Pacific Northwest
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FFTF Technical Documents for Gateway for Accelerated Innovation in Nuclear (GAIN)

September 2021

DL Nielsen
RP Omberg

U.S. DEPARTMENT OF ENERGY

Prepared for the U.S. Department of Energy
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Executive Summary

The Gateway for Accelerated Innovation in Nuclear (GAIN) provides the nuclear energy community with access to the technical, regulatory, and financial support necessary to move new or advanced nuclear reactor designs toward commercialization. GAIN provides the nuclear community with a single point of access to the broad range of capabilities (i.e., people, facilities, materials, and data) across the U.S. Department of Energy (DOE) complex and its National Lab capabilities.

The Fast Flux Test Facility (FFTF) is the most recent liquid metal reactor (LMR) to be designed, constructed, and operated by DOE. The 400-MWt sodium-cooled, fast-neutron flux reactor plant was designed for irradiation testing of nuclear reactor fuels and materials for liquid metal fast breeder reactors. Following the demise of the breeder reactor program in the United States, FFTF continued to play a key role in providing a test bed for demonstrating performance of advanced fuel designs and demonstrating operation, maintenance, and safety of advanced liquid metal reactors. The FFTF Program provides valuable information for potential follow-on reactor projects in the areas of plant system and component design, component fabrication, fuel design and performance, reactor control, prototype testing, and site construction. This report provides documents related to three important aspects of FFTF design and operation: 1) irradiation behavior of structural alloys and absorber materials, 2) thermohydraulics of rod bundles (i.e., coolant mixing), and 3) natural circulation heat transfer in the areas of modeling and validation. These technical documents are believed to be of interest to the nuclear industry and in particular to designers of new liquid metal reactors.

Acronyms

CDE	Core Demonstration Experiment
DHX	Dump Heat Exchanger
DOE	U.S. Department of Energy
FCMI	Fuel Cladding Mechanical interaction
FFM	Fuel Failure Mockup
FFTF	Fast Flux Test Facility
FM	Ferritic-Martensitic
FOTA	Fuels Open Test Assembly
FSAR	Final Safety Analysis Report
FTR	Fast Test Reactor
GAIN	Gateway for Accelerated Innovation in Nuclear
HEDL	Hanford Engineering Development Laboratory
IHX	Intermediate Heat Exchanger
IAEA	International Atomic Energy Agency
IST	Inherent Safety Tests
LMR	Liquid Metal Reactor
LMFBR	Liquid Metal Fast Breeder Reactor
LMFR	Liquid Metal Fast Reactor
LOEP	Loss of Electrical Power
ODS	Oxide Dispersion Strengthened
PCMI	Pellet/Cladding Mechanical Interaction
PPS	Plant Protection System
MOX	Mixed Oxide
MW	Megawatt
NE	Nuclear Energy
WHC	Westinghouse Hanford Company

Contents

1.0 Introduction

The Gateway for Accelerated Innovation in Nuclear (GAIN) provides the nuclear energy community with access to technical support and data necessary to move new or advanced nuclear reactor designs toward commercialization. The FFTF Program provides valuable information for potential follow-on reactor projects in the areas of plant system and component design, component fabrication, fuel design and performance, reactor control, prototype testing, and site construction. This report provides the results of a document search that was conducted in three specific technical areas of FFTF's design and operation:

1. Irradiation behavior of structural alloys and absorber materials
2. Thermohydraulics of rod bundles and associated coolant mixing
3. Natural circulation heat transfer (with emphasis on both modeling and validation)

Following are brief abstracts of each document attached to this report in the three topic areas. These technical documents are believed to be of interest to the nuclear industry and in particular to designers of new liquid metal reactors.

1.1 Irradiation Behavior of Structural Alloys and Absorber Materials

Development of materials for use in LMFBRs in the United States utilized a successful iterative process of making performance predictions for various materials and potential core component designs, designing laboratory and irradiation experiments to measure important properties and to determine the effects of key design variables on performance. Experiments were then fabricated to the extent feasible with commercially viable processes with well characterized materials, experiments were conducted under controlled conditions, and postirradiation examinations were performed to determine the actual performance and to explain any anomalies or failures. This allowed improvement of performance and design codes, and the process could be repeated with improved materials and designs until convergence was made on a design which would meet its objectives. This was then followed with systematic, statistically significant matrices of proof tests addressing both steady state irradiation performance and the transient or off-normal performance of the core component(s). The following documents provide information on development of materials that were used for FFTF reactor components and that were proven to withstand the conditions presented in a liquid metal fast reactor.

