a large-scale mixed-oxide core is being optimized from the viewpoint of the enhancement of PSF by varying the linear heat rate of fuel, core geometry, and core configuration. As an alternative option, PNC has been conducting a core design study based on mixed-nitride fuel. Since the sodium-bonded nitride fuel has a high thermal conductivity and a very high melting temperature, it can be expected to enhance PSF without sacrificing the excellent FBR core performance. In the United States, the emphasis is now on metal core development, which has the advantage of strong passive safety features and excellent breeding characteristics. These various programs allow the full range of options to be explored.

Among the above Japanese design studies, special focus is being given to the negative reactivity effect of core radial expansion as a key candidate for the enhancement PSF in the RSS. Other key features include the enhancement of thermal expansion of control rod drivelines and an extended flow coastdown.[16]

Therefore, a strong incentive exists in Japan to develop an experimental database and analytical models as key parts of a long-term R&D program leading to designs that include a detailed knowledge of reactivity feedback effects of core expansion and bowing for a broad spectrum of offnormal conditions. Among these passive safety tests, ULOF tests are given first priority in Japan because the ULOF accident has been selected as representative of a beyond design basis event (BDBE) for future FBRs pertaining to PSA studies and other safety assessments for Japanese FBR plants.

Because the FFTF has a relatively large core of about a 1,040 L in volume, which is composed of mixed-oxide fuel and excellent in-core instruments, data from the proposed FFTF test Phase IIB are expected to be especially valuable for use in the development and validation of analytical models on reactivity effects of core radial expansion and bowing.

4. FFTF PHASE IIB

The Phase IIB series of passive safety tests proposed for the FFTF represents a first step in addressing the correctness of mechanistic computer models for the prediction of bowing reactivity feedback during an upset event. The design of these tests, discussed in subsequent papers in this session, is the result of a joint international collaboration effort of engineers from Europe, Japan, and the United States. It provides a unique opportunity to measure feedback data that emphasizes the bowing effect.

By way of perspective, the reactivity feedback separation test program concluded in the 1986 Phase I tests in FFTF measured roughly 200 state points over a wide range of reactor conditions. Evaluations of the results have been discussed elsewhere.[17] The critical implication derived from these tests is that reactivity feedback from core structural interaction is the most difficult effect to predict and needs to be experimentally verified at power-to-flow ratio (P/F) values greater than 1.0. "Flowering" cores have received considerable attention as a way to introduce negative reactivity feedback during accident events where P/F exceeds 1.0. Consequently, the importance of core structural interactions in accident mitigation is the central theme of the Phase IIB FFTF PST experiments. A series of measurements designed to obtain bowing reactivity feedback at P/F greater than 1.0 has been an area of particular focus. These tests will be done at low core power to minimize interference from fuel temperature feedback effects and to allow the core axial temperature rise to exceed nominal operating conditions, while maintaining primary system temperatures within safety bounds.

A key feature of the test series is to conduct one complete set of experiments with the core in a "barrel" shape, and then rotate the outer row of subassemblies 180 degrees and repeat the experiments for the new "wheat-sheave" shape. This approach would appear to maximize differences in bowing feedback characteristics and provide an excellent set of data for validation of the structural models.

Dynamic tests will also be performed as an integral part of the test series. These are described in a companion paper at this conference.[18] The dynamic experiments will confirm the use of static data to predict core feedback under accident conditions and are expected to be valuable in helping validate design codes.

Initial pretest calculations at Westinghouse Hanford Company (Westinghouse Hanford) indicate that ULOF tests (without GEMs) can likely be conducted up to a power level of at least 30%, without subjecting the core to unacceptable temperature transients. In order to gain a perspective of how such temperature transitions might compare to the earlier Rapsodie, EBR-II, and FFTF experiments, Fig. 2 replicates Fig. 1 but contains, as an overlay, calculated temperature traces for ULOFs starting at 10% and 50% of full power, respectively.

These calculations were based on neutronic calculations with 3DB and PERT3D to obtain power and reactivity worth distributions, and the SASSYS neutronics, thermal-hydraulics code to compute the transient response.

