HEDL-SA-3702

# FIVE YEARS OPERATING EXPERIENCE AT THE FAST FLUX TEST FACILITY

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April 1, 1987

American Nuclear Society
International Conference On Fast
Breeder Systems: Experience Gained
and Path to Economical Power Generation
Sept 13-17, 1987
Richland, WA.





**HANFORD ENGINEERING DEVELOPMENT LABORATORY** — Operated by the Westinghouse Hanford Company, P.O. Box 1970, Richland, WA, a subsidiary of the Westinghouse Electric Corporation, under U.S. Department of Energy Contract No. DE-AC06-76FF02170.

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ABSTRACT

The Fast Flux Test Facility (FFTF) is a 400 Mw(t), loop-type, sodium-cooled, fast neutron reactor. It is operated by the Westinghouse Hanford Company for the United States Department of Energy at Richland, Washington. The FFTF is a multipurpose test reactor used to irradiate fuels and materials for programs such as Liquid Metal Reactor (LMR) research, fusion research, space power systems, isotope production and international research. FFTF is also used for testing concepts to be used in Advanced Reactors which will be designed to maximize passive safety features and not require complex shutdown systems to assure safe shutdown and heat removal. The FFTF also provides experience in the operation and maintenance of a reactor having prototypic components and systems typical of large LMR (LMFBR) power plants.

FFTF achieved criticality in February 1980 and first achieved 100% power in December of that year. In April 1982, the FFTF began its first 100-day irradiation cycle. Beginning in Cycle 5 in June, 1984, the cycle lengths have gradually increased to a 138-day operating period during Cycle 9A in 1987. Since the start of operation, FFTF has operated exceptionally well, continuing to set new performance marks. Highlights include achieving a 71% Capacity Factor in 1985 as well as an average Operational Efficiency Factor of 96.7% from 1983 through 1986. Two hundred and forty-seven fuel assemblies (36 of which are experiments) and over 52,000 individual fuel pins have been irradiated through Cycle 9A, some in excess of 155,000 MWd/MT Metal burnup. Specialized equipment and systems unique to sodium-cooled reactor plants have performed very well.

## INTRODUCTION (System Description)

The FFTF (Figure 1) is a 400 Mw(th) sodium-cooled fast neutron flux test reactor located on the United States government-owned Hanford Reservation in southeastern Washington state.

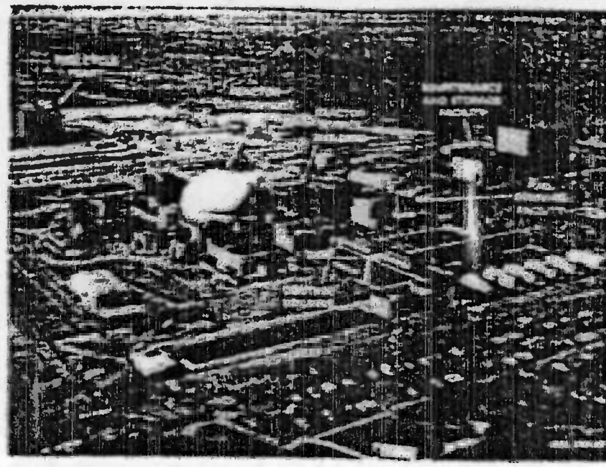

Figure 1. Fast Flux Test Facility.

The reactor is operated for the United States Department of Energy by the Westinghouse Hanford Company.

This three-loop plant was designed for testing fuels and materials in a prototypic thermal and neutronic environment. The neutron energy spectrum is typical of oxide-fueled fast reactors. Nickel reflectors surround the core to intensify the neutron flux, thus permitting the plant to rapidly attain high burnup levels for component irradiation. (The peak burnup goal of 80,000 megawatt days per metric ton of metal (MWd/MTM) for the reference driver fuel, for instance, was achieved in only three operating cycles.) The plant's ability to rapidly achieve high burnup levels is complemented by the ability to rapidly obtain preliminary test data from irradiated core components in the Interim Examination and Maintenance Cell, an in-plant facility.

The reactor is located in a shielded cell in the center of the containment building (Figure 2). Heat is removed from the reactor by over 477 $m^3$ (125,000 gallons) of liquid sodium which circulates under low pressure through three primary coolant loops. Three

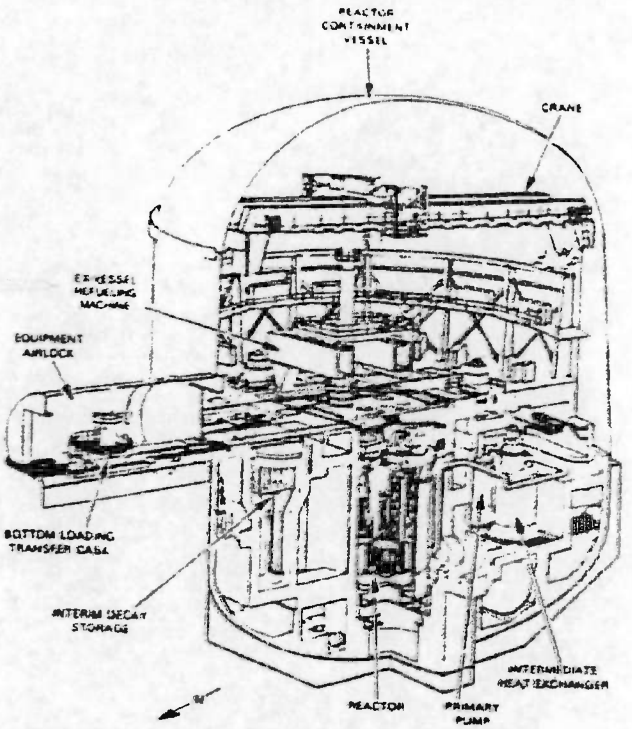


Figure 2. Reactor Containment Building.