1. ***REIC Report No. 45 on The Effects of Neutron Radiation on Structural Materials, June 1967***

   This 253-page document is a compilation of the data available on the effects of radiation on tensile, creep, fatigue, impact, and hardness properties of various structural materials. These properties are given as a function of test temperature, irradiation temperature, and radiation fluence. Specifically, the following reactor materials are covered in the report: 1) aluminum alloys, 2) magnesium alloys, 3) beryllium, 4) zirconium alloys, 5) mild steels, 6) stainless steels, 7) nickel alloys, and 8) refractory metals. Data on the effects of radiation on the mechanical properties of selected materials at cryogenic temperature also are included.

2. ***Fast Flux Test Facility Core System,* American Nuclear Society Conference, 1990**

In the late 1970's and early 1980's, numerous core system design and materials issues were facing the LMR community. This paper provides a review of LMR core system accomplishments and provides an excellent road map through the maze of issues that faced reactor designers at that time associated with fuel pin and fuel assembly performance, irradiation of structural materials, and performance of absorber assemblies. The extensive core systems irradiation program at FFTF addressed each of these principal issues. As a result of the progress made, the attention of long-range LMR planners and designers can shift away from improving core systems and focus on reducing capital costs to ensure the LMR can compete economically in the 21st century with other nuclear reactor concepts.

3. ***Irradiation-Induced Swelling in Commercial Alloys*, Journal of Nuclear Materials, 1981**

Examination of a wide range of commercial alloys indicates that a minimum in swelling occurs at 40-50% nickel. Swelling in these alloys was measured after irradiation to peak fluences of 14.7×10^{22} n/cm^2 (E>0.1 MeV) at temperatures ranging from 400 to 650^{o}F. The minimum in swelling extends over the full range of irradiation test temperatures. Swelling in the alloys appears to be more sensitive to pre-irradiation aging treatments than to relative position on the Fe-Cr-Ni ternary diagram. Pre-irradiation aging of an alloy removes swelling inhibitors from the matrix with a consequent increase in swelling. One of the more important swelling inhibitors removed is carbon. Carbon in solution in the configuration of atomic complexes, which may contain both lattice defects and carbon atoms, can reduce swelling by screening of the dislocation stress fields and hence cause a reduction in the dislocation-interstitial bias factor. A reduction in swelling with increasing equivalent chromium content was also observed.

4. **HEDL-TC-2299, *Evaluation of a Mechanical Attachment for HT9 Ducts*, January 1983**

The lack of irradiated-weldment data on the ferritic HT9 Advanced Alloy prompted development work of an alternative attachment of the hexagonal duct to both the inlet and handling socket subassemblies of a test article. The alternative was a mechanical attachment involving several pins installed in each flat of the duct extending into the inlet and handling socket subassemblies. These pins, each held in place by a snap fastener, resist the predominately tensile loads ordinarily accommodated by the circumferential welds on a 316 SS duct. This report provided analyses to support the use of the mechanical attachment on four FFTF irradiation experiments. Consideration was given to thermal and irradiation swelling phenomena, mechanical loading, fabricability and safety both during and after irradiation. It was not the intent of the analyses to support core-wide use of the attachment concept. A summary of the results of the analyses, mechanical attachment design concept, and recommendations are provided in this report.

5. **HEDL-TC-2293, *Development of HT9 Alloy Duct Attachment*, February 1984**

The HT9 alloy steel continued to show excellent potential for LMFBR fuel ducts. Sufficient data had been generated by materials irradiation studies at HEDL and at national laboratories to proceed with testing of fuel assemblies containing HT9 alloy ducts. Development of a method for attaching end hardware to HT9 alloy ducts had to be completed in order to fabricate fuel assemblies using HT9 alloy ducts. The program objective was to develop methods for attaching an HT9 alloy duct to the handling socket (upper-end hardware) and to the shield and inlet nozzle (lower-end hardware). The end hardware components could be fabricated from either HT9 or 316 stainless steel. Results from the ferritic alloy HT9 weld development program indicated that both welded and mechanical attachment of HT9 alloy duct to end hardware were feasible. Prototypic attachments of both designs were fabricated and tested and both met the basic structural needs for a Fast Breeder Reactor fuel assembly. When comparing the mechanical with the welded attachment, various trade-offs were apparent between the two basic designs, which are discussed in this report.

