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Fast Flux Test Facility Benchmark Status for FY19

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Executive Summary

The Department of Energy's Office of Nuclear Energy Advanced Reactor Technologies (ART) Program works on promising Generation-IV R&D. The Program focus is on reactor concepts that could dramatically improve performance in sustainability, safety, economics, security, and proliferation resistance. The near-term ART Program goal is to reduce technical barriers (e.g., by resolving key feasibility and performance challenges) to improve the prospects of advanced technology reactor systems. In the long-term, integrated advanced reactor system designs will be developed that implement the chosen missions and complete the testing and demonstration necessary for licensing.

The Fast Flux Test Facility (FFTF) is the most recent Liquid Metal Reactor (LMR) to operate in the United States, from 1982 to 1992. The plant description and plant data from one specific passive safety test is now an International Atomic Energy Agency (IAEA) Coordinated Research Project (CRP). This report is one in a series documenting the PNNL efforts to support this FFTF loss of flow benchmark for the IAEA.

Previous reports[1,2,4] described the efforts to:

- Draft FFTF Loss of Flow without Scram (LOFWOS) test #13 benchmark specifications
- Revise draft of FFTF LOFWOS Test #13 benchmark specifications for international use
- Draft Joint ANL/PNNL FFTF LOFWOS Benchmark report
- Work with ANL to update the model of the specific FFTF passive safety benchmark test in SASSYS, providing comparable results of six other LOFWOS tests, and investigating timing and thermocouple calibration issues
- Continue support of FFTF LOFWOS benchmark as IAEA CRP, including review of and providing comments on draft Benchmark report, and travel to IAEA in Vienna to plan CRP proposal.

This report describes the following efforts in FY19:

1. Continued to work with ANL in their efforts to model another series of FFTF passive safety tests in SASSYS
    a. Provided ANL with recorded data from the first 10 steps in steady-state reactivity feedback measurements performed during FFTF Cycle 8A
    b. Answered ANL's questions related to the data provided
    c. Provided ANL with basic data from all steps in the reactivity feedback measurements during FFTF Cycle 8A
2. Continued support of FFTF LOFWOS benchmark as IAEA CRP
    a. Reviewed and commented on Benchmark report
    b. Investigated comments from participants on Benchmark report
    c. Assisted in issuance of Revision 1 to the FFTF LOFWOS Benchmark report

Acronyms and Abbreviations

ANL	Argonne National laboratory
ART	Advanced Reactor Technology
CDE	Core Demonstration Experiment
CR	Control Rod
CRP	Coordinated Research Project
CW	Cold Work
DFA	Driver Fuel Assembly
DHX	Dump Heat Exchanger
EFPD	Effective Full Power Days
FFTF	Fast Flux Test Facility
FP	Fission Product (pairs)
GEM	Gas Expansion Module
HEHB	High Enrichment High Burnup (Absorber)
IAEA	International Atomic Energy Agency
ICSA	In-Core Shim Assembly
LMFBR	Liquid Metal Fast Breeder Reactor
LMR	Liquid Metal Reactor
LOFWOS	Loss of Flow without Scram
MOTA	Materials Open Test Assembly
MWd//kg	Megawatt days per kilogram
OD	Outer Diameter
PDS	Plant Data System
PIOTA	Post Irradiation Open Test Assembly
PNNL	Pacific Northwest National Laboratory
PPS	Plant Protection System
PST	Passive Safety Testing
R&D	Research and Development
RTCB	Run to Cladding Breach
SFR	Sodium Fast Reactor
SR	Safety Rod
SS-304	Type 304 Stainless Steel
SS-316	Type 316 Stainless Steel

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1.0 Introduction

As part of the U.S. Department of Energy, Office of Nuclear Energy Advanced Reactor Technology (ART) program, Pacific Northwest National Laboratory (PNNL) is supporting the preparation of an International Atomic Energy Agency (IAEA) Coordinated Research Project (CRP) of one particular passive safety test at the Fast Flux Test Facility, the loss of flow test from 50% power to natural circulation with Gas Expansion Modules (GEMs).

In July 1986, a series of unprotected transients were performed in FFTF as part of the Passive Safety Testing (PST) program. Among these were thirteen unprotected (with the plant protection system intentionally disabled) loss of flow without scram (LOFWOS) tests. The goals of this program included confirming the safety margins of FFTF as a liquid metal reactor, providing data for computer code validation, and demonstrating the inherent and passive safety benefits of its specific design features.. During the first set of LOFWOS tests, the primary loop pony motors were left on to maintain flow rates of approximately 9% of the nominal full flow. These tests were initiated at 10%, 20%, 30%, 40%, and 50% of the nominal power level. The pony motors were turned off for the second set of tests, allowing the primary system to transition to natural circulation flow rates without pony motor assistance. These tests were run at 10%, 20%, 30%, 40%, 45% and 50% of the nominal power level.

The test selected for the IAEA benchmark is LOFWOS Test #13, which was initiated at 50% power and full flow with the primary pump pony motors turned off. The benchmark specification is intended to support international collaborative efforts on the validation of simulation tools and models in the area of Sodium Fast Reactor (SFR) safety.

