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FFTF Operating Experience - 1982 - 1984

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FFTF OPERATING EXPERIENCE 1982 - 1984

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ABSTRACT

The Fast Flux Test Facility (FFTF) is a 400 Mwt sodium-cooled fast reactor operating at the Hanford Engineering Development Laboratory, Richland, Washington, to conduct fuels and materials testing in support of the U. S. Liquid Metal Fast Breeder Reactor (LMFBR) program.

Startup and initial power testing included a comprehensive series of non-nuclear and nuclear tests to verify the thermal, hydraulic, and neutronic characteristics of the plant. A specially designed series of natural circulation tests were then performed to demonstrate the inherent safety features of the plant.

Early in 1982, the FFTF began its first 100 day irradiation cycle. Since that time the plant has operated very well, achieving a cycle capacity factor of 94% in the most recent irradiation cycle. Seventy-five specific test assemblies and 25,000 individual fuel pins have been irradiated, some in excess of 80 MWd/Kg.

INTRODUCTION

The Fast Flux Test Facility is a 400 MW (thermal) sodium-cooled fast test reactor located on the government-owned Hanford site in southeastern Washington State. The FFTF is operated by the Westinghouse Hanford Company for the United States Department of Energy.

The FFTF is a three-loop plant designed primarily for the purpose of testing full-scale core components in a prototypic thermal and neutronic environment. Design of the plant emphasized features to enhance this test capability. Heat removal is by sodium-to-air heat exchangers rather than steam generators. Nickel reflectors surround the core in lieu of a breeding blanket to intensify the neutron flux in the core region. The net result is a plant that is relatively quick and easy to start up and operate and that provides the flexibility needed to fulfill its test objectives. Figure 1 gives the more significant technical parameters of the plant.

The reference driver fuel loading is comprised of about 15,000 pins of co-precipitated U-Pu mixed oxide fuel of 22.5 and 26% Pu. The fuel was manufactured commercially by Kerr-McGee and NUMEC, about equally divided between the two. Each fuel subassembly contains 217 pins spaced by wire wrap and identified by a unique Xe-Kr gas mixture to identify possible cladding failure. Fuel cladding is Type 316 stainless steel (20% cold work), 0.230 inch (5.842 mm) OD by 0.015 inch (0.381 mm) wall thickness.

The reactor itself is uniquely designed to monitor sodium flow rate and exit temperature from each individual core position. In addition, eight core positions can have open-loop test assemblies with instrument leads through the reactor head. Two of these positions have been utilized for Fueled Open Test Assemblies (FOTA). These units have up to 45 thermocouples and an exit flowmeter monitoring fuel performance. An Absorber Open Test Assembly (AOTA) is also presently in core. It has a number of in-core pressure sensors in addition to the in-core thermocouples.

Extensive reactor core characterization measurements were completed to provide the neutron and gamma spectra and profiles, fission rates and other physics data needed to design and evaluate tests irradiated at FFTF. Initial measurements used active sensors traversed axially in an open thimble located near core center. These were followed by passive foil experiments widely distributed throughout the

FFTF TECHNICAL PARAMETERS

• THERMAL POWER		400 MW
• REACTOR VESSEL INLET OPERATING TEMPERATURE	680°F	360°C
• REACTOR VESSEL OUTLET OPERATING TEMPERATURE	938°F	503°C
• NOMINAL CORE ΔT	300°F	167°C
• PRIMARY SODIUM FLOW CAPABILITY	43,500 gpm	2745 ltr/sec
• PRIMARY SODIUM DYNAMIC HEAD	500 ft	152.4 m
• CORE DIAMETER	4 ft	121.92 cm
• CORE HEIGHT	3 ft	91.44 cm
• PEAK FAST FLUX		7×10^{15} n/cm^2-sec
• DECAY HEAT REMOVAL		NATURAL CIRCULATION

Figure 1

core in two separate irradiations, one at low power and the other at full power.

This paper discusses the results achieved in the first three cycles of operation beginning in April, 1982 and carrying through to the fourth reactor cycle which began in January, 1984.

Summary Of Operations

Startup and initial operation of the FFTF was conducted in conformance with a comprehensive Acceptance Test Program (ATP). Objectives of the ATP were to verify plant performance within design criteria and to demonstrate plant operability. An equally important objective was to measure safety-related parameters and confirm safety margins designed into the system. These latter aspects have been addressed in separate papers (refs. 1, 2, 3, 4) and are not treated here. Figure 2 shows the key events in the acceptance test program leading to the first cycle of operation.

Cyclic operation of the FFTF (See Fig. 3) began in April, 1982. Since that time, an average of two 100 day reactor operating cycles have been completed each year. Cycle capacity and availability factors have increased with each successive cycle and are approaching the maximum attainable, considering the testing mission the facility was designed to perform.