intermediate heat exchangers separate radioactive sodium in the primary system from non-radioactive sodium in the 68 $m^3$ (18,000 gallon) secondary systems. Secondary sodium loops transport the reactor heat from the intermediate heat exchangers to the air-cooled tubes of the twelve dump heat exchangers.

Table 1 provides some significant technical design/operating parameters of FFTF.

The reference driver fuel load was comprised of about 16,000 pins of co-precipitated U-Pu mixed oxide fuel of 22.5 and 27 percent Pu. Each fuel subassembly contained 217 pins spaced by wire wrap and identified by a unique Xe-Kr gas mixture to identify possible cladding failure. Fuel cladding was Type 316 stainless steel. Refueling is only performed when the reactor is shutdown.

Adjacent to the plant is a simulator of the FFTF control room which includes the FFTF Main Heat Transport System (HTS), Reactor, and associated controls, indications and alarms. The Operations Training Simulator (OTS) provides simulation which is close enough to the predicted plant responses to allow procedures to be developed for plant control and to conduct effective training for the operators before actual testing or operation occurs. Tests have included conditions outside the normal operating conditions. The

TABLE 1

FFTF TECHNICAL PARAMETERS

| | |
|---|---|
| • THERMAL POWER | 400 MW |
| • REACTOR VESSEL INLET TEMPERATURE | 360°C (680°F) |
| • REACTOR VESSEL OUTLET TEMPERATURE | 503°C (938°F) |
| • NOMINAL CORE $\Delta T$ | 167°C (300°F) |
| • PRIMARY SODIUM FLOW CAPABILITY | 2744 l/s (43,500 gpm) |
| • PRIMARY SODIUM DYNAMIC HEAD | 152.4 m (500 ft) |
| • CORE DIAMETER | 121.92 cm (4 ft) |
| • CORE HEIGHT | 91.44 cm (3 ft) |
| • PEAK FAST FLUX | $7 \times 10^{15}$ n/cm$^2$-sec |
| • NORMAL HEAT REMOVAL | SODIUM-TO-AIR HEAT EXCHANGERS |
| • BACKUP HEAT REMOVAL | NATURAL CIRCULATION |

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OTS also supports the continuing training and recertification of qualified operators, supervisors and managers.

This paper reports on FFTF operating experience during the first five years of power operation. Overall plant performance is discussed as well as experience with those systems and components that are unique to sodium-cooled reactors.

FFTF OPERATING HISTORY

Startup and initial operation of the FFTF was conducted in conformance with a rigorous Acceptance Test Program (ATP). A comprehensive series of nuclear and non-nuclear tests were conducted to verify the thermal, hydraulic and neutronic characteristics of the reactor. Of special interest were the natural circulation tests conducted to demonstrate an inherent safety feature of the FFTF. These tests were completed very successfully and have been reported previously.[1] These tests were conducted again in the summer of 1986 as part of a series of Passive Safety Tests which included testing of a new device called a Gas Expansion Module (GEM). The natural circulation test was again very successful and is reported in a separate paper.[2] Cycle 1 operation began on April 16, 1982, marking the end of facility acceptance testing and the beginning of cyclic operations. Initially 100-day cycles were the norm, but for cycle five and beyond, gradually increasing cycle run times were achieved. Cycle 9A achieved 137.7 Effective Full Power Days (EFPD). Operation of the plant has been highlighted by ever-increasing performance capability.

Table 2 shows the cumulative statistics by cycle through the first 5 years of operation. Loading plans have been devised which permit 150 EFPD irradiation periods in subsequent cycles. Capacity Factor (Figure 3), a measure of the plant operation at 100% power, steadily increased annually from 56.9% in 1983, the first full year of operation, to 71% in 1985. Due to the many startups and shutdowns as well as operation at low power levels for Passive Safety Testing, the capacity factor was only 46.2% during 1986. However, 1987 will see a return to higher capacity factors with the annual goal again targeted at 70%. Availability Factor (Figure 4) likewise increased annually from 61.1% in 1983 to 73% in 1985. Again, due to the Passive Safety Testing conducted in 1986, availability fell to 56.8% but is projected to be greater than 70% during 1987. But perhaps the best measure of the FFTF's operational excellence is the Operational Efficiency Factor (Figure 5). This measure of how well the operations are conducted against a pre-approved plan has averaged 96.6 from January 1983 through December 1986. During this period, only 3.4% forced outage time occurred.