Data from additional Passive Safety Tests at FFTF were deemed useful by Argonne National Laboratory (ANL) to verify the operation of the SASSYS code for the benchmark. Previously, data from six additional LOFWOS tests were provided to ANL, but information on other tests that separated the reactivity feedback contributions, the Cycle 8A tests, were also requested.

During February and March of 1986 an extensive series of physics-related experiments was conducted in Cycle 8A at the FFTF to support the Passive Safety Test (PST) program. The primary purpose of the tests was to determine the magnitude and source of temperature reactivity feedback effects existing in the reactor. This test series was part of a national effort to develop a reactor design that could accommodate a large range of accident conditions without relying on special engineered safety components or continuous electrical power to operate coolant pumps. The PST tests in Cycle 8A consisted of 198 measurements of control rod positions at selected power and coolant conditions. The reactor power was varied between 10% and 100% while coolant conditions covered a range of 67% to 100% flow rate and 577°F to 680°F core inlet temperatures. All reactor plant conditions during the test series remained within existing reactor operational limits. The magnitude of the associated temperature reactivity feedbacks between test states was determined by converting rod movement to reactivity. These data can be used in nuclear reactor analysis codes to compare measured to predicted reactivity feedback effects to verify detailed reactor models for moderate-sized liquid metal reactors.

This report is the latest in a series of status reports documenting efforts to prepare and support the Cycle 8C benchmark specification. Reference 1 was the initial draft of the specifications. Reference 2 was a modified specification that took the first steps to convert the benchmark specification for international use. Reference 3 was a joint PNNL-ANL (Pacific Northwest National Laboratory-Argonne National laboratory) draft benchmark specification. Reference 4 described work done with ANL to update the model of the specific FFTF PST benchmark test in SASSYS, comparable results of six other LOFWOS tests, and investigated timing and thermocouple calibration issues. It also described PNNL support for using this benchmark as an IAEA CRP, such as reviewing the draft of the benchmark report, and travel to IAEA in Vienna to plan the IAEA CRP proposal. Reference 5 is a revised benchmark specification based on feedback from the trip to the IAEA and feedback from the benchmark participants.

This report describes efforts in FY19 to:

1. Continue to work with ANL in their efforts to model another series of FFTF passive safety tests in SASSYS
    a. Provided ANL with data from all steps in the reactivity feedback measurements during FFTF Cycle 8A
2. Continue support of FFTF LOFWOS benchmark as an IAEA CRP
    a. Reviewed and commented on the Benchmark report
    b. Investigated comments from participants on Benchmark report
    c. Contributed to ANL issuing Revision 1 to the Benchmark specification

Section 2 contains a description of the Cycle 8A steady-state reactivity feedback measurements. Section 3 summarizes the work completed in FY19 to provide ANL with information and data on those measurements. Section 4 and Appendix A describe the kick-off meeting at the IAEA on the CRP for the FFTF LOFWOS Test. Section 5 describes support provided for the use of the FFTF LOFWOS Test as an IAEA CRP. Section 6 describes future work.

2.0 Description of the Cycle 8A Reactivity Feedback Tests

During February and March of 1986 an extensive series of physics-related experiments was conducted in Cycle 8A at the FFTF to support the Passive Safety Test (PST) program [6]. The primary purpose of the tests was to determine the magnitude and source of temperature reactivity feedback effects existing in the reactor. These feedbacks occur due to neutron balance changes (production versus loss) in response to changing reactor temperature conditions. This test series was part of a national effort to develop a reactor design that could accommodate a large range of accident conditions without relying on special engineered safety components or continuous electrical power to operate coolant pumps.

The specific goals of the Cycle 8A tests were to:

- Measure the reactivity feedbacks associated with the temperature of the fuel material
- Identify and quantify all operational feedback mechanisms
- Provide data to evaluate techniques for separating individual and groups of reactivity feedback effects
- Provide data to verify the steady-state reactivity feedback portion of the FFTF transient prediction models.

The PST tests in Cycle 8A consisted of 198 measurements of control rod positions at selected power and coolant conditions. The reactor power was varied between 10% and 100% while coolant conditions covered a range of 67% to 100% flow rate and 577°F to 680°F core inlet temperatures. All reactor plant conditions during the test series remained within existing reactor operational limits. The magnitude of the associated temperature reactivity feedbacks between test states was determined by converting rod movement to reactivity. Additional measurements were conducted before and during the test series to provide accurate rod worth information for post-test conversion of rod movement data to reactivity. These data can be used in nuclear reactor analysis codes to compare measured to predicted reactivity feedback effects to verify detailed reactor models for moderate-sized liquid metal reactors.

The primary sodium coolant flow rate was required to be within the range of 67% to 100% of full flow. Operation at flow rates below 67% would interfere with a Plant Protection System (PPS} low-flow trip and could produce excessive resonant vibrations in the pumps at rates near 61%. Also, the safety analyses for FFTF did not address extended plant operation under such conditions.