At the completion of Cycle 3 the reactor had accummulated 336 Equivalent Full Power Days (EFPD) of operation with a cycle capacity factor of 94%. Operation of the sodium systems has been excellent in the five years since sodium fill. Driver fuel performance has been flawless and more than 25,000 fuel pins have been irradiated in 75 specific test assemblies. The design burnup goal for driver fuel (80 Mwd/Kg) was achieved at the end of Cycle 3 and one driver fuel evaluation assembly will exceed 100 MWd/Kg during Cycle 4. In summary, operation of the plant in support of the irradiation test

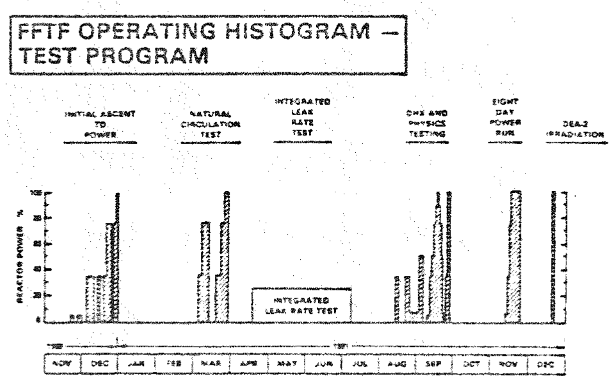
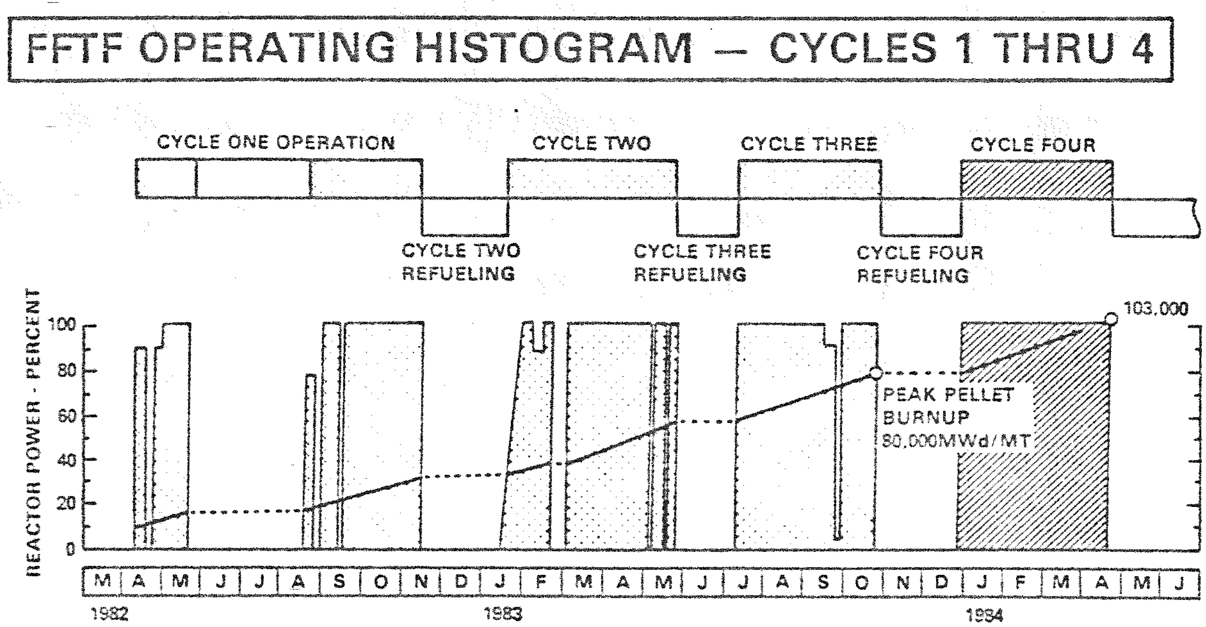


Figure 2



CHRONOLOGY OF KEY EVENTS

4 16 82	CYCLE 1 BEGINS	3 01 83	REACTOR AT FULL POWER
4 16 82	REACTOR SHUTDOWN FROM 52 PERCENT POWER	4 30 83	REACTOR SCRAM
4 23 82	REACTOR AT FULL POWER	5 03 83	REACTOR AT FULL POWER
5 24 82	REACTOR SCRAM	5 07 83	REACTOR SCRAM
6 03 82	PRIMARY PUMP P3 BOUND	5 09 83	REACTOR AT FULL POWER
8 23 82	REACTOR STARTUP IN PROGRESS	5 22 83	REACTOR AT HOT STANDBY – CYCLE 2 COMPLETE
9 02 82	SHUTDOWN FOR DHX MODULE RECOVERY	7 04 83	START CYCLE 3
9 09 82	REACTOR AT FULL POWER	7 18 83	REACTOR AT FULL POWER
9 10 82	REACTOR SCRAM	9 11 83	POWER REDUCED TO 91% FOR DHX MODULE ISOLATION
9 12 82	REACTOR AT FULL POWER	9 26 83	POWER REDUCED TO 83% 2ND DHX MODULE ISOLATION
11 11 82	REACTOR AT HOT STANDBY – CYCLE 1 COMPLETE	9 28 83	POWER REDUCED TO 5% TO RECOVER DHX MODULES
1 17 83	CYCLE 2 BEGINS	9 30 83	REACTOR AT FULL POWER
1 30 83	REACTOR AT FULL POWER	10 23 83	REACTOR AT HOT STANDBY – CYCLE 3 COMPLETE
2 20 83	REACTOR SHUTDOWN FOR MIDCYCLE REFUELING	1 01 84	CYCLE 4 BEGINS
2 28 83	REACTOR RESTART FOR ACCELERATED STARTUP	1 11 84	REACTOR AT FULL POWER

Figure 3

program, has been excellent. Details of the completed operating cycles follow.

