Irradiation Testing Vehicles for Fast Reactors from Open Test Assemblies to Closed Loops

Nuclear Engineering Division

About Argonne National Laboratory

Argonne is a U.S. Department of Energy laboratory managed by UChicago Argonne, LLC under contract DE-AC02-06CH11357. The Laboratory's main facility is outside Chicago, at 9700 South Cass Avenue, Argonne, Illinois 60439. For information about Argonne and its pioneering science and technology programs, see www.anl.gov.

DOCUMENT AVAILABILITY

Online Access: U.S. Department of Energy (DOE) reports produced after 1991 and a growing number of pre-1991 documents are available free via DOE's SciTech Connect (<http://www.osti.gov/scitech/>)

Reports not in digital format may be purchased by the public from the National Technical Information Service (NTIS):

U.S. Department of Commerce
National Technical Information Service
5301 Shawnee Rd
Alexandria, VA 22312
www.ntis.gov
Phone: (800) 553-NTIS (6847) or (703) 605-6000
Fax: (703) 605-6900
Email: **orders@ntis.gov**

Reports not in digital format are available to DOE and DOE contractors from the Office of Scientific and Technical Information (OSTI):

U.S. Department of Energy
Office of Scientific and Technical Information
P.O. Box 62
Oak Ridge, TN 37831-0062
www.osti.gov
Phone: (865) 576-8401
Fax: (865) 576-5728
Email: **reports@osti.gov**

Disclaimer

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor UChicago Argonne, LLC, nor any of their employees or officers, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of document authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof, Argonne National Laboratory, or UChicago Argonne, LLC.

Irradiation Testing Vehicles for Fast Reactors from Open Test Assemblies to Closed Loops

prepared by
James J. Sienicki and Christopher Grandy
Nuclear Engineering Division, Argonne National Laboratory

December 15, 2016

ABSTRACT

A review of irradiation testing vehicle approaches and designs that have been incorporated into past Sodium-Cooled Fast Reactors (SFRs) or envisioned for incorporation has been carried out. The objective is to understand the essential features of the approaches and designs so that they can inform test vehicle designs for a future U.S. Fast Test Reactor. Fast test reactor designs examined include EBR-II, FFTF, JOYO, BOR-60, PHÉNIX, JHR, and MBIR. Previous designers exhibited great ingenuity in overcoming design and operational challenges especially when the original reactor plant's mission changed to an irradiation testing mission as in the EBR-II reactor plant. The various irradiation testing vehicles can be categorized as: Uninstrumented open assemblies that fit into core locations; Instrumented open test assemblies that fit into special core locations; Self-contained closed loops; and External closed loops. A special emphasis is devoted to closed loops as they are regarded as a very desirable feature of a future U.S. Fast Test Reactor. Closed loops are an important technology for irradiation of fuels and materials in separate controlled environments. The impact of closed loops on the design of fast reactors is also discussed in this report.

Table of Contents

Abstract	ii
Table of Contents	iii
List of Figures	v
List of Tables.....	vii
1 Introduction.....	8
2 EBR-II Test Vehicles	9
2.1 Introduction	9
2.2 Instrumented Subassembly Test (INSAT)	12
2.3 In-Core Instrument Test (INCOT)	18
2.4 Breached Fuel Test Facility (BFTF)	24
2.5 Fuel Performance Test Facility (FPTF)	28
2.6 Large Diameter Irradiation Test Assembly (LITA)	33
2.7 (Nuclear Instrument Test Facilities (NITF)	34
2.8 Uninstrumented Test Assemblies.....	39
2.9 References	42
3 Fast Flux Test Facility (FFTF) Test Vehicles	43
3.1 Introduction	43
3.2 Fuels Open Test Assembly (FOTA)	43
3.3 Post-Irradiation Open Test Assembly (PIOTA).....	45
3.4 Vibration Open Test Assembly (VOTA)	46
3.5 Absorber Open Test Assembly	46
3.6 Materials Open Test Assembly (MOTA).....	46
3.7 Closed Loops and Closed Loop In-Reactor Assemblies.....	49
3.8 FFTF Test Assembly Locations	56
3.9 References	57
4 JOYO Test Vehicles.....	58
4.1 Introduction	58
4.2 Un-instrumented Test Assemblies	61
4.3 Instrumented Test Assembly (INTA).....	64
4.4 Upper Core Structure Irradiation Plug Rig (UPR).....	65
4.5 Materials Testing Rig with Temperature Control (MARICO)	65
4.6 Ex-Vessel Irradiation Rig (EXIR).....	68
4.7 References	69
5 BOR-60 Closed Loops	70
5.1 Introduction	70
5.2 Independent Lead Channel (ILC).....	70
5.3 Sodium Closed Loops	70
5.4 References	74
6 PHÉNIX Closed Loop Concept	75
6.1 Introduction	75
6.2 BAUPHIX Closed Sodium Loop	75
6.3 References	78
7 Jules Horowitz Reactor (JHR) Closed Loops	79
7.1 Introduction	79
7.2 CALIPSO Closed NaK Loop	85