Another limitation on the tests was that associated with the allowable range on the Heat Transport System temperatures, primarily the secondary loop cold leg temperatures. The limits on these temperatures accepted for this test program were <640°F and >560°F. If these limits were exceeded slightly, no important plant limits would be violated; however, it would not be possible to use the automatic secondary cold leg temperature controllers. The temperatures were adjusted so that they approached to within 5°F of each limit. This gave as wide a temperature range to the data as was practical. In several instances more than one test type was begun at the same sequence step. In such cases, the data collection and plant adjustment requirements of both test types were required to be met. These possible complications were accepted in designing the sequence because they substantially reduced testing time.

During normal operation of FFTF, the vertical positions of the six control rods were maintained within 0.5 in. of each other. There was an operational limit that the position of any two control rods may not differ by more than 2.5 in. During these tests the reactor was maneuvered between each of the steps using only secondary control Rod #8. This permitted a more accurate interpretation of the core reactivity change between the states. The small reactivity changes were made possible by periodically repositioning the control rod bank at the unnumbered steps. These positioning steps were performed after the final data had been obtained in the preceding step. The control rods were rebanked with control Rod #8 positioned above the bank by the indicated amount.

The tests in Cycle 8A can be categorized into seven types:

1. **Fuel Temperature --** The reactor inlet and outlet temperatures were held constant while the reactor power and coolant flow rate were varied. The inlet temperature was controlled by adjustments to the heat removal rates of the secondary coolant loops. The outlet temperature was maintained by keeping the power-to-flow ratio constant. Obviously, to first order, the temperatures of the coolant and structural materials in the reactor do not change in this measurement. Any reactivity effect observed must then be a direct result of changes in the temperatures of the fuel material. The data obtained in this measurement can be used to quantify the positive reactivity effect expected from the fuel material during a loss-of-flow incident. In such an incident it is necessary for the power level to drop because the heat removal capability has been substantially reduced. As the power drops the fuel will cool, adding reactivity. The data can also be used to subtract out possible fuel feedback effects from subsequent measurements of structural feedbacks. A measurement was performed at the beginning and end of each series. The results should be identical (within the reproducibility of the reactor state) since they are mirror images of each other. It is possible that the fuel may have restructured during the higher power series and thus changed the fuel temperatures during the second measurement. One of the purposes of the test was to assess such changes. In each series, the power-to-flow ratio was reduced by a relative factor of 0.67, corresponding to the allowable flow range. The approximate fuel temperature range covered was from 2441°F down to 980°F. These tests were all performed at the highest coolant inlet temperature possible to more closely represent the final conditions of a loss-of-flow incident.

2. **Structural Effects --** Constant Average Coolant Temperature (Tavg): The temperature of the fuel pin columns was held constant by keeping both the reactor power level and the axially averaged coolant temperature constant while the flow rate was increased. The change in power-to-flow ratio was accommodated by increasing the core coolant inlet temperature. This test generates feedbacks from all structural feedback components in the reactor because both inlet and outlet temperatures change. Holding the average coolant temperature constant tends to reduce the contribution from radial expansion of the core. The large change in the temperature drop across the core, however, enhances the contribution from assembly bowing. Two measurements of this type were made in each test series. The initial inlet temperatures for the first test in each series were selected so that the inlet temperature reached after the flow increased corresponded to the highest inlet temperature possible for the series. The changes in the inlet temperature ranged from 48°F down to 11.2°F for initial power-to-flow ratios of 0.96 down to 0.22, respectively. The initial inlet temperature for the second test in each series corresponded to the lowest acceptable secondary loop cold leg temperature (>560°F). This second test attempted to detect and quantify a suspected dependence of the bowing reactivity effect on inlet temperature.

3. **Structural Effects --** Constant Outlet Coolant Temperature (Tout): This measurement is very similar to Test Type 2 in that an attempt was made to eliminate any fuel feedbacks. As the flow rate was increased (or decreased) the core outlet temperature was held constant. Some change in reactor power was necessary to maintain the fuel temperatures constant. As in Test Type 2, only structural feedbacks are observed. Holding the core outlet temperature constant eliminates expansion of the control rod driveline and any radial expansion at the top of the core. Bowing certainly contributes but the expansion effect of the core support plate is readily observed. A single measurement of this type was performed in each series and was initiated at the selected highest outlet temperature. The power-to-flow ratio increased during the measurement while attempting to maintain the calculated fuel temperature. In the two highest power-to-flow ratio series the lowest flow rates attainable were greater than 67% because the minimum allowable secondary cold leg temperatures were reached.

4. **Temperature Coefficient --** The temperature of the coolant entering the reactor core was varied in this measurement while holding the reactor power level and flow rate constant. All components in the reactor experience a uniform temperature increase. The major feedbacks come from uniform radial expansion. The contribution from assembly bowing is small because the temperature gradients across the ducts remain nearly constant. There may be some changes in the gaps at the load planes, however, and this could cause a small bowing effect. The fuel effects are small because the fuel temperatures are most sensitive to changes in reactor power. The temperature coefficient was measured many times in each series. A prime consideration in selecting the states was that data be obtained at the initial and final conditions of Test Types 1, 2, and 3. It was sometimes necessary to break a measurement into two parts to avoid excessive changes in the nuclear instruments calibration. Measurements were made at both 100% and 67% flow, where possible. In all series, continuous full-flow temperature coefficient data over the maximum temperature range encountered in the series were obtained.