Cycle 1 The first cycle began on April 16, 1982 with a low startup rate to allow restructuring of the new fuel in the core. All systems operated as expected until the fuel failure monitoring system detected a small fission gas leak. The subsequent gas tag analysis identified the faulty element (an experiment) and reactor operation continued uninterrupted. After 30 days of trouble free operation, the plant was automatically scrammed due to inadvertent auxiliary system valve operation during routine maintenance. During the recovery from the scram, several primary pump problems arose which took several months to resolve (discussed later in paper). After the pumps were returned to service, the plant operated at full power for 53 consecutive days, surpassing the previous U.S. record for an LMFBR.

Plant parameter monitoring during the early portion of the cycle detected a small, gradual increase in primary system pressure drop. Although this phenomenon did not affect reactor operation, it was of high interest to the LMFBR community. Extensive plant tests to characterize and understand this phenomenon were performed during Cycle 1 and the subsequent cycles to date. This is discussed in more detail in the system performance section of the paper. Cycle 1 was completed on November 11, 1982 with a scheduled reactor shutdown and 101.6 EFPD of irradiation exposure.

Cycle 2 Cycle 2 operation was initiated on January 18, 1983. Full-power operation (Cycle 2A) began on January 30 following several planned holds at various power levels to allow restructuring of fuel and gathering of reactor pressure drop data. During the initial power ascent, two phenomena were observed: 1) a power fluctuation of approximately 0.7% and 2) an unusual number of experimental tag gas releases (occurring earlier than planned) from a materials experiment. Both these phenomena continued throughout Cycle 2 but did not impact overall plant operation.

On February 19, a planned 8-day reactor shutdown was initiated to install a power-to-melt fuel experiment. A rapid programmed startup (to induce limited melting in the experiment) was utilized to return the reactor to full power for Cycle 2B. During this cycle, two diagnostic tests were conducted to determine whether an experimental absorber assembly in a control rod position was the cause of the previously observed power oscillations. This is discussed in more detail later in the paper. Cycle 2 was completed on May 22, 1983 after 100.5 EFPD. The cycle capacity factor was 83% which exceeded the program objective for this early operating cycle.

Cycle 3 Cycle 3 operation, the most successful to date, began on July 4, 1983. Shortly after achieving 35% power on July 6, the plant automatically scrammed due to a voltage transient caused by an electrical storm. The plant was restarted and a slow startup was conducted to allow fuel restructuring. Full power was achieved on July 18. The plant operated at full power for 56 days until September 11, when one of the dump heat exchangers had to be isolated due to a fan trip, requiring power reduction to 91%. On September 26, a second module was isolated, again due to a fan problem. The power was reduced to 5% to recover the isolated modules on September 28, and then the plant returned to full power. Shutdown was initiated on October 22 after over 100 days of uninterrupted nuclear operation. The cycle capacity factor was an outstanding 94%.

Perhaps the most significant event of Cycle 3 occurred at the end of the cycle when three leading fuel assemblies in the reactor achieved the goal burnup of 80 MWd/Kg. This is a major milestone in FFTF's test program since it verifies the integrity of the fuel design and sets the stage for extending fuel life by up to two additional years. The three leading assemblies and four additional ones, which approached the 80 MWd/Kg goal, are highly characterized test assemblies - all using standard driver fuel pins. Most of these assemblies were removed from the reactor during the Cycle 4 refueling outage, but one assembly remained in the reactor for another cycle.

Cycle 4 The outage for Cycle 4 began with core refueling, which was completed by November 18, 1983. Then, in a three-week period, the Materials Open Test Assembly (MOTA-1A) was removed, remotely disassembled, irradiated specimens removed, and a new reconstituted MOTA-1B assembly returned to the reactor.

Throughout the outage, an ambitious surveillance program was performed, which for the first time since the start of nuclear operation, included use of periscopes to inspect the primary sodium piping and pump and IHX guard vessels in one loop. No signs of deterioration were observed.

The plant returned to full power operation on January 11, 1984. By the completion of Cycle 4, the one leading driver fuel assembly will have achieved a burnup of more than 100 MWd/Kg.