Perhaps the most interesting observation made during the first 102-EFPD of operation was that of an increasing pressure drop across the reactor core. Although this presented no limitation to a second operating cycle, additional data gathering and evaluation were required. The increase in pressure drop across the reactor core continued but at a much lower rate during Cycle 2. Core sub-

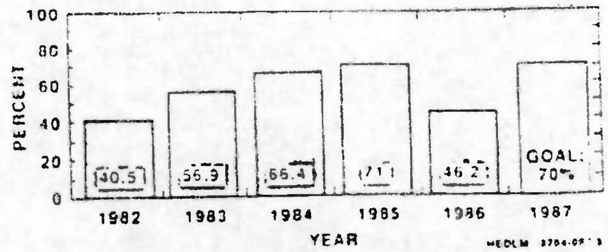


Figure 3. FFTF Capacity Factor.

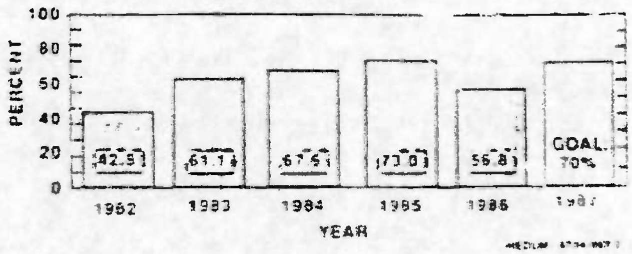


Figure 4. FFTF Availability Factor.

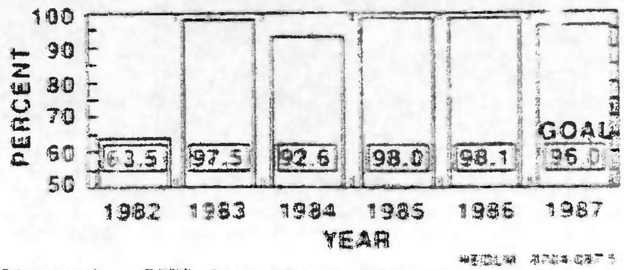


Figure 5. FFTF Operational Efficiency Factor.

assembly changes and midcycle refueling outages, combined with the lower rate of increase observed during Cycle 2, resulted in an end-of-cycle (EOC) value of 153 psi. This compared with 161 psi at the end of Cycle 1. Analytical evaluations and testing performed

TABLE 2

OPERATING STASTISTICS BY CYCLE

| | 1 | 2 | 3 | 4 | 5 | 6 | 7 | 8 | 8A |
|---|---|---|---|---|---|---|---|---|---|
| EFPD FOR CYCLE | 101.5 | 100.5 | 101.5 | 109.5 | 122.7 | 134.0 | 122.8 | 63.0 | 137.7 |
| TOTAL PLANT EFPD AT EOC | 134.5 | 234.8 | 336.3 | 445.8 | 568.5 | 702.5 | 825.3 | 888.3 | 1026.0 |
| CAPACITY FACTOR (%): | 50.3 | 83.1 | 93.5 | 99.5 | 93.5 | 74.3 | 90.3 | 39.5 | 93.2 |
| AVAILABILITY FACTOR (%): | 53.0 | 90.5 | 99.0 | 100.0 | 94.6 | 78.5 | 94.6 | 58.1 | 98.0 |
| MAXIMUM FUEL BURNUP AT EOC MWd/MTM) | 35,000 | 60,000 | 81,000 | 105,000 | 129,000 | 135,700 | 154,700 | 135,000 | 147,500 |

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during Cycles 1 and 2 have eliminated all but one theory: an increase in surface roughness in the inlet region (e.g., shield-orifice blocks) of the core assemblies. This increased roughness is believed to be caused by a silicon-based crystal growth. Although the increased pressure drop had little impact on the operation of the FFTF, this phenomenon must be considered in the design of future liquid metal cooled reactors.

During Cycle 2B, small random fluctuations (<2% peak-to-peak) in reactor power were experienced for the first time and they were attributed to a newly inserted experimental control rod. This control rod had more radial flexibility and a higher $^{10}B$ enrichment than standard FFTF control rods. A series of diagnostic tests were conducted to confirm the role of this rod in the power oscillations and to quantify the effects for Cycle 3 operation. At EOC 2, it was completely withdrawn from the core region. Two advanced-design control rods, prototypic of the advanced FFTF absorber design, have been subsequently tested. During later cycles, tests to verify irradiation performance and determine operational characteristics were conducted and radial movement within the clearance between the inner and outer duct is believed to be of source of the fluctuations.

Cycle 2B also introduced the first "interactive" experiment into the reactor. A special 40-ft-long experiment, the Materials Open Test Assembly (MOTA), can contain over 2500 material test specimens in 40 separate canisters. A computer control system permits individual canister temperature measurement and independent adjustments in relation to surrounding sodium temperatures. These measurements permit a more accurate correlation between irradiation effects and temperature than has been previously possible - an essential step in the development and demonstration of extended-life fuel assemblies.

As more experience was gained in operating the plant, new ways to improve efficiency of operations were found. For example, a special Dump Heat Exchanger (DHX) isolation test was conducted at EOC 3 to determine if an isolated DHX could be returned to operation without requiring reduction of reactor power to 5%. The test confirmed that by slightly opening the DHX isolation valve to maintain a trickle flow of sodium and by controlling the temperature with natural circulation air flow, it was possible to restore a DHX module while the reactor is operating at 91% power. Changes which resulted from this test have substantially helped improve cycle capacity factors.