5. **Flow Coefficient --** The flow rate of the coolant entering the reactor core was varied in this measurement while the reactor power level and the coolant inlet temperature were held constant. The major reactivity feedbacks in this test come from assembly bowing because the power-to-flow ratio changes dramatically. There are contributions from all components except the core support plate. Flow coefficient measurements covering all combinations of the highest and lowest flow rates and inlet temperatures at the minimum power level of each series were specified. Two identical ones were made near the initial power level at the beginning and end of each series. A single test at the initial condition of the controlled transient was also made. Besides the previously mentioned corrections, these data provide the capability to rapidly and easily assess the impact of future modifications to FFTF to enhance its inherent safety characteristics. They are particularly sensitive to changes in the bowing characteristics of the reactor. This measurement was initiated at 75% flow and at a power-to-flow ratio noticeably less than one. The initial flow rate was chosen because the response or speed of the flow controller mechanism increases with increased flow rate. The low power-to-flow ratio was deemed desirable from a plant safety point of view since the power increases when the flow increases. It was expected that the power-to-flow ratio would

decrease, however. The flow change transient that this test statically represents was initiated by a flow rate decrease from ~85% flow. The data obtained in this test are equally applicable to such a sequence.

6. **Controlled Transient --** This measurement statically simulated a transient that was conducted later in Cycle 8B in which the coolant flow rate was rapidly reduced without any control rod movement. The reactor power was allowed to seek a new level to compensate for the reactivity change caused by the associated change in coolant temperatures. This compensation was obtained by changing the fuel temperatures. All feedback components contributed in this measurement. However, the sign of the fuel feedbacks was opposite that of most of the other components because they were compensating. This test was a small-scale static demonstration of the kinds of compensating effects that will occur in a Loss-of-Flow incident.

7. **Power Coefficient --** The reactor power level was changed in this measurement while the coolant inlet temperature and the flow rate were held constant. The dominating feedbacks come from the fuel temperature changes. The power coefficients measured in each series covered the power, flow, and inlet temperature range with one duplicate measurement. The duplicate can detect any changes in the reactor over the test period and provide a measure of the reproducibility of the reactor.

3.0 Information Provided to ANL

In addition to the description of the Cycle 8A Steady-State Reactivity Feedback Measurements given in Section 2, other information on the measurements were provided to ANL. This information included:

- A complete description of the Cycle 8A core loading.
- Axial details (constituents and dimensions) of components in test assemblies that were not in the Cycle 8C loading. Similar information for assemblies irradiated in Cycle 8C were previously documented [1].
- Compositions of all assemblies in the Cycle 8A loading used for neutronic analyses
- Documented data for the first 10 steps of the Cycle 8A reactivity feedback measurements from an analysis of those tests [7]. These data include, for each step: reactor power, primary flow, inlet temperatures, control rod positions and reactivity changes due to control rod movements and fuel burnup.
- A spreadsheet file for each of the first 10 measurement steps containing data recorded by the FFTF PDS.
- A table of documented data for all steps of the Cycle 8A reactivity feedback measurements [7]. These data include, for each step: reactor power, primary flow, inlet temperatures, control rod positions and reactivity changes due to control rod movements and fuel burnup.

After reviewing the transmitted information and spreadsheets, ANL had questions, and the answers are provided in Section 5.3 .

4.0 First IAEA Meeting (CRP) on FFTF Benchmark

A first Research Coordinating Meeting (RCM) on the new coordinated research project (CRP) on ”Benchmark analysis of FFTF Loss of Flow Without Scram” was held at the IAEA headquarters in Vienna, Austria on October 22-25, 2018. Participants are listed in Figure 4.1.

Figure 4.1 Participants in First IAEA CRP Meeting

Participants

Participating Organizations

Name	*Country*	*Organization*
Ms Xiaoyan YANG	China	CIAE
Mr Jun YANG	China	CIAE
Mr Jin WANG	China	INEST
Mr Siyu LYU	China	NCEPU
Ms Dalin ZHANG	China	XJTU
Mr Suizheng QIU	China	XJTU
Mr Antoine GERSCHENFELD	France	CEA
Mr Emil Fridman	Germany	HZDR
Mr Vincenzo Anthony DI NORA	Germany	HZDR
Mr Mattia Massone	Germany	KIT
Mr Natesan Kumaresan	India	IGCAR
Mr Anuj Trivedi	India	ISSSA
Mr Alessandro Petruzzi	Italy	NINE
Mr Domenico De Luca	Italy	NINE
Mr Fabio Giannetti	Italy	Sapienza Uni of Rome
Mr Norihiro Doda	Japan	JAEA
Mr Kazuya Ohgama	Japan	JAEA
Mr Jae-Ho Jeong	Korea	KAERI
Mr Fabio Alcaro	Netherlands	NRG
Mr Sergey Tsaun	Russia	IBRAE
Mr Andrey Volkov	Russia	IPPE
Mr Francisco Alavarez Velarde	Spain	CIEMAT
Mr Janne Wallenius	Sweden	KTH
Ms Sara Bortot	Sweden	KTH
Mr Stefan Radman	Switzerland	EPFL
Mr Konstantin Mikityuk	Switzerland	PSI
Mr Tyler Sumner	United States	ANL
Mr Anton Moisseytsev	United States	ANL
Mr Joseph Kelly	United States	NRC
Mr David Wootan	United States	PNNL
Mr Rodolfo Vaghetto	United States	TAMU
Mr Yassin Hassan	United States	TAMU
Mr Olumuyiwa Omotowa	United States	TerraPower