In Japan, PNC has been conducting the pretest analyses of the proposed FFTF Phase IIB program. The analyses comprise the following: (1) neutronic calculations with CITATION to obtain the reactivity worth and power distribution in the core, (2) static thermohydraulic calculations with TETRAS which generate the duct wall temperature profile as a function of power and flow, (3) static core mechanics analysis with BEACON (three-dimensional beam model code) and FINAS (three-dimensional finite element analysis code) to evaluate the thermal deformation of the core structure, (4) reactivity calculations of the core deformation such as the subassembly bowing with PERKY (two-dimensional perturbation code), and (5) plant and core dynamics analyses with SSC-L and SAS4A. The SSC-L has been upgraded so as to incorporate multidimensional thermo-hydraulic models, inter-subassembly (S/A) heat and mass

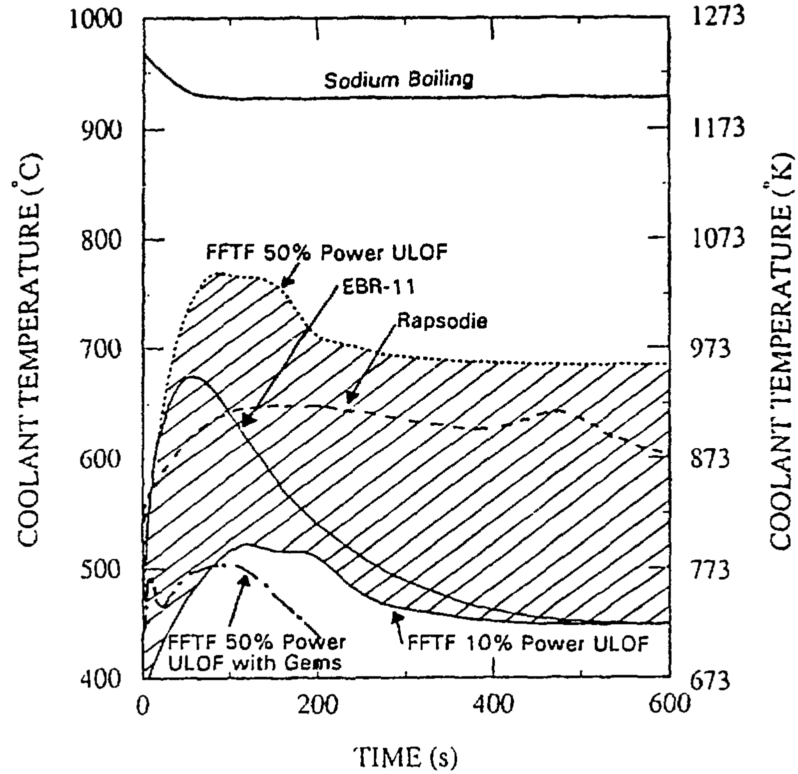

Fig. 2. Expected FFTF Phase IIB limits overlaid on earlier reactor test data.

redistribution models, variable fuel-clad gap models, and reactivity feedback models on control rod thermal expansion and S/A bowing. It was confirmed that these analytical tools are applicable to all aspects of the Phase IIB program.

Based on results of the pretest analysis by PNC, the following insights were obtained[19]:

1. For core bowing analysis, the results of the three-dimensional model (BEACON) were compared with the results of single S/A model in SAS4A code. It was confirmed that the single S/A model with an adequate internal (effective) gap between S/As can reasonably estimate the core bowing, although a single S/A model has some applicable limits.

2. Three-dimensional core bowing analysis confirmed that the initial conditions of gaps between S/A pads at the above core load pad yoke (ACLP) or the top load pad yoke (TLP) produce strong effects on the core bowing models. This leads to the conclusion that a parametric survey on the effect of the initial core (permanent) deformation should be conducted. Therefore, the proposed FFTF Phase IIB test plan that includes both a core with wheat-sheave-shaped S/As and a core with barrel-shaped S/As as an initial condition of core configuration is strongly supported.

3. The results of pretest analysis indicate that the supporting mechanisms for the S/As' inlet nozzles at the core support plate strongly affect core bowing. The degree of core bowing is suppressed if the condition is changed from free support to fixed support. However, the results indicate that the ULOF test in the proposed FFTF Phase IIB test plan has the potential to produce reactivity changes of about 20 to 30¢, whose magnitude corresponds to meaningful data, even if the support condition of the inlet nozzle of S/As is assumed to be fixed at the core support plate.