RECENT CORE PERFORMANCE

Operation of the plant during the first three cycles has confirmed that the nuclear characteristics are well within design predictions with all parameters remaining inside the operating envelope defined by the Technical Specifications. Temperature and power coefficients have remained substantially negative as the core burnup reaches an equilibrium value. Stability margins are large and reproducible. There were no unanticipated reactivity effects until the beginning of Cycle 2, when the reactor operators observed occasional fluctuations in reactor power, as large as 0.7% of full power in magnitude, measured peak-to-peak. As 100% power was reached, the magnitude of the fluctuations increased to approximately 1% peak-to-peak. Although the plant design can readily accommodate 2% power fluctuations, previous estimates showed that power fluctuations of approximately 0.2% could be expected from the reactor.

An investigation into the cause of the unexpected phenomenon centered on experiments installed in the reactor during the refueling period just prior to the start of Cycle 2. During diagnostic testing, the operators noted that vertical movements of Control Rod 6 caused changes in the magnitude and frequency of the fluctuations. This evidence pointed to an experimental test absorber assembly (ADVAB-2) as the potential initiator of the power fluctuations. The ADVAB-2 experiment had been installed into the Control Rod 6 Position during the previous refueling period.

This experimental control rod utilizes slightly different coolant orificing and a modified driveline with an additional joint to permit rotational centering of the movable portion in the stationary duct. The combination of slightly different hydraulic characteristics and a more flexible drive system apparently causes the absorber bundle to randomly oscillate in the clearance available in the duct and perturb the reactor power. Procedures were developed to permit operation of the reactor with the rod fully withdrawn if the oscillations exceeded 2%. To date, this problem has not required any special operational restrictions and the oscillations have not exceeded 1%. The experiment continues to be irradiated and will be taken to goal exposure.

Physics parameters measured during the startup and initial operations have always been close to predicted values. For example, during the Cycle 4 startup, the secondary rod bank height at criticality was 13.1 inches vs. a predicted 12.8 inches (maximum height - 36 inches). These data generally indicate sufficient reactivity to complete an operating cycle of 110-112 EFPD.

Control rod drop times measured at start of Cycle 4 showed eight of the rods within the normal time of 640-670 milliseconds. An advanced absorber test assembly, with a round cross-section and other advanced design feature changes, dropped in 390 milliseconds (vs. a predicted 410). An excellent result!

PLANT SYSTEMS PERFORMANCE

The FFTF Main Heat Transport System (MHTS) consists of three essentially identical parallel loops. A schematic diagram of an HTS loop is shown in Figure 4. Each of the HTS loops is

HTS LOOP SCHEMATIC

(ONE OF THREE)