IAEA

Name		
Mr Dohee Hahn	Director, NENP	IAEA
Mr Vladimir Kriventsev	Scientific Secretary	IAEA
Mr Chirayu Batra	Co-Scientific Secretary	IAEA
Mr Mikhail Khoroshev	Fast Reactor Team	IAEA

Administrative Support

Name		
Ms Khurshida Abdurasulova	Team Assistant, NPTDS	IAEA

The Agenda for the meeting is presented in Appendix A. An introduction to the CRP was made by the IAEA, followed by a presentation on the details of the test. Each participant organization provided an overview of their plans for analysis of the benchmark. A detailed discussion of the CRP workplan concluded the meeting.

The CRP is being implemented as a programmatic activity of IAEA Project 1.1.5.3 "Advanced technology for fast and gas-cooled reactor" starting with the IAEA Program and Budget Cycle 2018 – 2019. Project 1.1.5.3 has the prime objectives to enable Member States to make informed decisions on the development of new or advanced fast reactor designs and to increase cooperation between Member States in achieving advances in fast reactor technology development through international collaborative R&D.

The second Research Coordination Meeting (RCM) for the FFTF CRP, is planned for May, 2020 at IAEA headquarters in Vienna, Austria.

5.0 Support of FFTF Benchmark as IAEA CRP

Argonne National Laboratory issued the benchmark specification report as an ANL document, ANL-ART-102 prior to the first RCM at IAEA HQ. A revision 1 of this document was issued by ANL after the first RCM [5]. Review comments were incorporated into the revision 1 benchmark specification report and files. In addition to the specification report, Excel spreadsheets contained FFTF assembly compositions, FFTF Cycle 8C history, initial conditions for the test, and transient boundary conditions for the test. After Revision 1, additional investigation was completed on several topics.

5.1 Revision 1 Information

Specific information that went into preparing revision 1 were:

- The Exposure of each DFA was apparently computed as the product of cumulative EFPD values and assembly fission powers during Cycle 8C. Data for each of the four DFA types were plotted using a different marker. The plot clearly showed that the FP density in lattice no. 3608 is an outlier, which, as noted, was due to a transcription error.

- There is a good explanation for poor correlation for the type 4.2 DFAs. It was common in FFTF operations to shuffle DFAs to different core locations during refueling outages to achieve testing goals, meet operational requirements and maximize fuel usage. Oftentimes, DFAs were moved inward towards the core center. This was true for the type 4.2 DFAs, where four or the six assemblies were relocated sometime before Cycle 8C, and three of those were moved inboard to higher flux locations. As a result, the MWd values computed for half of the 4.2 DFA were too high. For example, 4.2 DFA 16389 was irradiated for 312 EFPD in row 4 location 3402 before being moved to row 2 location 3201 during the Cycle 8B outage. In Cycles 8B and 8C, it only accumulated another 8 EFPD. If MWd values for the 4.2 DFAs had been computed by integrating assembly EFPDs and fission powers over each cycle of core residency, the correlation would have been good. The other types of DFAs were also shuffled, but not to the extent of the 4.2 DFAs. So, their correlations were not affected as much. However, correlations would undoubtedly have been even higher if the MWd data had been computed by integrating over each cycle of assembly residency. However enough information was not provided to perform such calculations. The fission product (FP) atom densities for the six type 4.2 driver fuel assemblies (DFAs) were inadvertently omitted from Table B-4 in PNNL-25903. The values added by ANL to the "FFTF_Assembly_Compositions" spreadsheet in Table A-3s in the "Driver (Simple)" page are correct.

- The material composition of the GEM orifice region listed in the compositions spreadsheet in the "Other" page, was not computed, but was instead assumed to be the same as the composition of the orifice region for row 7 radial reflectors. This assumption was made because the composition of the GEM orifice region does not affect the results of neutronics calculations.

- The density, thermal conductivity, and heat capacity of the MOX fuel pellet and insulator pellet depends primarily on the burnup. The oxide fuel pellet fractures, densifies, and restructures with accumulated burnup. Values were not provided because the actual values vary with time, and participant fuel performance codes should be able to provide estimates.