6. **HEDL-TC-2299, Addendum 1, *Evaluation of a Mechanical Attachment for HT9 Ducts*, April 1984**

   This addendum documents the results and conclusions of the mechanical testing performed in support of the mechanical attachment for HT9 ducts. All testing, specified in the initial documentation (see item 4 above), was performed by the Alloy Development group at HEDL.

7. **HEDL-TME 84-15, *Fuels and Materials for LMFBR Core Components*, April 1984**

   This report reviews development in the United States of fuels and materials for the Liquid Metal Fast Breeder Reactor. Included are the status of technology for fuels and materials for the LMFBR core components, how they got there, and a brief look at where they were headed. The programs described in the report represent the development achieved over the past two decades, which involved many U.S. government and industrial laboratories.

8. **WHC-SP-0454, *Effect of Irradiation on the Fracture Behavior and Tensile Properties of an HT9 Duct*, May 1989**

   Alloy HT9 is a leading candidate material for LMR applications because of insignificant swelling and low thermal expansion. To evaluate applications for this alloy, the ACO-1 experiment was conducted. The HT9 alloy was used for cladding and duct for the FFTF Test Assembly. A part of the postirradiation portion of the experiment was the mechanical property testing of the irradiated duct. Results of tensile and fracture toughness tests on samples cut from the HT9 duct are presented in this report.

9. ***A Liquid-Metal Reactor Core Demonstration Experiment Using HT-9,* ANS Conference, June 1991**

   The use of ferritic/martensitic HT-9 alloy as the cladding and duct material for the Core Demonstration Experiment (CDE) directly contributed to the attainment of high fuel burnup levels critical to the viability of an economical liquid-metal reactor fuel system. The CDE, a partial core loading of fuel and blanket assemblies in the FFTF successfully attained its irradiation exposure goal of three years. The CDE clearly demonstrated the capability of the advanced mixed-oxide (MOX) fuel and uranium dioxide blanket designs for a LMR heterogeneous core to attain high exposures and burnups, leading to a cost-competitive option to the conventional light water reactor. The experiment used advanced assembly and pin designs plus the HT-9 swelling-resistant ferritic alloy for cladding and duct material. Lead fuel experiments continued irradiation significantly beyond the 3.5-year residence period in efforts to establish the ultimate lifetime of the fuel system.

10. ***Dimensional Changes in FFTF Austenitic Cladding and Ducts,*** **ANS Conference 1990**

As the standard cladding and duct material for the FFTF driver fuel, 20% cold-worked 316 stainless steel has provided good service up to a fast fluence of 16 x 10^{22} n/cm^2 in extreme cases. The titanium-stabilized variant of 316 SS, called D9, has extended the useful life of the austenitic alloys by increasing the incubation fluence necessary for the onset of volumetric swelling. Duct flat-to-flat, length and bow, pin bundle distortion, fuel pin diameter and length, as well as cladding volumetric swelling have been examined for high fluence components representing both alloys. These data emphasize the importance of the swelling process, the superiority of D9, and the interrelation between deformation in the duct, bundle, and individual pins.

11. ***Fuel Pin Mechanical Behavior: Ten Years of LMR Experience,*** **ANS Conference 1990**

This paper reviews some of the key developments on the mechanical behavior of mixed oxide fuel pins achieved during the 1980’s. Fuel pin performance data obtained on high and low swelling cladding alloys are reviewed. The role of fuel-cladding mechanical interaction (FCMI) effects is emphasized including a discussion of cladding breach experience and criteria. In addition, the challenges and achievements of the next decade are described.

12. ***Irradiation Performance of Fast Reactor MOX Fuel Pins with Ferritic/Martensitic Cladding Irradiated to High Burnups*****, Journal of Nuclear Materials, March 2011**

The ACO-3 irradiation test, which attained extremely high burnups of about 232 GWd/t and resisted a high neutron fluence (E>0.1 MeV) of about 39 x 10^{26} n/cm^2 as one of the lead tests of the Core Demonstration Experiment (CDE) in the FFTF, demonstrated that the fuel pin cladding made of ferritic/martensitic HT-9 alloy had superior void swelling resistance. The measured diameter profiles of the irradiated ACO-3 fuel pins showed axially extensive incremental strain in the MOX fuel column region and localized incremental strain near the interfaces between the MOX fuel and upper blanket columns. These incremental strains were as low as 1.5% despite the extremely high level of the fast neutron fluence. Evaluation of the pin diametral strain indicated that the incremental strain in the MOX fuel column region was substantially due to cladding void swelling and irradiation creep caused by internal fission gas pressure, while the localized strain near the MOX fuel/upper blanket interface was likely the result of the pellet/cladding mechanical interaction (PCMI) caused by cesium/fuel reactions. The evaluation also suggested the PCMI was effectively mitigated by a large gap size between the cladding and blanket columns.