Irradiation of full-size test assemblies using an advanced alloy began during Cycle 4 along with run-to-cladding breach tests, i.e., reference fuel operated beyond design life limits. Startup of the Fuel Storage Facility (FSF) also began during this period. The FSF is a sodium-filled tank containing a carousel with storage positions for 466 fuel assemblies or canisters intended for the storage of spent or irradiated fuel. Sodium fill of the FSF storage vessel was completed in February, 1984. Approximately 117 $m^3$ (31,000 gallons) of sodium brought the sodium to the desired level. Sodium purification and NaK heatup and purification for the two FSF heat exchangers continued through the year.

During the early stages of the fifth outage, some problems were encountered with the instrument and control circuitry in the ex-vessel fuel handling machine. These problems initially delayed refueling activities about one week. In addition, difficulty was encountered in disconnecting primary control rod A-1. A bellows failure had allowed sodium into an area that caused binding and required replacement of the control rod drive mechanism. The time lost to these delays was quickly recovered, and the entire outage effort was completed on schedule.

In the IEM Cell, the highest burnup assembly to date (100,000 MWd/MTM), was processed while decay heat levels were in excess of 4 kW. Corkscrew deformation of the pins in the fuel region was observed.

The Fast Flux Test Facility (FFTF) completed its fifth major operating cycle in November 1984. The total cycle length of 123 full-power days established a new operating record for the advanced test reactor and surpassed the old record of 110 days set earlier in the year with Cycle 4.

In latter 1984, while preventive maintenance was being performed, a sodium leak from an electromagnetic sodium makeup pump was discovered prior to Cycle 6 startup. This failure marked the first leak of the reactor primary sodium coolant in the plant since initial sodium fill. Subsequent pump replacement delayed the plant's return to power by about three weeks.

Tests performed during a mid-Cycle 6 shutdown revealed that high exposure ($9 \times 10^{22}$ n/cm$^2$) and steep flux gradients across the reflectors in Row 7 had caused distortion that resulted in higher-than-expected withdrawal forces for fuel assemblies in Row 6. Consequently, the 13 highest-exposure Row 7 reflectors were discharged and the remaining 19 were rotated 180° in an attempt to extend their residence lifetimes. Both a Row 7 reflector and an adjacent fuel test assembly were examined in the Interim Examination and Maintenance (IEM) Cell to characterize the withdrawal forces. The reflector showed significant bowing while the fuel assembly's measured bow was in the predicted range. This confirmed that the primary cause of the high Row 6 withdrawal forces rests with the Row 7 reflectors and not with adjacent fuel assemblies in Row 6. In fact, the core was so "tight" that a new technique had been developed by the next outage whereby up to five nonadjacent fuel assemblies were removed to "loosen" the core. Though time-consuming, this proved to be a successful technique.

By the end of the first six operating cycles, there had been a total of seven fission gas releases as the result of pin breaches. Two of these events had detectable delayed neutron emissions associated with them. Four of these pin breaches had occurred during Cycle 6 operation, one of which resulted in a sizeable release of fission products into the primary system. Planning for installation of a cesium trap began to evolve, due to interest in future operations with failed fuel and reduction in radiation levels in the primary sodium should future tests involve run-beyond-cladding breach tests. Modifications to install this equipment continued through Cycle 9A.

Also, by the end of Cycle 6, over 35,000 driver fuel pins had been irradiated; 15,000 exceeded the peak burnup design goal of 80,000 MWd/MTM and 4,000 exceeded 100,000 MWd/MTM. Based on this excellent performance, the decision was made to extend the exposure limit from three to up to five cycles, depending on core position. This increased residence time extends the current fuel supply to the year 1990 and resulted in a substantial cost saving to the program.

Cycle 7 operations began in August 1985 and the reactor testing was conducted in two parts: high-power testing (above 75% power) early in the cycle and low-power testing (about 20% power) at the end of the cycle. The reactor was manually shut down in December to replace the two driver fuel assemblies (DFAs) identified as potential fission gas sources. Since both assemblies had identical tag gas and burnup histories, differentiation was not possible. The release of additional fission gas from one of the assemblies when it was removed from the core confirmed the identity of the leaker. This was the only DFA to breach, but this DFA was well beyond design lifetime at 104,000 MWd/MTM.

The Cycle 8A Outage proved to be a successful demonstration of the "mini-outage" concept -- a full refueling without ex-vessel transfers. Key factors in the efficient refueling were the reliability of refueling machines and the fact that the core was not "tight." This last factor was due to assembly rotations in previous refuelings that have compensated for the distortions caused by irradiation.

The plant returned to power operations in February 1986, to begin an extensive series of Inherent Safety Tests (IST). These tests were designed to accurately measure the reactivity feedbacks in a prototypic Liquid Metal Reactor (LMR). These feedbacks are the changes in the balance of neutron production and loss that occur when the temperature conditions in the reactor change. The IST series for Cycle 8A consisted of 198 measurements of the position of control rods at selected power and coolant conditions. The power of the reactor was varied from 10% to 100%. Coolant conditions covered the range of 67% to 100% flow rate and 577°F to 680°F core inlet temperature. Plant conditions for these tests all fell within current operational limits. Evaluations indicate the results are consistent with predictions made before the tests.