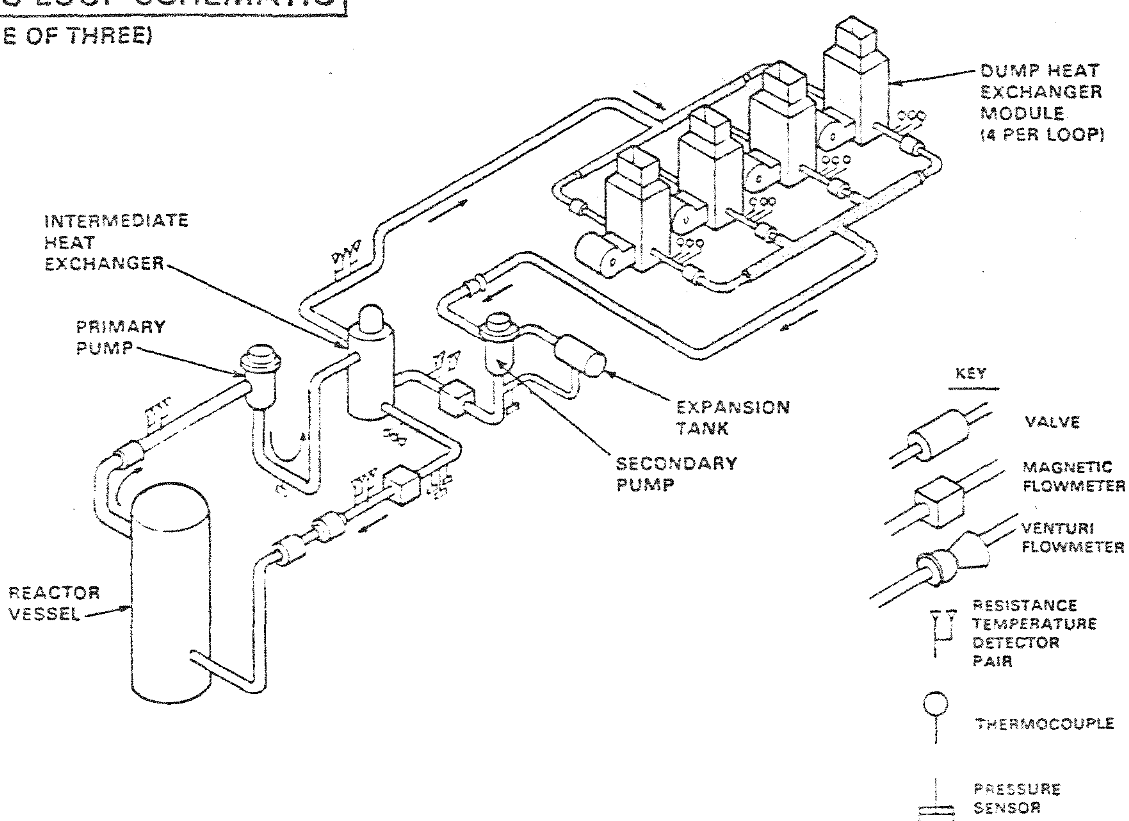


Figure 4

composed of a primary loop (the reactor vessel being common to all three loops) and a secondary loop (the three secondary loops being completely separate from one another). The coolant in both the primary and secondary loops is liquid sodium. Heat generated in the reactor is transferred around the primary loop and then into the secondary loop via a tube and shell heat exchanger. Ultimate heat rejection from the secondary loops is to ambient air via forced air flow dump heat exchangers (DHX).

Heat Exchanger Thermal Performance

Operation through the first three cycles has demonstrated a very constant IHX thermal performance of approximately 7530 $W/m^2 - {}^oC$ (1325 $BTU/Hr\text{-}Ft^2 - {}^oF$) with little variation between the three units (approximately 3%). DHX thermal performance has also been constant with little variation between units. DHX fan horsepower restrictions limit reactor power to less than 100% at ambient temperatures above approximately 29°C (84°F) consistent with predictions. Finally, no significant changes have been noted in DHX heat loss characteristics during periods of plant shutdown. Shutdown heat loss is an operational concern and also has safety implications (e.g., potential for premature freezing during off-normal plant events).

Pump Performance

Pump hydraulic performance is checked quarterly by comparing measured operating characteristics (pump speed, flow and head) to the original pump curve. No significant changes or variations between pumps have been noted. Pump coastdown times are also measured periodically and no significant changes have occurred.

As stated previously, a plant scram occurred approximately one month after the start of Cycle 1. This was caused by the inadvertent draining of the electrolyte from the loop 1 primary pump liquid rheostat. In the process of restarting the primary pump main motors following refill of the rheostat, the motor on the loop 1 pump arced due to an accumulation of oil and carbon dust in the brushes. Replacement of this motor with a spare was required; the motor was subsequently rebuilt. Approximately one week later while the plant was still shut down and operating on pony motors, the pump in primary loop 3 seized. High applied torques were initially unsuccessful in rotating the pump. However, subsequent heating of the upper portion of the pump plus mechanical exercising of the shaft were successful in returning the pump to operation without requiring removal.

Following extensive evaluations, it was concluded that the cause of the seizure was the relocation of a sodium compound deposit which had remained in the upper (cool) portion of the pump shaft/thermal baffle annulus from a pump flooding incident in 1979. Early in Cycle 1 this pump reached its maximum equilibrium temperature, in the shield region, for the first time. After the scram the shield plug was still cooling when the seizure occurred. Although redistribution of this sodium deposit had a significant effect on pump operation at pony motor speed no effect on main motor operation was anticipated and none has been observed. In any case, seizure of one primary pump at full power is an analyzed plant event with demonstrated acceptable results. Shaft swing checks performed in July of 1983 indicate slightly less than nominal annulus clearance and thus the continued presence of sodium deposits in this pump. Further heating of the upper portion of the pump during subsequent power operation is expected to have reduced the sodium deposit somewhat. Periodic evaluation of the annulus clearance is utilized to assess continued pump operability.

Loop Hydraulic Characteristics

Nominal values of the HTS pressure drop are approximately 980 kPa (144 psi) and 544 kPa (80 psi) for the primary and secondary loops respectively. Early in Cycle 1 it was observed that the primary system pressure drop and pump speed were increasing while at constant flow. The primary system pressure drop continued to increase as Cycle 1 operation proceeded but at a continually decreasing rate. At the time of the plant scram in late May 1982, the primary system pressure drop had increased from approximately 980 kPa (144 psi) to approximately 1055 kPa (155 psi). Upon plant restart, it was found that partial recovery of the pressure drop increase to approximately 1027 kPa (151 psi) had occurred. The pressure drop continued to increase through all of Cycle 1B such that by the end of the cycle the value was approximately 1096 kPa (161 psi).

A slight reduction in primary system pressure drop of 14 kPa (2 psi) was expected from Cycle 1 to Cycle 2A due mainly to replacement of one in-core shim by a fueled assembly. In fact a 54 kPa (8 psi) reduction was experienced, again indicating partial recovery of the pressure drop increase. A similar trend of increasing pressure drop during full power operation and partial recovery during shutdown has been experienced throughout the first three cycles of operation. However, both the rate of increase and amount of recovery have become increasingly smaller with time. Figure 5 summarizes the primary system pressure drop history. The solid lines represent actual pump discharge pressure (equivalent to loop pressure drop) while the dashed lines represent the increase over "clean" core component values (i.e., account for core configuration changes and replacement of early core components).