13. ***Performance of Fast Flux Test Facility Driver and Prototype Driver Fuels*****, ANS Conference, 1990**

This paper discusses the experience gained with seven driver fuel designs irradiated in the FFTF during its first ten years of operation. These designs include austenitic stainless steel-clad $(U,Pu)O_2$ fuels, as well as $(U,Pu)O_2$, UO_2, and U-10Zr fuels clad with ferritic-martensitic (HT9) steel in wire-wrapped pin bundles contained in a hexagonal duct. Selected observations of fuel pin behavior are reviewed, including pin lifetime results from a test with pin powers high enough to cause fuel melting at the beginning-of-life. The excellent performance of several of these fuel designs under transient safety testing in the Transient Reactor Test facility are also reviewed.

14. ***Structural Materials for Liquid Metal Cooled Fast Reactor Fuel Assemblies – Operational Behavior*, IAEA Nuclear Energy Series**

This report summarizes the results of two IAEA sponsored technical meetings, conducted in 2008 and 2011. One of the main objectives of the meetings was to share and exchange information on stainless steel structural materials for liquid metal cooled fast reactor fuel assemblies, producing a final report that would: 1) identify the different varieties of austenitic, nickel based, ferritic-martensitic (FM) and oxide dispersion strengthened (ODS) steels having demonstrated success or potential improved performance as structural components of fast reactor fuel assemblies, with particular emphasis on fuel cladding; 2) summarize the manufacturing processes of liquid metal fast reactor (LMFR) fuel cladding tubes, rods for end plugs, sheets, wrappers, wires, etc. starting from ingot preparation; 3) summarize the irradiation behavior of these steels in fast reactor service; and 4) focus in particular on the ODS variants of ferritic and FM steels as the path forward to achieving higher burnup of fuel in fast reactors. This report represents the culmination of numerous past and ongoing activities conducted by IAEA member states.

1.2 Thermohydraulics of Rod Bundles and Associated Coolant Mixing

The thermal and hydraulic characteristics of the FFTF fuel were based upon a “reference” core configuration which was established by the FFTF project. In reality, the actual configuration in the FFTF reactor at any given point in time varied in several respects from the reference configuration (e.g., type of test assemblies, number of fixed-shims, etc.). However, since it is impossible to anticipate or analyze all potential core configurations, the reference core served as the basis for all design calculations. The hydraulic characteristics of FFTF were controlled to a large degree by the fuel assemblies, therefore, small perturbations in the core configuration would generally have a negligible effect upon the thermal and hydraulic parameters of the core. Major changes from the reference configuration required analysis on a case-by-case basis. The following documents discuss analytical work and tests that were done to characterize flow and temperature conditions in the core and to verify these thermohydraulic conditions.

1. **BNWL-1207 FF – *An Experimental Study of Pressure Drop in Wire Wrapped FFTF Fuel Assemblies*, September 1969**

Fuel assembly pressure drop is a prime concern in the design of the Fast Test Reactor (FTR). Knowledge of the pressure drop—flow characteristics of the core is required to optimize the design of the reactor. The complicated geometry of the FFTF fuel assemblies prohibited the accurate analytical prediction of pressure drop. These fuel assemblies consisted of 217 wire wrapped fuel pins enclosed in a hexagonal flow tube. Liquid sodium flowed through the interstitial channels between the fuel pins to provide cooling. A typical flow channel is tri-cusped with an equivalent hydraulic diameter of approximately 0.10 inch. The flow channels were distributed periodically in the axial direction by the spirally wound wire wraps. Several calculation techniques for predicting pressure drop in wire wrapped bundles can be found in references to this paper; however, the geometrical parameters in the data used to compile these empirical relations were not prototypic of FFTF fuel assemblies. To provide the necessary pressure drop information, an experimental parametric study was performed with water as the test fluid. Conventional Reynolds number modeling permitted the direct application of the water data to liquid sodium flow. This report presents the results of the experimental program.