Another development during Cycle 8 was the identification of the failed fuel pin in assembly DE-9. DE-9 was the first breached assembly to be processed in the IEM Cell. When the pin weighing system did not definitely pinpoint the breach, a fuel pin washing technique was developed. The subsequent washing of each fuel pin and the removal of the wire wrap eliminated all viewing obstructions from the pin cladding surface. After washing 23 pins, the breach was located by visual examination. There is now high confidence in the capability of locating and identifying a breached pin the Interim Examination and Maintenance (IEM) Cell. If weight difference data cannot pinpoint a breach, there is now a proven backup technique for isolation and identification.

During Cycle 8C, a series of Loss of Flow without Scram (LOFWOS) tests was performed from power levels of up to 50% with flow reduction to both pony motor and natural circulation conditions. Thirteen LOFWOS tests were conducted from power. These tests included 14 power ascents, 15 zero-power tests, 15 primary control rod worth measurements, and 29 startups for zero-power physics tests. There were no unplanned scrams and conduct of testing was error-free. The overall system response was in close agreement with predictions.

Months of preparation culminated in the final week of the Cycle 9 outage with a successful conduct of the complex Integrated Leak Rate Test (ILRT) of the reactor's containment structure. To prepare for the test, most systems and equipment in containment had to be altered in some way. Results of the ILRT showed the FFTF containment leak rate to be at least 20 times lower than test allowable.

FFTF's first full-size metal fuel experiment was installed in Cycle 9, as well as the first isotope production tests. In addition, 16 long-life assemblies (10 fuel and 6 blanket) were installed to begin the Core Demonstration Experiment (CDE). Because of the design of the CDE assemblies, maximum power was reduced to 291 MWt. This refueling effort, though large and complicated, was completed five days ahead of schedule.

## MAJOR ACCOMPLISHMENTS

### Fuel Failure Identification

Early detection and characterization of breached test material specimens and experimental fuel pins in the Fast Flux Test Facility is accomplished by monitoring noble gas radioisotopes. Another paper presented at this meeting will address the gas tagging system in more detail. Since reactor operation began, over 100 individual capsules, one driver and nine test assembly breaches have been successfully identified. A large number of tag gas releases result from the MOTA stress rupture capsules. The gas tag system performed very well, identifying individual capsules, even in multiple release events.

### Passive Safety Tests

These tests are part of a national effort to develop a reactor design that can accommodate a large range of accident conditions without relying on engineered safety components or continuous electrical power to operate the coolant pumps. The natural or inherent characteristics of the reactor will shut down the reactor and carry away the residual heat being produced in the fuel.

Demonstration and characterization of the passive safety capabilities of liquid metal reactors (LMRs) is well underway at FFTF. Two series of ISTs were completed as planned during Cycle 8. One series of tests included a delayed pony motor trip (DPMT) test, a steady-state natural circulation (SSNC) test and a loss of flow without scram (LOFWOS) to pony motor flow test. Results of this series of tests were all close to predictions.

Also completed was a series of LOFWOS tests from power levels up to 50% with flow reduction to both pony motor and natural circulation conditions. This series of tests demonstrated the ability of the Gas Expansion Modules (GEMs) to provide the negative reactivity to turn a LOFWOS event into a benign reactor shutdown (Figure 6).

The Gas Expansion Module (GEM) (Figure 7) is designed to passively shut down the reactor, even in the remote event of an unprotected loss-of-flow event.[3] The device utilizes its internal volume to trap argon cover gas. The gas bubble remains trapped as long as the assembly remains immersed in sodium. The unique feature of the GEM is that it can instantly react to a reduction in flow by sensing the loss of inlet plenum pressure. This loss of pressure permits the gas bubble inside to expand, displacing the sodium down below the core level. This displacement, in the first row of the radial reflector or blanket, causes increased radial and axial neutron leakage from the core which facilitates reactor shutdown. The degree of neutronic shutdown is a function of position, number of GEMs utilized and core geometry.

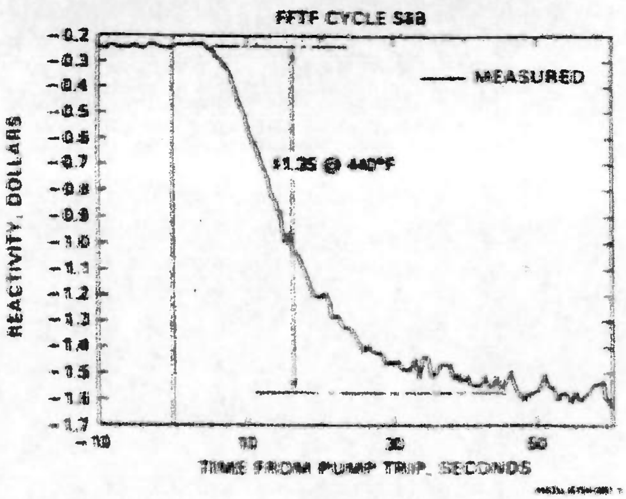


Figure 6. Zero Power GEM Testing Results.