FFTF PRIMARY PRESSURE DROP

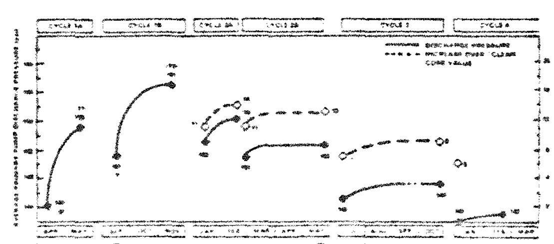


Figure 5

An investigation into the history of the primary system hydraulic performance during the Acceptance Test Program indicated that the pressure drop increase phenomenon had existed from the beginning of operation. However, the power runs were of such short duration that the increases were not readily apparent. It was also observed that complete recovery of any increases occurred during periods of reactor shutdown. As plant operation continued, several theories as to the cause of the pressure drop increase were developed. These theories ranged from mechanical effects to gas entrainment. However, the only theory that has withstood the test of considerable analysis and in-plant testing is an increase in surface roughness in the inlet orifice region (see Figure 6) and lower portion of the fuel pin bundle in the core assemblies. The increased roughness is believed to be caused by the deposition of silicon based crystals which form in the cooler regions of the heat transport system. The source is postulated to be silicon leached from the core and hot leg steel in addition to some contribution from construction residue (e.g., silicon carbide). The phenomenon observed in FFTF is consistent with testing previously done at HEDL specifically to study the hydraulic effects of silicon leaching and crystal deposition in sodium loops (ref. 5).

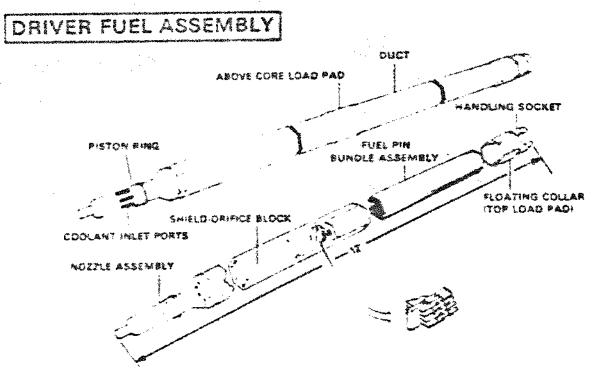


Figure 6

The increasing primary system pressure drop has had no significant impact on plant operation. Sufficient pump capability was available to maintain 100% primary flow. The only other impact was a more frequent recalibration of the IHX discharge pressure instruments which are utilized as a diverse measurement of primary loop flow in the Plant Protective System. Data indicates that only those assemblies in the core during Cycle 1 operation have been significantly affected. As more of these are replaced during future refueling outages, the system pressure drop should approach the "new core" value.

No change in secondary loop hydraulic characteristics has been observed. This can perhaps be explained by the fact that secondary loop hot leg temperature is 462°C (863°F) VS 501°C (938°F) in the primary hot leg (much of the core is considerably hotter). HEDL test loop results indicate that the silicon leaching/crystal growth phenomenon is initiated when the steel reaches approximately 525°C (977°F). Also the secondary loops do not have a high velocity/high pressure drop region such as the shield orifice block of the core assemblies. The hydraulic performance of both the primary and secondary systems will continue to be closely monitored in future operating cycles.

Fuel Failure Monitoring Systems

The fuel failure monitoring system, consisting of on-line cover gas monitoring and delayed neutron detection, has been fully functional since plant startup. The cover gas monitoring system flows a continuous 28,000 sccm flow from the reactor vessel to the monitoring cell, where 900 sccm are passed through a charcoal column adjacent to a germainium diode detector. Four fission product/tag gas activation isotopes are monitored by single channel analyzers for indications of gas release. Back up evaluation is provided by a multichannel analyzer. Tag gas (Xe, Kr mixture) identification is facilitated by concentration of xenon and krypton on a cryogenic charcoal trap, which is then processed in a mass spectrometer to determine the isotopic ratios. Detection of delayed neutrons from fission product release to the reactor sodium coolant is provided by BF_3 proportional detectors in shielded enclosures on each primary sodium loop.

During the first cycle, a single pin in a test assembly developed a small gas leak, the tag was detected and identified and the assembly was removed. Likewise, noble gas tag releases are monitored and used to identify Materials Open Test Assembly (MOTA) creep-rupture specimen failures.

The Delayed Neutron Monitor (DNM) background count rates have stayed essentially constant during the first three cycles. This count rate is associated with the photoneutrons produced by the interaction of gamma rays from the sodium with the deuterium contained in the water in the concrete walls of the cells. None of the observed count rate is believed to be associated with the delayed neutrons which would be expected if the sodium were to come in exposed fuel. No count rate trend is apparent within the normal statistical fluctuations noted. Consideration is being given to a fission product source test that would establish the sensitivity of the DNM system to the injection rate of fission fragments into the sodium at various core locations. Such a test would establish operational and safety limits for DNM count rates during future FFTF operations with breached fuel pins. In the meantime, calculations indicate that the DNM is sufficiently sensitive to detect cladding breaches that are much smaller than can be tolerated in the reactor, thereby giving a wide margin of safety.