2. **FRT-1582 – *Steady State Thermal and Hydraulic Characteristics of the FFTF Fuel Assemblies*, December 1975**

   This 188-page document provides a summary of the thermal and hydraulic characteristics of the FFTF fuel assemblies. The data shown are based upon a “reference” core configuration which was established by the FFTF project. In reality, the actual configuration could vary in several respects from the reference configuration (e.g., type of test assemblies, number of fixed-shims). However, since it was impossible to anticipate or analyze all of the potential core configurations, the reference core served as the basis for all design calculations. The hydraulic characteristics of the FFTF were controlled to a large degree by the fuel assemblies. Small perturbations in the core configuration, in general, had a negligible effect upon the thermal and hydraulic parameters of the core. For example, increasing the number of fixed-shim assemblies from three to fifteen would only increase the maximum cladding temperature in the core by 2^{o}F. The data presented in this report were believed to be representative of the design values expected in FFTF. However, it was noted that major perturbation to the reference configuration may result in some differences in the characteristics of the core and would need to be evaluated on a case-by-case basis.

3. **Conference Paper 7401401-P3:1139 – *Effect of Partial Blockages in Simulated LMFBR Fuel Assemblies*, December 1975**

   Experimental data from the Oak Ridge National Laboratory LMFBR Fuel Failure Mockup Facility (FFM) were reviewed to evaluate possible effects of hypothetical partial blockages in FFTF fuel subassemblies. The FFM was an engineering-scale sodium flow loop in which fuel rod assemblies were simulated with 19-rod bundles of electric cartridge heaters having heat flux and external configuration similar to FFTF fuel. Data were obtained with bundle inlet blockages of 13 and 24 channels and with heated-zone internal blockages of six channels. Various analytical models were used to interpret the data. The ORRIBLE code was used to compute the flow and temperature distributions in the rod bundles downstream of the recirculating wake zone. (ORRIBLE uses a simplified flow formulation and is computationally stable for cases where partial areas of the rod bundle are blocked, but it has no provisions for computing detailed flow distributions within the recirculating wake zone behind a blockage). Temperature distributions in the recirculating wake zone immediately behind the blockage were estimated using simple arbitrary representations of the wake flow with varying levels of recirculation velocity and blockage sizes. Temperatures within the blockage itself were calculated using HEATING, a three-dimensional heat conduction code. Heat conduction computations were made for heat-generating and non-heat-generating blockages with arbitrary shapes and boundary conditions. This paper provided a review of partial results obtained from the program at that time; more definitive conclusions were expected upon overall completion of the program.

4. **HEDL-SA-1838-FP, *CORA, Transient Analysis Code for a Cluster of Reactor Core Assemblies*, American Nuclear Society Conference June 3-7, 1979**

This paper provides a presentation given at the ANS conference in 1979 discussing Westinghouse Hanford Company's (WHC) development of a liquid metal cooled reactor transient analysis code to answer certain questions concerning thermal and hydraulic coupling of reactor assemblies. At the time there was no such operational code, so WHC embarked on developing their own code, which was provisionally operational within a year. The name of the code is CORA, which stands for Coupling of Reactor Assemblies. It is a steady state/transient, core thermal hydraulics code for liquid metal cooled fast reactors. The version of the code discussed in this presentation is specific for FFTF. However, the paper states that CORA could be made general for LMFBR type reactors with a reasonable amount of additional code development.

5. **HEDL-SA-2007, *Instrumented Fuels Test for FFTF*, ANS Conference, June 8-13, 1980**

The Fuels Open Test Assembly (FOTA) was developed for fuels testing at the FFTF. FOTA is a test vehicle designed to contain and support instrumented fuel experiments in the FFTF reactor. The initial two FOTA experiments characterized the reference driver fuel assembly performance and provided experimental data evaluating thermohydraulic models that were used to predict assembly performance, evaluate natural circulation cooling and core characterization of the reactor. This paper describes the design features and fabrication of the first two FOTA instrumented fuel experiments and provides a brief description of the FOTA test vehicle.

6. **HEDL-SA-2330 FP, Rev. 1, *Analysis of In-Core Coolant Temperatures of FFTF Instrumented Fuels Tests at Full Power*, American Nuclear Society Conference, June 7-12, 1981**

Two full-size highly instrumented fuel assemblies were inserted into the core of the FFTF in December of 1979. Following installation, FFTF underwent two ascents to 100% power (400 MW) and also conducted natural circulation tests. The major objectives of those instrumented tests were to provide verification of the FFTF core conditions and to characterize temperature patterns within FFTF driver fuel assemblies. This paper provides a review of the results obtained during the power ascents and during irradiation at a constant reactor power of 400 MWt. The results obtained from these tests verified the conservative nature of the design methods used to establish core conditions in FFTF. The success of these tests also demonstrated the ability to design, fabricate, install and irradiate complex, instrumented fuels tests in FFTF using commercially procured components.