4. To extract reactivity effects due to core bowing from other reactivity effects with a high degree of accuracy from measured data, it is essential to suppress reactivity effects except core bowing in the tests and to evaluate all the reactivity effects separately with a high degree of accuracy. As the uncertainty of the reactivity effects from fuel expansion is considered to be relatively large, the pretest analysis justifies the initial power under ULOF conditions in the FFTF Phase IIB test to be set below 30% rated power. From the same viewpoint, it is also concluded that the proposed large flow transient tests are important and are expected to reduce uncertainties of reactivity effects except core bowing.

5. The pretest analysis also indicates the value that could be added by an in-core instrumented fuels open test assembly (FOTA). Measurements on fuel temperature and flow coastdown by the FOTA with a high degree of accuracy would help develop more elaborate analytical models on reactivity effects due to fuel-clad interactions and the Doppler effect.

It was originally envisioned that the FFTF Phase IIB test series would be concluded during the end of calendar year 1992. Detailed planning for the test has been delayed, however, because of a March 1992 decision by the United States Department of Energy to place the reactor in a standby mode pending the determination of a long-term mission. Consequently, it now appears that the test series could be delayed at least a year. Given the international consensus supporting the value which such a test program could render, however, there is mounting technical support for the program to proceed, even if somewhat delayed.

5. CONCLUSIONS

Among the various types of new nuclear power concepts being proposed for the generation of electricity in the coming decades, LMRs look exceptionally attractive, particularly with regard to passive safety. It now appears possible to design LMR electrical generating plants that are capable of surviving the most aggressive types of accidents that can be envisioned, and to do so even if the plant protective system fails completely.

In order to use advantageously the intrinsic capabilities of this technology, however, it is important to have an experimental database available to validate the computational models used to analyze such accident conditions. Despite the favorable and encouraging results from the earlier Rapsodie, EBR-II, and FFTF whole-core experiments, some of the prevailing reactivity feedback effects that come into play during such postulated scenarios are not yet fully understood. Core bowing is perhaps the most difficult phenomena to model.

Consequently, the proposed Phase IIB PST program for FFTF is focused heavily on providing data to validate transient computer models in this area. The pretest analyses conducted to date indicate that a test series can be devised to provide extremely useful data for model validation. The international program, now underway, has provided an excellent framework for identifying weak points in the current modeling capabilities and searching for ways to design a safety test program to provide needed experimental data. Gathering data from two fundamentally different core configurations, namely a barrel-shaped core and then a wheat-shaped core (obtained by rotating the outer row of subassemblies by 180 degrees), is expected to yield extensive unique and useful data to modelers now struggling with ways to properly model transient core structural effects.

The biggest uncertainty in the FFTF program at present is the schedule itself. Originally charted for a testing period of three to four months during late CY 1992, it now appears that the test series will be delayed at least a year. Because of the redirection in the priorities in the United States energy program, It is possible the FFTF will not be available to carry out the proposed tests. On the other hand, it is also possible that even greater emphasis might be placed on this test series and it could be expanded to a test window of six months or more. If so, this would allow the inclusion of even more international analysts to participate in the design of the tests to optimize the benefit to advanced LMR systems, world-wide.

Ideally, a successful PST program in FFTF would be followed by similar programs in other reactors. Such diversity would considerably enhance the variety of designs that could be confidently analyzed for major accident conditions. But even if it is not possible to continue with a Phase IIB program in FFTF, the work to date indicates that additional whole-core experiments should be carried out somewhere. The potential for developing truly robust machines, which can be well characterized and understood, is simply too great to pass up.

REFERENCES

1. R. J. Neuhold, J. F. Marchaterre, and A. E. Waltar, "A New Safety Approach in the Design of Fast Reactors," *Nuclear Technology*, Vol. 89, pp. 83-91 (January 1990).

2. R. P. Omberg, D. M. Lucoff, A. Padilla, and A. E. Waltar, "The Role of LMRs for Next Generation Power Reactors: A Strategy for Sensible Change," *Proceedings of International Topical Meeting on Safety of Next Generation Power Reactors*, Vol. I, pp. 89-95 (May 1988).