- The insulation for the piping was designed to limit heat loss to less than ~60 BTU/hr ft^2 at maximum operating temperatures. The piping and component insulation is spaced and supported from the component surface by the heater support assembly, which is covered by a 0.025 inch stainless steel sheeting. The sheeting forms a 1 inch annulus between the component outer surface and the insulation inner surface to support the heaters and forms a path for leak detection. The insulation is primarily:

    - 28 inch primary hot leg piping, 7 inch blanket of refractory fiber with stainless steel jackets
    - 16 inch primary cold leg piping, 5.5 inch blanket of refractory fiber with stainless steel jackets
    - 16 inch secondary hot leg piping, 7 inch blanket of refractory fiber with stainless steel jackets
    - 16 inch secondary cold leg piping, 5.5 inch blanket of calcium silicate with stainless steel jackets

- The control rod drivelines are located inside a guide tube in the instrument tree that provides protection of the control rod driveline from cross flow. Sodium enters the control rod guide tube at the open-ended bottom of the guide tube, which is above the small gap between the handling socket and the instrument tree. There are two possible flow paths in the guide tube. One path is straight up past the driveline, past the dashpot, and out the top. When the CR is fully withdrawn, the flow exits through the top. When the CR is fully inserted, this flow path is nearly sealed. The other path is out through 3 slots, 1.44 inch x 3 inch that form the first exit.

- Radial alignment of the core assemblies was controlled by the 4.730 inch (12.014 cm) room temperature hole spacing in the core support plate. Hot full power conditions were based on expanding this dimension.

- The pressure values are delta-p, pressure losses, and do not have the atmospheric pressure difference added.

- The 42 inch plenum length and the composition files includes the plenum spring, plenum spacer, and tag gas capsule, everything from the top of axial reflector to top end cap. The plenum spacer is one component in this region that is a cylinder with both ends flattened with a 0.100 inch hole.

- Fuel pins of different enrichment (inner zone versus outer zone fuel) differ in length, as the outer driver enrichment pins have longer lower end caps so they would stick up and be noticeable if they were accidently put in an inner zone assembly. The fuel columns were at the same axial location, relative to core dimensions, in the two pins.

5.2 Discussion of Apparent Discrepancy in FFTF Benchmark Primary Hydraulic Data

After Revision 1 was issued, the following items were investigated and posted to the FFTF CRP website in response to questions:

A question was posed to the CRP coordinators about why the required pump head derived from the analytical pump model seems to differ from the specified total system pressure loss. This discrepancy was investigated and the conclusions are discussed below.

Table 2.3 in the FFTF benchmark specification specifies the FFTF nominal operating conditions at full power; they are correct and appropriate with the following correction/clarification:

> The third entry (identified as "Core Mass Flow Rate") is actually the total reactor vessel volumetric flow rate assuming that the three pumps are operating at their design flow rate (0.9148 m^3/s as specified in Table 6.5). The "Primary Mass Flow Rate per Loop" (2.63×10^6 kg/hr) specified in the seventh line in Table 2.3 is correct for the nominal operating conditions selected for the plant (2.5444 m^3/s total reactor vessel flow rate; ~92.7% of the "design" flow rate). It should also be noted that only about 86% of the total pump/reactor vessel flow passes through the reactor core; the remaining ~14% flows through various bypass paths within the reactor vessel (thus the difference between "Average Core Outlet Temperature" (526.7°C) and "Primary Hot Leg Temperature" (503.3°C) shown in the table.

Table 6.9 specifies the reactor vessel (including the core) and primary loop pressure losses at nominal operating conditions. The values in this table were derived from original plant documentation. An assessment of that data has led to the following two conclusions:

- The core corresponding to the 106.4 psid pressure drop from the original plant documentation included five low flow assembles (three in-core fixed shim assemblies and two material test assemblies). However, the core loading during the LOFWOS testing included only three low flow assemblies (one in-core fixed shim and two material test assemblies). The effect of this core makeup difference is estimated to result in a reduction of approximately 5% in the core pressure drop (at a constant total system flow rate).

- In general, all calculated pressure losses from the original plant documentation are believed to include some amount of conservativism. The magnitude of this conservatism is likely to be on the order of +10% on all calculated values.

Based on the above discussion, it is suggested that the reactor vessel and primary loop pressure loss values currently provided in Table 6.9 of the FFTF benchmark specification be reduced by 10% (except the total primary system and core basket center to outlet plenum should be reduced by 15%) in order to more accurately model the FFTF primary system hydraulics.

The primary pump performance curve and ANL homologous pump model discussed in Section 6.4 of the benchmark were also reviewed again and compared to prototype pump test results. The model and information used appear to be reasonable and are not believed to contribute significantly to the observed discrepancy.

5.3 Cycle 8A Passive Safety Test Information

The following information on the Cycle 8A tests was provided to ANL:

- The reactor was initially at 100% power and 100% flow with an inlet temperature of 680°F, until about 9:17 on 2/17/1986. At that time, power was slowly reduced until 12:34. From then on until the start of the tests, the power varied from 95% to 95.5%. The flow remained at 100% and the inlet temperature remained at 680°F.

- Two columns were added to the first 10 test results. The first new column contains reactivity changes from the initial state due to all control rod movements. The second new column contains cumulative reactivity losses from the initial state due to fuel burnup.

- A description of test assemblies that were new in Cycle 8A were provided.