7. **HEDL-SA-2203-FP, *Thermal Hydraulics of Handling FFTF Fuel in Sodium and Argon*, *Comparing Test Results with Calculated*, ANS Conference, November 17-21, 1980**

This paper discusses the understanding of thermal hydraulic characteristics for handling FFTF fuel that was gained from extensive tests using an instrumented, electrically heated, prototypic fuel test article. The tests were performed to determine the temperature distributions in FFTF fuel as it is handled in transfer pots as well as out of pots during disassembly for examination. The detailed thermal mapping needed for handling fuel in storage locations, in the reactor, and during transfers in handling machines was obtained for fuel in stagnant sodium and stagnant atmospheres of argon and helium.

8. **WARD-1522, *Simulation of Interchannel Coolant Mixing in a Sodium Cooled Reactor Fuel Assembly***

Coolant mixing in reactor assemblies has been widely studied through experiments in models using fluids much easier and cheaper to handle than sodium. A simulation problem is faced in applying the results of the past experiments or planning new experiments for LMFBR conditions. The experimental conditions must be defined and a correlation is needed between the reactor (sodium) conditions and the mixing data obtained in other fluids. This paper presents methodology that allows calculation of the actual mixing flowrate from experimental values. However, it applies to bare rod bundles only, since the turbulent mixing component alone is considered. In the case of gridded assemblies, for example, another component due to flow scattering must be added which, being purely hydraulic, needs no correction. The total mixing flowrate is determined by evaluating separately the turbulent and flow scattering components and applying Equation 2 provided in this report to the turbulent part alone.

1.3 Natural Circulation Heat Transfer

FFTF natural circulation tests were completed in March 1981. They demonstrated that with simulated emergency loss of all electrical power, the sodium coolant continued to circulate naturally through the loops by convection, removing decay heat from the reactor core. The plant was designed with progressive elevation differences between the reactor core, Intermediate Heat Exchanger (IHX) and Dump Heat Exchanger (DHX), in order to provide sufficient thermal head for natural convection. Sufficient flow was required so that fuel damage was precluded for design basis conditions, including allowances for uncertainties in the maximum decay heat source. The objectives of the natural circulation tests were therefore to:

1. Demonstrate natural circulation capability of the plant to remove decay heat during emergency Loss of Electrical Power (LOEP) events.

2. Establish a progression of natural circulation transient tests to ensure that no damage to FFTF fuel or components occurs.

3. Collect data for verification and improvement of analytical models used for predicting natural circulation under design basis conditions.

Following are documents that relate to the development and implementation of the Natural Circulation Test Program at the FFTF. Documents 11 and 12 are of related interest, but not directly associated to the natural circulation testing done at FFTF.

1. **8451530, *FFTF Natural Circulation Tests Summary*, April 30, 1984**

This letter provides a brief summary of the FFTF natural circulation tests.

2. **HEDL-SA-2007, *Instrumented Fuels Test for FFTF*, ANS Meeting June 8-13, 1980**

   In support of the LMFBR Fuels Development Program, HEDL designed the Fuels Open Test Assembly (FOTA) for fuels testing at FFTF. The FOTA is a test vehicle designed to contain and support instrumented fuel experiments at FFTF. The initial two FOTA experiments were used to characterize the reference driver fuel assembly performance in the reactor and provide experimental data to evaluate thermohydraulic models used to predict assembly performance. Instrumentation data from these experiments were also used in the evaluation of the natural circulation cooling capabilities and core characterization of FFTF. This paper describes the design features and fabrication of the first two FOTA instrumented fuel experiments and presents a brief description of the FOTA test vehicle.

3. **HEDL-SA-2330-FP, *Analysis of In-Core Coolant Temperatures of FFTF Instrumented Fuels Tests at Full Power*, ANS Meeting June 7-12, 1981**

   Two full size highly instrumented fuel assemblies were inserted into the core of the FFTF in December of 1979. The two assemblies underwent two ascents to 100% power (400 MW) and underwent natural circulation testing. The major objectives of these instrumented tests were to provide verification of the FFTF core conditions and to characterize temperature patterns within FFTF driver fuel assemblies. This paper provides a review of the results obtained during the power ascents and during irradiation at a constant reactor power of 400 MWt. The results obtained from these instrumented tests verified the conservative nature of the design methods used to establish core conditions in FFTF. The success of these tests also demonstrated the ability to design, fabricate, install and irradiate complex, instrumented fuels tests in FFTF using commercially procured components.