Sodium Purification

The primary system contains 950,000 lb. (430,920 Kg) of sodium. Purity in the system is maintained by a 400 gallon (1.51 m^3) NaK cooled flow-through cold trap with a flow rate of 60 gpm (0.227 m^3/min) at an inlet temperature of 900°F (482°C). Each secondary loop contains 133,900 lb. (60,737 Kg) of sodium. Purity in each secondary loop is maintained by a 190 gallon (.72 m^3) air cooled flow-through cold trap with a flow rate of 10 gpm (0.379 m^3/min) at an inlet temperature of 600°F (315°C).

No difficulties have been encountered in the operation of these systems. Tests of impurity source rates, by isolation of the cold traps during power operation, have demonstrated that sources are much lower than expected in the secondaries and are comparable to design in the primary. The measured sodium impurity sources are 30 g/day in the primary and 0.9 g/day in each secondary, versus an expected 20 g/day and 10 g/day respectively.

Head-Mounted Components

Three in-vessel handling machines are used to transfer core components from core positions to in-vessel storage positions or to one of the three fuel transfer ports. Considerable effort was expended during the installation and initial operation of these units to achieve dependable operation. Since start of fuel loading in 1979, the three machines have performed all required tasks without significant maintenance.

The nine control rod drive mechanisms have given five years of service with only limited maintenance. There has been no detectable change in performance with service time. Early in life two of the units experienced bellows leakage, most likely faulty at time of installation. These leaks caused the position indication rods to bind from sodium oxide formation. The units were replaced and there have been no further problems of consequence.

The instrument trees provide temperatures and sodium flow rates from the discharge of each individual core component. They must be rotated away from the core during refueling. Two of the units have been trouble-free. The third unit, however, has experienced high rotating torque possibly due to sodium frost formation in the superstructure. The torque to rotate seems to have stabilized, for the past two years, and no further impact on operation is expected.

Valves

Performance of sodium and cover gas system valves has been excellent. There are about 350 bellows-sealed valves, 4-inch (100 mm) and smaller in service. There have not been any bellows failures. One valve was replaced because of excessive leakage across the seat but it was later determined that it had been supported improperly.

Twenty-four valves with freeze-sealed stems on 8-inch (200 mm) sodium piping have given good service with essentially no maintenance.

The three swing-check valves in the 16-inch (400 mm) primary piping have performed well at temperatures up to 425°C. Performance of these valves is checked annually by shutting down and then restarting one primary loop pony motor at a time.

PERSONNEL EXPOSURE & PLANT RADIATION LEVELS

Plant personnel radiation exposures are very low, as shown in the table below. The values are within the statistical variation in the dosimetry system for very low doses and are at preoperational levels. An on-going program monitors the build up of radiation levels in the various parts of the plant. Of most interest is the increase in activity of the primary system piping and components since this has a direct bearing on the difficulty to perform future maintenance. Radiation measurements recently made in the primary HTS Loop 3 cell show that corrosion product buildup continues essentially as predicted. A residual activity of 230 mrem/hr was measured in the vicinity of the cold leg, which broke down to 131 mrem/hr from Mn-54 and 99 mrem/hr from Na-22. Although small quantities of Co-60 had been predicted (2-4 mrem/hr) in the hot leg area, none has been detected so far. Predictions now say that residual Mn-54 activity after five years of operation will be in the 200-300 mrem/hr range near the cold leg components.

IRRADIATION TEST PROGRAM

Since the major objective of the FFTF is to test core components, it is appropriate to briefly summarize the program status and equipment capabilities. The Cycle 4 experiment loading is described in Figure 7 on the next page. As can be seen, approximately half of the core loading consists of experimental assemblies of some type. Some of the more unique designs are described on the following page.

PERSONNEL EXPOSURE LEVELS

GROUP	2nd Quarter CY-83 No. in Group	2nd Quarter CY-83 Ave. mrem/ Person	3rd Quarter CY-83 No. in Group	3rd Quarter CY-83 Ave. mrem/ Person	4th Quarter CY-83 No. in Group	4th Quarter CY-83 Ave. mrem/ Person
Operations	114	10	87	13	109	1
Support Services	185	4	174	2	177	3
Fuel Handling Cell	30	0	30	3	29	2

FOTA

Fuels Open Test Assemblies are contact temperature and flow instrumented test assemblies with 217 standard driver fuel pins. These tests are designed to verify basic reactor thermal-hydraulic performance and fuel pin cladding conditions. Two FOTAs were inserted with the initial core load. The temperature data obtained during the initial ascent to full power provided the necessary confidence that predicted and actual operating conditions were compatible, so that full power testing was able to progress satisfactorily. The FOTA assemblies were also instrumental in permitting accurate monitoring of core conditions during the early natural circulation test series. One FOTA was removed after Cycle 3 and one continues in the reactor during Cycle 4.