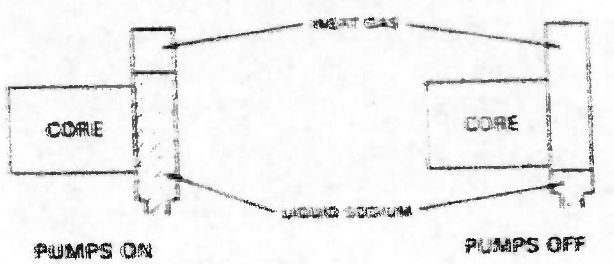


Figure 7. Gas Expansion Module.

Thirteen tests were conducted between 10% and 50% power. In the most aggressive test (50% power/100%), the peak sodium temperatures were lower than predicted. Nine GEMs were tested 30 times and the conclusion is that the GEM is a viable device for safely terminating a LOFWOS event in a LMR.

Advanced Reactor Design Experiments

In support of the innovative liquid metal reactor development program, natural convection air cooling tests were performed using the FFTF Interim Decay Storage Facility (IDS). The IDS is a 43-foot high sodium-filled vessel that is used to store irradiated core components and test assemblies. An annulus that is located outside an enclosing guard tank was designed to provide backup emergency natural draft air cooling. Using this backup system, natural convection air cooling tests were conducted for periods of 3 to 5 hours starting from sodium temperatures of 400°F, 550°F, 700°F, 550°F-repeat and 400°F-repeat. The vessel and guard tank of IDS are well instrumented and additional temporary instrumentation was installed during these tests to measure air coolant temperature rise and flow rate.

A thermal model of IDS, which was developed during initial FFTF acceptance testing, was used for test predictions and in the evaluation of local heat transfer data. Results from the tests indicate that in each case, natural convection air flow was initiated rapidly when inlet and exit valves were opened and effectively cooled the IDS. Overall heat removal was greater than expected and local heat transfer was significantly higher than predicted, using the forced convection Dittus-Boelter relationship for the local heat transfer coefficient. In these tests, local heat transfer by thermal radiation was comparable to convection. Extrapolation of the IDS results to other large annular natural convection cooling concepts appears reasonable.

Reactor Containment Building Integrated Leakage Rate Test

A significant part of the surveillance requirements for the FFTF involves the assurance of isolation of radioactive contaminants from the environment due to either normal operation or as the consequence of an accident. The reactor primary containment building which surrounds the nuclear reactor and its heat transport system acts as a final barrier to mitigate the consequence of a radiological release. Appendix J of 10CFR50 defines the basis for a surveillance program to ensure that the primary containment will perform as designed for the life of the plant.

The most significant test prescribed by Appendix J of 10CFR50, the Reactor Building Containment Integrated Leak Rate Test (ILRT), involves simulating as closely as is practical the predicted conditions within the primary containment after the most severe postulated accident. The leakage of air from the primary containment to the environment is measured to demonstrate that offsite exposure to postulated radioactive contaminants will not exceed the calculated maximum. Various plant design and reactor specific features determine a predicted maximum pressure expected to exist within the primary containment under accident conditions. All valves that close containment penetrations are aligned to the position they automatically would assume after a radiological accident. Lines postulated to rupture inside the primary containment are drained to the extent practical of fluid and vented to containment atmosphere for the duration of the test. Lines postulated to rupture outside the primary containment are drained to the extent practical of fluid and vented to the environment. The ILRT is conducted at a test pressure equal to or greater than the calculated peak pressure expected for the Design Basis Accident (DBA). With 1.7 psig as the predicted peak pressure during a DBA, 2 psig was selected as the appropriate test pressure. An acceptable leakage rate of 0.5 weight % per day at 2 psig meets 10CFR100 dose consequence limits for site selection.

The integrated leakage rate was determined by the Absolute (mass point) Method. The Absolute Method of leakage rate testing constituted the determination and calculation of air losses by containment leakage over a 24 hour period by the means of direct pressure, temperature, and humidity observations during the test period. This method assumes that the temperature variations during the test will be insufficient to effect significant changes in the internal volume of the structure. In practice, however, the calculated mass of air within the FFTF containment displays a distinct nonlinear change which correlates with diurnal temperature changes within the building. The period of apparent mass change is approximately twenty-four hours and appears to be associated with the daily temperature changes. Since the FFTF containment is constructed of a thin steel shell with a few inches of external insulation, this diurnal effect may be due to small changes in the containment volume caused by solar heating of the vessel.

Human Resources: Twelve Hour Shift Schedule

In 1985 the operating crews at FFTF changed from an 8-hour/day to a 12-hour/day rotating shift schedule. The primary objectives of the schedule change were to improve the average level of qualification by reducing attrition and increasing job satisfaction. Although attrition at FFTF is not high by industry standards, a reduction in attrition would raise the average level of experience and qualifications for the operating crews and would reduce training costs.

Before trying the 12-hour/day shift schedule, the primary concern was that operators would become fatigued after working 12 hours per day for four consecutive days, and that operator fatigue could lead to errors that would jeopardize plant safety. The effect of the 12-hour shift on safety was assessed by comparing the number and severity of off-normal events on the 8- and 12-hour shifts. The result of this analysis indicates that there was no statistically measurable difference between the 8- and 12-hour shifts in either the number or severity of events. The evidence on operator alertness was contradictory, however, since evidence from a variety of other sources, including computerized tests of alertness, indicated that most operators are generally somewhat more fatigued on the 12-hour shift. However, as was mentioned, this has not resulted in a statistically measurable increase in the number or severity of off-normal events.