- CDE stands for Core Demonstration Experiment, which was loaded in Cycle 9A as a heterogeneous region of 10 fuel and 6 blankets. The ACO tests that were in Cycle 8A were lead test assemblies for this later test configuration.

- All the ACOs and FO-2 were irradiated at least one full cycle (122 EFPD) before Cycle 8A. So, the ID of the annular pellets is immaterial since fuel restructuring would have effectively eliminated the central void and the gap between the pellet OD and the cladding ID. Planar smear density inside the fuel pins might be a more useful parameter to consider. The initial inner diameter for annular pellets was added. FO-2 was a lead test for the ACOs. It contained 12 different fuel pellet variations its 169 pins. 86 fuel pins in the inner pin rows were test pins (11 variations). The other 83 fuel pins in the outer rows were filler pins. Parameters are provided for the assembly average pin in addition to the range.

- D9-3 was a RTCB test that in the reload design reports for Cycle 8A was originally going to be replaced with a 3.1 DFA with similar neutronic properties. However later evaluation indicated D9-3 should continue irradiation in Cycle 8A. The flow tables were not changed. Therefore, the flow rate for D9-3 should be the same as Assembly Flow Code 5.

- Average values in addition to a range were provided. HEHB did not have any fuel, but had differing pin diameters, differing axial heights of absorber segments as well as differing enrichments and empty pins, so a detailed description was provided.

- The correct number of Refl-2 assemblies is 15. The reflector assembly in location 1702 was changed to a Refl-2 after the flow distribution table was prepared. A new flow distribution table was not prepared for the final core loading.

- Plenum information for the other assembly types was provided. Comparable plenum information could not be easily located for FC-1, but if important, a good approximation would be to make the ratio of plenum spacer dimensions to clad inner diameter similar to the 169 pin ACO tests.

- Pin pitch information was provided.

- The fuel test assembly regions below the pins are the same as in standard DFAs.
- Information for all Cycle 8A tests was provided to ANL

6.0 Future Work

Potential future work to improve the FFTF LOFWOS Benchmark specification is listed below:

- Continue support for the IAEA CRP "Benchmark Analysis of FFTF Loss Of Flow Without Scram Test"
- Attend and participate in IAEA CRP meeting tentatively scheduled for May, 2019 in Vienna, Austria
- PNNL efforts to support this project should include coordination and planning of the second research coordination meeting (responding to the participants questions and revising the benchmark specifications as necessary), supporting ANL in performing the analysis of the benchmark test, and supporting ANL in analyzing other Passive Safety Tests that could potentially affect the benchmark analyses.

7.0 References

1. D. W. Wootan, A.M. Casella, and J.V. Nelson, *Benchmark Specifications for FFTF LOFWOS Test for Safety Analysis*, PNNL-24752, September 2015.
2. D. W. Wootan, A.M. Casella, and J.V. Nelson, *Benchmark Specifications for FFTF LOFWOS Test for International Use*, PNNL-25903, September 2016
3. T. Sumner, A. Moisseytsev, F. Heidet, D. W. Wootan, A.M. Casella, and J.V. Nelson, *Benchmark Specifications for FFTF LOFWOS Test #13*, ANL-ART-102, December 2017
4. J.V. Nelson and D. W. Wootan, *Fast Flux Test Facility Benchmark Status for FY18*, PNNL-27970, September 2018
5. T. Sumner, A. Moisseytsev, F. Heidet, D. W. Wootan, A.M. Casella, and J.V. Nelson, *Benchmark Specifications for FFTF LOFWOS Test #13*, ANL-ART-102, Rev. 1, December 2017
6. Lucoff, D. M., *Passive Safety Testing at the Fast Flux Test Facility*, WHC-SA-0046-FP, 1987, Westinghouse Hanford Company, Richland, Washington.
7. Knutson, B. J. and R.A. Harris, 1989, *FFTF Inherent Safety Tests: Results of Cycle 8A Steady-State Reactivity Feedback Measurements*, WHC-EP-0117, Westinghouse Hanford Company, Richland, Washington

APPENDIX A

Agenda of 1st RCM on the CRP on Benchmark Analysis of FFTF Loss of Flow Without Scram Test

IAEA
International Atomic Energy Agency

1st Research Coordination Meeting (RCM)

of the

Coordinated Research Project (CRP)

on

Benchmark Analysis of FFTF Loss of Flow Without Scram Test

IAEA Headquarters

Vienna, Austria,
Room M3 (M-Building)