CYCLE 4 EXPERIMENT LOADING

TYPE	NUMBER IN CORE	OBJECTIVE
REFERENCE OXIDE FUEL VERIFICATION	20	ESTABLISH REFERENCE FUEL DESIGN LIFE AND INCREASE IF POSSIBLE.
IMPROVED OXIDE FUEL DEVELOPMENT	15	EXTENDED LIFETIME WITH NONSWELLING ALLOYS AND REDUCED FABRICATION COSTS.
REFERENCE OXIDE BLANKET DEVELOPMENT	2	BLANKET DESIGN FOR FUTURE REACTORS.
ABSORBER (CONTROL ROD) MATERIALS	5	INCREASED LIFETIME, VENTED PINS, LARGER DIAMETER PINS.
SAFETY TESTS	2	PROVIDE PRE-IRRADIATED PINS FOR TRANSIENT TESTING.
STRUCTURAL MATERIALS	5	LONG-TERM IRRADIATION EFFECTS ON STRUCTURAL MATERIALS. ADVANCED ALLOY DEVELOPMENT.

Figure 7

VOTA

The Vibration Open Test Assembly is a 40-foot (12.2 m) long, nonfueled test article designed to measure vibrational and nuclear startup characteristics within the core. It was inserted with the initial core load and provided the needed confidence during initial full flow and power operation that there were not abnormal vibrations in the core. Neutron and gamma flux detectors provided valuable data to verify predicted startup values. The VOTA was removed at the end of Cycle 3.

MOTA

The Materials Open Test Assembly (shown in Figure 8 on the next page) is being used to obtain fast flux irradiation data on materials specimens at a rate much higher than was previously available in the USA. It is a 40-foot (12.2 m) long, instrumented open test assembly. The 12-foot (3.6 m) duct contains eight tiers (levels) containing up to six specimen canisters each. Five tiers are located in the core region, two tiers are above the core, and one tier is located below the core.

During reactor operation, test specimens inside these canisters are subjected to varied irradiation conditions depending on their location in the reactor and the heat and radiation emitted by surrounding fuel and test assemblies.

The MOTA vehicle design permits automatic control of specimen temperatures in 31 of the test canisters. These canisters feature a constant low flow of reactor sodium through an inner chamber containing the test specimens. Specimen heating is via gamma energy deposition. Temperature control is achieved by varying the gas mixture and resulting thermal conductivity in an argon-helium "gas-gap" surrounding each canister. This "gas-gap" design permits researchers to study alloy behavior in fast reactor conditions at temperatures ranging from 830 to 1300°F (443-704°C). The remaining nine canisters are open to the main reactor environment and sodium coolant. Using a computerized control system, MOTA permits direct control and close measurement of specimen temperatures to within a few degrees.

Each of MOTA's 40 canisters contains from 30 to 100 small metal specimens, including tiny pressurized tubes which are used for creep rupture tests. A tag-gas system provides accurate identification of in-reactor stress rupture of these experiments. Many of the specimens are subjected to additional structural material testing after they are removed from the reactor. Some are returned to the reactor after interim examination. A total of 2,000 specimens are in the current MOTA test program.

During MOTA's first period of irradiation testing in 1983, all design goals were met. Successful temperature monitoring of all 40 canisters was performed, as well as individual temperature control of the 31 special canisters.

MATERIALS OPEN TEST EXPERIMENT (MOTA)

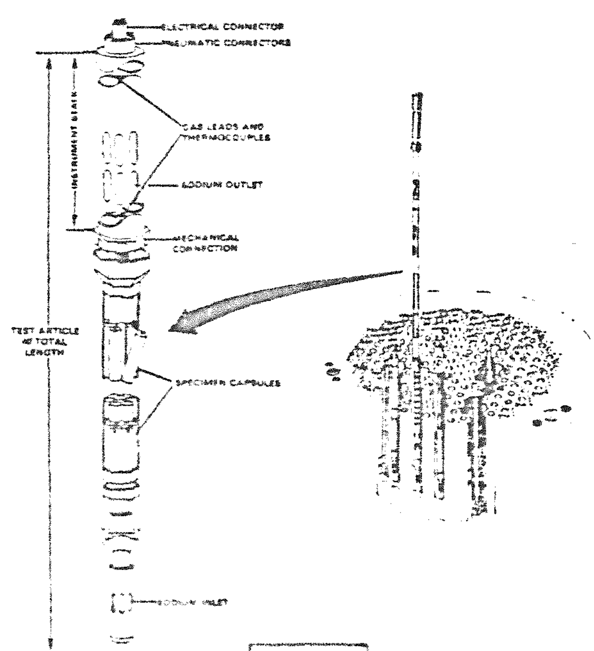


Figure 8

CONCLUSIONS

Operation of the FFTF during its first three cycles can only be summarized as excellent. Performance of the sodium systems has been virtually flawless. The core continues to be very stable and predictable. Improvements resulting from operational experience have been numerous, thus permitting FFTF to achieve an increasing capacity factor each cycle. Future efforts will be directed toward streamlining operations to reduce costs, maximize plant availability, and carry out our assigned mission in LMFBR research and development.

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