As a result of the schedule change, it seems clear that plant performance improved in two ways. The operator error rate in keeping the Technical Specification Compliance Logs was already very low on the 8-hour shift. On the 12-hour shift, the rate was even lower. Also, the operator interface with craft personnel resulted in an increase in the productivity of the craft personnel.

Fully 84% of the operators were in favor of the 12-hour shift. Although the change caused at least two operators to leave, there is reason to believe that the general increase in satisfaction will eventually lead to a lowered rate of attrition. Thus, the evidence indicates that the 12-hour shift has already met one of its goals (general increase in satisfaction), and will likely meet its other goals (reduced attrition and a higher average level of qualifications), without sacrificing plant performance or safety.

The principal conclusion is that the 12-hour/day shift schedule is a reasonable alternative to an 8-hour/day schedule at FFTF. Since no other facility is exactly like FFTF in all respects, consideration should be given to such factors as employe age, sex, commute time, individual differences, the characteristics of the 12-hour shift schedule, and modifications in shift-turnover procedures before other facilities adopt a 12-hour shift.

## UNIQUE PROBLEMS TO SODIUM-COOLED REACTORS

### Binding of Primary Pump Shaft

A major problem developed during Cycle 1 when one of the three primary sodium pumps would not rotate. During a shutdown period after a scram and subsequent restart of primary sodium pump P-1, a flash-over occurred in the brush/slip-ring area of the pump motor. The resulting pump overspeed and flow imbalance among the three primary coolant loops forced sodium into the shaft annulus and contributed to the binding of the shaft in primary sodium pump P-3. The upper portion of the pump was disassembled, revealing a reduced shaft clearance in the pump baffle area. This reduced clearance was attributed to sodium residue from the unplanned transient. The pump was freed by heating the shield plug and pump tank. After reassembly, vibration and coastdown measurements were taken which verified no additional problems existed.

### Electromagnetic Sodium Makeup Pump Leak

In November 1984, a primary sodium leak was discovered upon entry into a pump cell to investigate a shorted trace heat element. At the time of discovery, the FFTF had completed five operating cycles and was in a refueling and maintenance period prior to startup for the sixth operating cycle. The leak had occurred in the Auxiliary Liquid Metal System, which includes two electromagnetic makeup pumps, P-38 and P-39, which operate in parallel and take suction from the primary system overflow tank, T-42. The pumps are located in steel-lined cells which normally contain a nitrogen atmosphere.

Prior to the discovery, the two primary makeup pumps were frozen off during installation of temporary electrical power for maintenance on the high voltage feeder circuits. Review of plant data suggested that failure occurred either during sodium meltout following the installation of temporary power or during the high flow operation immediately following meltout.

Cleanup and decontamination of the spilled sodium and replacement of the failed pump began immediately after discovery. Initial cleanup and decontamination removed approximately 0.3 $m^3$ (75 gallons) of sodium from the pump cell and reduced the general radiation levels from 100 mrem/hr to 5 mrem/hr. Because of the caustic environment, breathing air and full protective clothing was required to work in the pump area. As work progressed, protective clothing requirements were reduced accordingly. The entire project was accomplished in 33 days, involving 366 entries into the pump cell and yet resulted in a very low overall exposure of 4.5 man-rems.

Examination of the pump duct showed that the failure, which was a small irregular hole, was probably caused by cavitation erosion when the pump was operated at flow rates >3.8 l/s (>60 gpm). The damage was accelerated by duct deformation from multiple freeze-thaw cycles. A replacement pump was installed, along with enhanced leak detection capability on all normally inaccessible sodium pumps. Visual inspections were performed on the other remaining sodium EM pumps. It was concluded that failure of the EM type pumps is not expected if the sodium is kept molten and the pump is operated at flow rates which preclude cavitation.

## FUTURE MISSIONS

### FFTF Power Addition

Demonstration of reliable power generation is an essential stepping stone to LMR commercialization. With the cancellation of CRBR, modification[4] of FFTF is a viable option to fulfill this mission in the U.S. The power addition design philosophy is to minimize interfaces between the nuclear reactor and the conventional electric power plant. Upset conditions in the power plant must be prevented, as much as possible, from affecting reactor operation. The extent to which this philosophy can be effectively implemented may have a bearing on future plant design and operations.

The 400 MW of thermal energy produced at FFTF is currently not utilized. Dump Heat Exchangers reject this heat to the atmosphere. The proposed modification (Figure 8) would add steam generators to two of the existing three secondary sodium cooling loops. The steam from both loops will be combined to drive a single turbine-generator, producing 108 MW of electricity. The power will be fed to the Bonneville Power Administration grid and is sufficient to meet all Hanford loads.

An environmental evaluation was performed in July 1983, which concluded that the anticipated impact of the project is negligible. The project site is within an area which is currently secured and stabilized in conjunction with FFTF operation. No gaseous or particulate effluents will be discharged to the atmosphere.