4. **PNL-3259, *COBRA-WC: A Version of COBRA for Single-Phase Multiassembly Thermal Hydraulic Transient Analysis*, July 1980**

   The objective of this report is to provide the user of the COBRA-WC (Whole Core) code a basic understanding of the code operation and capabilities. Included in this manual are the equations solved and the assumptions made in their derivations, a general description of the code capabilities, an explanation of the numerical algorithms used to solve the equations, and input instructions for using the code. Also, the auxiliary programs GEOM and SPEC-SET are described and input instructions for each are given. Input for COBRA-WC sample problems and the corresponding output are given in the appendices.

5. **PNL-SA-8318, *COBRA-WC Model and Predictions for a Fast Reactor Natural-Circulation Transient*, January 1980**

   The COBRA-WC (whole core) code was used to predict the core-wide coolant and rod temperature distribution in a liquid metal fast reactor during the early part (first 220 seconds) of a natural circulation transient. Approximately one-sixth of the core was modeled including bypass flows and the pressure losses above and below the core region. Detailed temperature and flow distributions were obtained for the two test fuel assemblies. The COBRA-WC model, the approach, and predictions of core-wide transient coolant and rod temperatures during a natural circulation transient are presented in this paper.

6. **PNL-SA-10557, *COBRA-WC Pretest Predictions and Post-Test Analysis of the FOTA Temperature Distribution During FFTF Natural-Circulation Transients,* ANS 1982 Winter Meeting**

   Natural circulation tests at FFTF demonstrated a safe and stable transition from forced convection to natural convection and showed that natural convection may adequately remove decay heat from the reactor core. The computer code, COBRA-WC, was developed by the Pacific Northwest National Laboratory (PNNL) to account for buoyancy-induced coolant flow redistribution and interassembly heat transfer, effects that become important in mitigating temperature gradients and reducing reactor core temperatures when coolant flow rate in the core is low. This report presents work sponsored by DOE with the objective of checking the validity of COBRA-WC during the first 220 seconds of the FFTF natural-circulation (plant-startup) tests using recorded data from two instrumented Fuel Open Test Assemblies (FOTAs). Comparison of COBRA-WC predictions of the FOTA data is a part of the final confirmation of the COBRA-WC methodology for core natural-convection analysis.

7. **HEDL-SA-1353, *FFTF Dump Heat Exchanger Design and Development*, First Joint US/Japan Seminar on BRP Plant Components**

   This report is a brief summary of the history of procurement of the FFTF Dump Heat Exchangers. A description of the design basis and features of the large main heat transport system DHXs is included. Design evolution is discussed and testing performed as part of the design process, including finned tube qualification, header hydraulic tests, air-pressure drop test, insulated panel tests and friction/wear testing, are summarized. Manufacturing experience, including development of an integral finned tube, development of welding procedures for joining the finned tubes, and header fabrication are also presented. Testing results obtained with the prototype unit are described, as are additional feature model tests that were performed to define possible methods of improving heat transfer performance for the plant units. The paper concludes with a summary of construction experience, status of plant modifications and a description of expected and transient plant operation, including decay heat removal.

8. **HEDL-SA-2957-FP, *FFTF Operating Experience 1982-1984*, Third International Conference on Liquid Metal Engineering and Technology in Energy Production, April 9-13, 1984**

   Startup of FFTF 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. This paper discusses various aspects of the reactor's operation and the operational cycles completed since startup.

9. **HEDL-SA-2326-FP – *FFTF Operating Experience with Sodium Natural Circulation*, ANS Meeting June 7-112, 1981**

   FFTF was designed for passive, back-up, safety grade decay heat removal utilizing natural circulation of the sodium coolant. This paper discusses the process by which operator preparation for this emergency operating mode was assured, in parallel with the design verification during the startup and acceptance testing program. Over the course of the test program, additional insights were gained through the testing program, through on-going plant analyses and through general safety evaluations performed throughout the nuclear industry. These insights led to development of improved operator training material for control of decay heat removal during both forced and natural circulation as well as improvements in the related plant operating procedures.