3. C. Essig, B. Berthet, E. Mascuso, E. Gesi, A. Gouriou, P. Bergeonneau, V. Ertel, A. Ertel, A. Vasile, A. Reetz, and A. Donati, "Dynamic Behavior of Rapsodie in Exceptional Transient Experiments," *Proc. Int. Topl. Mtg. Fast Reactor Safety*, Knoxville, Tennessee, April 21-25, 1985, p. 635, Oak Ridge National Laboratory (1985).

4. J. I. Sackett, W. K. Lehto, H. P. Planchon, and G. H. Golden, "Inherent Safety and Operability Testing in EBR-II," *Proceedings of International Topical Meeting on Safety of Next Generation Power Reactors*, Vol. I, pp. 385-393 (May 1988).

5. G. H. Golden, H. P. Planchon, J. I. Sackett, and R. M. Singer, "Evolution of Thermal-Hydraulic Testing in EBR-II," Nuclear Engineering and Design 101, pp. 3-12 (1987).

6. R. M. Singer, et al., "Steady State Natural Circulation Performance of the EBR-II Primary Heat Transport System," Nuclear Science and Engineering 63, pp. 75-82 (1977).

7. W. L. Bauman, et al., "EBR-II In-Vessel Natural-Circulation Analysis," ANL-82-66, NUREG/CR-2821 (1982).

8. D. Mohr and L. K. Chang, "Perturbations of Reactor Flow and Inlet Temperature for EBR-II Reactivity-Feedback Validation," Prac. of Intl. Topical Meeting on Fast Reactor Safety, 2, pp. 617-627 (April 21-25, 1985).

9. D. Mohr, et al., "Loss-of-Primary-Flow-Without-Scram Tests: Pretest Predictions and Preliminary Results," Nuclear Engineering and Design 161, pp. 45-56 (1987).

10. E. E. Feldman, et al., "EBR-II Unprotected Loss-of-Heat-Sink Predictions and Preliminary Tests Results," Nuclear Engineering and Design 101, pp. 57-66 (1987).

11. D. M. Lucoff, "Passive Safety Testing at the Fast Flux Test Facility," *Nuclear Technology*, Vol. 88, pp. 21-29 (October 1989).

12. J. B. Waldo, A. Padilla, Jr., D. H. Nguyen, and S. W. Claybrook, "Application of the GEM Shutdown Device to the FFTF Reactor," *Trans. Am. Nucl. Soc.*, Vol. 53, p. 312 (1986).

13. P. K. Shen, R. A. Harris, L. R. Campbell, W. A. Dautel, and A. E. Dubberley, "The United States Advanced Liquid Metal Reactor and the Fast Flux Test Facility Phase IIA Passive Safety Tests," *Proceedings of International Conference on Design and Safety of Advanced Nuclear Power Plants*, Tokyo, Japan (October 1992).

14. M. Miura, et al., "Safety Design Consideration for Japanese DFBR," *Proceedings of the International Conference on Fast Reactors and Related Fuel Cycles* (October 1991).

15. K. Aizawa, "Safety Design and Evaluation Policy for Future FBRs in Japan," *Proceedings of the International Conference on Fast Reactors and Related Fuel Cycles* (October 1991).

16. K. Sato, et al., "A Study of Enhanced Safety Systems for LMFBRs," *Proceedings of International Conference on Design and Safety of Advanced Nuclear Power Plants*, Tokyo, Japan (October 1992).

17. B. J. Knutson, R. A. Harris, D. H. Nguyen, and R. P. Omberg, "Fast Flux Test Facility Passive Safety Reactivity Feedback Measurements," WHC-SA-0205-FP, Westinghouse Hanford Company, Richland, Washington, (March 1988).

18. F. J. Heard, R. A. Harris, and A. Padilla, Jr. "Pretest Predictions of the Fast Flux Test Facility Passive Safety Test Phase IIB Transients Using United States Derived Computer Codes and Methods," *Proceedings of the International Conference on Design and Safety of Advanced Nuclear Power Plants*, Tokyo, Japan (October 1992).

19. A. Yamaguchi, et al., "PNC's Analysis of Passive Safety Test Phase IIB in the Fast Flux Test Facility," *Proceedings of the International Conference on Design and Safety of Advanced Nuclear Power Plants*, Tokyo, Japan (October 1992).