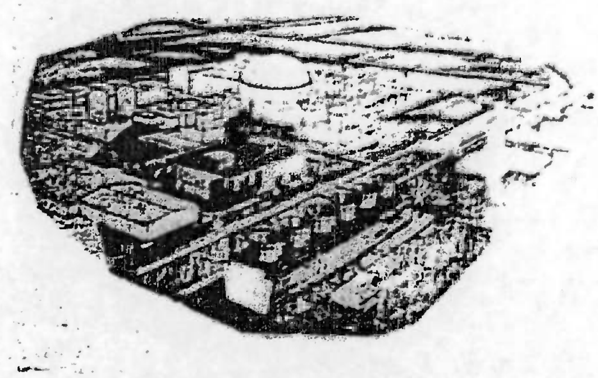

Figure 8. FFTF Power Addition.

### Extended Life Fuel

Evaluation of the FFTF driver fuel has been completed and the performance of this fuel system has been outstanding.[5] Only one driver fuel pin breach has been observed in five years of reactor power operation. The lifetime of the fuel has been demonstrated to be 50 percent greater than the design goal. Now, a third generation of mixed oxide fuel testing is in progress. The Core Demonstration Experiment (CDE) which consists of ten fuel and six blanket assemblies, started irradiation in late 1986 and is expected to continue irradiation until early 1991.

Progress in the development of absorber assemblies has matched that achieved with fuels. Lifetimes of reference absorbers fabricated with 316 SS components have been extended by at least 50 percent, and development tests have supported the design of a new FFTF absorber assembly with a design lifetime of 900 EFPD, a three-fold increase over that for the initial FFTF absorber assemblies.

Since 1980, the highly instrumented and temperature controlled Materials Open Test Facility (MOTA) has been the premiere facility supporting the development of advanced materials for use in nuclear environments. The LMR materials program has progressed from cold worked 316 SS to D9 austenitic alloys, the martensitic HT9 alloy including a modified HT9 with superior high temperature properties, and to the dispersion strengthened alloy MA957. Final evaluation of these alloys has been and will continue to be their performance as cladding and/or duct components of full size FFTF fuel assemblies.

A program to convert FFTF to a long lifetime metal fuel system has begun. The first two lead tests, using uranium/zirconium fuel and long life/low swelling components began irradiation in FFTF in 1987. Additional tests will be initiated in 1988 and beyond, leading to an expected conversion of FFTF to an all metal core starting in 1991. Irradiation of a lead prototypic-sized test for the Integral Fast Reactor (IFR) metal fuel program was initiated in FFTF in 1986.

Irradiation Programs

The FFTF is supporting international cooperation through cooperative LMR development programs with Japan and Switzerland. Two prototypic MONJU fuel and one prototypic MONJU blanket assemblies will begin irradiation in 1987. These tests, using cladding and duct components provided by PNC, will be irradiated under conditions prototypic of the MONJU reactor. Results from these tests and materials property Data from MOTA irradiations will support MONJU licensing activities. A joint uranium-plutonium carbide fuel assembly containing Swiss (EIR) fuel is currently being irradiated in FFTF.[6]

The MOTA is also being used as a major tool for the development and evaluation of materials for fusion reactors as well as materials for space power applications.

Production tests to demonstrate the potential for production of radioisotopes important for medical applications began in Cycle 9A.

SUMMARY OF OVERALL PERFORMANCE

In assessing the 5 years of FFTF operation, it is clear that the plant has operated reliably and safely. Operational experience with sodium systems and component performance unique to liquid metal reactors has been obtained which is vital to development of future LMR designs.

Unplanned automatic scrams are a measure of the plant's reliability. These occur from transients in operation, equipment failures, human error or spurious signals. Figure 9 illustrates the number of scrams that occurred through 1986. In general, FFTF has averaged two unplanned scrams per year.

The safety and reliability of the FFTF is further emphasized by an enviable record in keeping radiation exposures low to operational staff. (Figure 10) The five year average plant radiation exposure was 6.3 man-rem/year, compared to an industry average exceeding 600 man-rem/year for light water reactors.

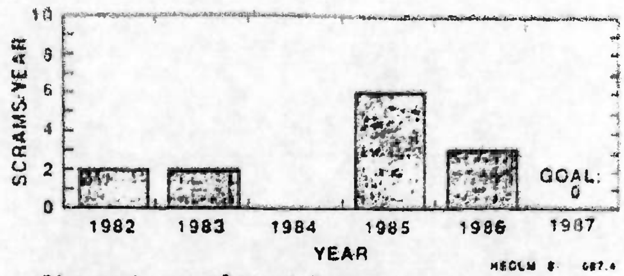


Figure 9. Unplanned Scrams.

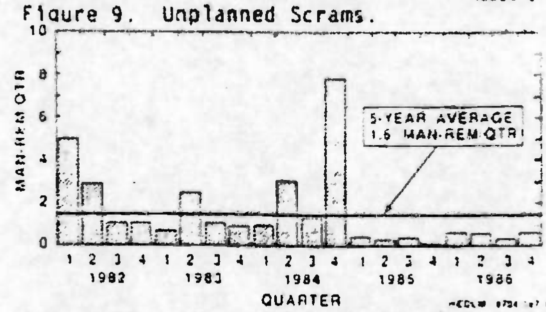


Figure 10. Personnel Radiation Exposure.

ACKNOWLEDGEMENT

The assistance of Joann Barker in typing this document is gratefully acknowledged.

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