The keystone of the operator preparation activity was an extensive natural circulation test program. A series of heat transport system sodium natural circulation tests were performed at FFTF over a nearly two-year period beginning in March of 1979. Initial testing was of one secondary loop only and included no nuclear heating. Following core loading and initial criticality in 1980, reactor scrams to natural circulation were performed at several power levels as part of the power ascent and plant acceptance testing. The test series culminated with a scram from full reactor power (400 MWt) to natural circulation in both the primary and secondary loops; the plant was subsequently cooled and maintained on natural circulation for approximately 30 hours. This final test demonstrated conclusively the capability of the reactor plant to safely dissipate decay heat in the unlikely case that all sources of electric power (for coolant pumps as well as normal control system operation) were lost. As a result of this work on operator training and procedure preparation as well as actual operating experience, the FFTF operators were well qualified to safely perform a plant shutdown following a complete loss of electric power.

10. **HEDL-SA-2437FP, *FFTF Natural Circulation Tests*, ANS Winter Meeting 1981** FFTF natural circulation tests were completed in March 1981. They demonstrated that with simulated emergency loss of all electrical power, the sodium coolant continued to circulate naturally through the loops by convection, removing decay heat from the reactor core. The purpose of this paper was to present the objectives, criteria and data demonstrating the natural circulation capability of the FFTF.

11. **Nuclear Engineering and Design 66 (1981) 437-446, *Prediction of Decay Heat Removal Capabilities for LMFBRs and Comparison with Experiments*, 1981** The expected mode of decay heat removal in most of the current LMFBR designs is via forced circulation of liquid sodium through the main coolant pumps driven by auxiliary pony motors (CRBRP, FFTF, PHENIX, PFR) or via other means such as electromagnetic pumps (EBR-II). Alternatively, when the main circuit is either completely or partially unavailable, an auxiliary cooling system or path is provided. Such an auxiliary system can be either completely independent (external core cooling system) or semi-independent. However, in the event of a complete loss of forced cooling the plants are designed to utilize buoyancy forces to provide free convection as a redundant and diverse means of decay heat removal. In this situation the buoyancy head that may exist between the hot and cold legs of the circuit could establish an adequate level of natural convection flow to dissipate the shutdown decay heat. This paper discussed investigating the decay heat removal capability of LMFBR plants in two steps: the transition phase and the established phase of natural circulation. Major parameters affecting the transition phase are briefly reviewed. This paper considers two analytical approaches to study plant behavior for the established phase, i.e., the long-term heat dissipation capability of the plant. Advanced thermohydraulic system simulation codes (SSC-L and SSC-P) are used to study the response of the CRBRP, FFTF and PHENIX plants. Simplified analyses were also performed for these plants, as well as to interpret the actual whole plant test data from PFR, PHENIX and EBR-II. All of these studies have

shown that the mixed mean reactor core temperature rise was proportional to the 0.58 exponent of the shutdown steady power level.

12. **Nuclear Engineering and Design 68 (1981) 323-336, *Verification Study of the FORE-2M Nuclear/Thermal-Hydraulic Analysis Computer Code*, 1981**

    Verification of the LMFBR core transient performance code, FORE-2M, was performed in two steps. Different components of the computation (individual models) were verified by comparing with analytical solutions and with results obtained from other conventionally accepted computer codes (e.g., TRUMP, LIFE, etc.). For verification of the integral computation method of the code, experimental data in TREAT, SEFOR and natural circulation experiments in EBR-II were compared with the code calculations. Good agreement was obtained for both of these steps. Confirmation of the code verification for undercooling transients was provided by comparison with the FFTF natural circulation experiments. This paper describes the approach that was used to verify the FORE-2M code.

13. **HEDL-TC-2785, *Test Design Description Volume IA for the FOTA HF185 Experiment*, September 1986**

    The reusable FOTA test HF185 was to monitor core thermal performance in the FTR during the Inherent Safety Tests (IST). In-fuel and coolant thermocouples were to correlate thermohydraulic performance models. Instrumentation data from the FOTA was also to be used in the Plant Protection System (PPS) to assure maintenance of acceptable core temperatures during the transients. The reusable feature of this FOTA would permit it to be used during the several planned IST tests and thus extend its useful life to several operating cycles. The purpose of this analysis was to assure that the modifications from the original design were acceptable. The operating conditions for the entire inherent safety test series were not known at that time, so the purpose of the report was to describe an acceptable operating envelope for the IST FOTA which went beyond the previously analyzed conditions.

14. **WHC-SD-FF-TDD-038, *Test Design Description for the IST FOTA (HF185) Experiment, Volume II, Part 1*, October 1991**

    This report provides fabrication data for the IST FOTA In-Fuel thermocouple fuel pins.

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