22–25 October 2018

~~Ref. No.: I32011-CR-1; EVT1703133~~

AGENDA

Day 1, Monday, 22 October 2018				
Start	*End*	*Topic*	*Speaker*	*Duration*
14:00	14:10	Welcome Address	Mr Dohee Hahn, IAEA	0:10
14:10	14:25	Self-Introduction of the participants	All	0:15
14:25	14:30	Nomination of the Chairperson	All	0:05
14:30	14:40	Discussion and Approval of the Agenda	All	0:10
14:40	**14:40**	**Session I: CRP Overview and Benchmark Specifications**		
14:40	15:00	CRP content, objectives, participants, expected outcomes and preliminary planning	Mr Vladimir Kriventsev, IAEA	0:20
15:00	***15:20***	***Coffee Break***		***0:20***
15:20	16:30	Presentation by ANL/PNNL on - ULOF experiment description - Benchmark Technical Specifications - Detailed core model description - Simplified homogeneous model - etc.	ANL/PNNL	1:10
16:30	***16:50***	***Coffee Break***		***0:20***
16:50	17:20	Presentation by ANL/PNNL on - Available inputs and expected outputs - Format of the results to be submitted - etc.	ANL/PNNL	0:30
17:20	18:00	Preliminary discussion on Benchmark Technical Specifications, inputs and outputs	All	0:40
18:00		**End of Day 1**		
18:00		***Catered reception sponsored by IAEA***		

Day 2, Tuesday, 23 October 2018				
Start	*End*	*Topic*	*Speaker*	*Duration*
9:30	9:30	**Session II: Participating Organizations Presentations**		
9:30	11:10	20 Mins Presentation by each participating organization o One/two slide on organization's existing capability (codes) to perform the required calculations o One/two slide on proposed FFTF model o One/two slides on draft technical specifications (distributed before meeting) and expected challenges and issues	1. Ms Xiaoyan YANG, CIAE, China 2. Mr Jin WANG, INEST, China 3. Mr Siyu LYU, NCEPU, China 4. Ms Dalin ZHANG, XJTU, China 5. Mr Antoine GERSCHENFELD, CEA, France	1:40
11:10	11:30	*Coffee Break*		0:20
11:30	12:30	20 Mins Presentation contd.	6. Mr Emil Fridman, HZDR, Germany 7. Mattia Massone, KIT, Germany 8. Mr Natesan Kumaresan, IGCAR, India	1:00
12:30	14:00	*Lunch Break*		1:30
14:00	15:50	20 Mins Presentation contd.	9. Mr Anuj Trivedi, ISSSA, India 10. Mr Alessandro Petruzzi, NINE, Italy (**30 mins**) 11. Mr Fabio Giannetti, Sapienza Uni of Rome, Italy 12. Mr Norihiro Doda, JAEA, Japan 13. Mr Jae-Ho Jeong, KAERI, Korea, Rep. of	1:50
15:50	16:10	*Coffee Break*		0:20
16:10	17:10	20 Mins Presentation contd.	14. Mr Fabio Alcaro, NRG, Netherlands 15. Mr Sergey Tsaun, IBRAE, Russia 16. Mr Andrey Volkov, IPPE, Russia	1:00
17:10	17:30	Discussion on day 2		0:20
17:30		**End of Day 2**		

Day 3, Wednesday, 24 October 2018				
Start	*End*	*Topic*	*Speaker*	*Duration*
9:30	9:30	**Session II: Participating Organizations Presentations contd.**		
9:30	11:10	20 Mins Presentation by each participating organization o One/two slide on organization's existing capability (codes) to perform the required calculations o One/two slide on proposed FFTF model o One/two slides on draft technical specifications (distributed before meeting) and expected challenges and issues	17. Mr Francisco Alavarez Velarde, CIEMAT, Spain 18. Mr Janne Wallenius, KTH, Sweden 19. Mr Stefan Radman, EPFL, Switzerland 20. Mr Konstantin Mikityuk, PSI, Switzerland 21. Mr Anton Moisseytsev, ANL, United States	1:40
11:10	11:30	*Coffee Break*		0:20
11:30	12:30	20 Mins Presentation contd.	22. Mr Joseph Kelly, NRC, United States 23. Mr David Wootan, PNNL, United States 24. Mr Rodolfo Vaghetto, TAMU, United States	1:00
12:30	14:00	*Lunch Break*		1:30
14:00	14:20	20 Mins Presentation contd.	25. Mr Olumuyiwa Omotowa, TerraPower, United States	0:20
14:20	14:50	EBR-II experience on application of UQ methodology	Mr Dominico De Luca, NINE, Italy	0:30
14:50	14:50	**Session III: Discussion, CRP Workplan, finalization of input requirements, task distribution, final action plan**		
14:50	15:50	Detailed discussion on Benchmark Technical Specifications, inputs and outputs	All	1:00
15:50	16:10	*Coffee Break*		0:20
16:10	17:10	Discussion on CRP Phases, preliminary workplan and action plan, establishing internal working groups	All	1:00
17:10	17:30	Description of the SharePoint portal to be used	Mr Chirayu Batra, IAEA	0:20
17:30		**End of Day 3**		

Day 4, Thursday, 25 October 2018				
Start	***End***	***Topic***	***Speaker***	***Duration***
9:30	9:30	**Session III: CRP Workplan, finalization of input requirements, task distribution, final action plan**		
9:30	11:00	CRP Workplan, finalization of input requirements, task distribution, final action plan	All	1:30
11:00	11:15	***Coffee Break***		0:15
11:15	12:15	Drafting of RCM report	Meeting Chair	1:00
12:15	12:30	Closing Remarks	IAEA	0:15
12:30	14:00	***Lunch Break***		1:30
14:00		**End of Day 4**		
		End of Meeting		

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