7.3	Closed Water Loops Inside of the Reflector	88
7.4	MADISON Closed Water Loop	88
7.5	ADELIN Closed Water Loop	88
7.6	LORELEI Closed Water Loop	89
7.7	CLOE Closed Water Loop	89
7.8	Comparison of JHR Closed Loops	97
7.9	References	99
8	Comparison of In-Reactor Closed Loops	101
8.1	Discussion	101
8.2	References	107
9	Impact of Closed Loops on Reactor Plant Design	107
10	Summary	119
	Acknowledgements	119

LIST OF FIGURES

Figure 1. Plan View of EBR-II Small Rotating Plug Showing Control Rod Locations.	10
Figure 2. Illustration of EBR-II Control Rod and Control Rod Drive.	11
Figure 3. Illustration of INSAT Installation.	13
Figure 4. Schematic Illustration of INSAT.	14
Figure 5. Extension Tube from Top of Fuel Assembly to Terminal Box.	15
Figure 6. Typical INSAT Experiment.	16
Figure 7. Illustration of Extension Tube, Drywell Liner Tube, and Instrument Leads.	17
Figure 8. Illustration of INCOT Installation.	19
Figure 9. Elevation View of INCOT.	20
Figure 10. Upper Portion of INCOT Thimble for Model 2 Sensor Test Assembly.	21
Figure 11. Middle Portion of INCOT Thimble for Model 2 Sensor Test Assembly.	22
Figure 12. Lower Portion of INCOT Thimble for Model 2 Sensor Test Assembly.	23
Figure 13. Illustration of BFTF.	25
Figure 14. Breached Fuel Test Facility.	26
Figure 15. Typical BFTF Deposition Sampler.	27
Figure 16. Illustration of Fuel Performance Test Facility.	29
Figure 17. Mark R-19C Subassembly.	30
Figure 18. Mark R-19D Subassembly.	31
Figure 19. Mark R-61A Subassembly.	32
Figure 20. Illustration of LITA.	33
Figure 21. Schematic Illustration of J-2 and O-1 NITFs.	35
Figure 22. Locations of O-1 and J-2 Thimbles.	36
Figure 23. Schematic Illustration of O-1 Instrument Thimble.	37
Figure 24. Schematic Illustration of J-2 Instrument Thimble.	38
Figure 25. Illustration of Fuels Open Test Assembly (FOTA).	44
Figure 26. Illustration of Post-Irradiation Open Test Assembly (PIOTA).	45
Figure 27. Illustration of Materials Open Test Assembly (MOTA).	47
Figure 28. MOTA Canisters.	48
Figure 29. Neutron Flux Distribution Along Height of MOTA Canisters.	48
Figure 30. MOTA-1 Series and MOTA-2 Series Canister Configurations and Positions in FFTF Core.	49
Figure 31. Illustration of FFTF Individual Closed Loop System.	50
Figure 32. Illustration of FFTF Closed Loop In-Reactor Assembly (CLIRA).	51
Figure 33. Schematic Illustration of FFTF Closed Loop Primary Sodium System.	52
Figure 34. Schematic Illustration of FFTF Closed Loop Intermediate Sodium System.	52
Figure 35. Illustration of FFTF Closed Loop Primary Sodium System.	53
Figure 36. FFTF Closed Loop Primary Sodium System Turned on Its Side.	53
Figure 37. Closed Loop System Sodium-to-Air Heat Exchanger (Dump Heat Exchanger).	54
Figure 38. Closed Loop In-Reactor Assembly (CLIRA).	55
Figure 39. FFTF Test Assembly Locations.	56
Figure 40. JOYO Mk-III Core Layout.	58
Figure 41. Illustration of JOYO.	59
Figure 42. Different Test Vehicles Available for Irradiations in JOYO.	60
Figure 43. Four Types of Un-instrumented Fuel Irradiation Rigs Providing Differences in Environment.	61
Figure 44. Un-instrumented Capsule Fuel Irradiation Rig.	62
Figure 45. Un-instrumented Materials Irradiation Rigs.	63

Figure 46. Illustration of Instrumented Test Assembly (INTA).	64
Figure 47. Illustration Upper Core Structure Irradiation Plug Rig (UPR).	65
Figure 48. Schematic Illustration of the Materials Testing Rig with Temperature Control (MARICO).	66
Figure 49. Specimens for Installation Inside of Canisters Mounted with Hinges on MARICO Test Assembly.	67
Figure 50. Illustration of MARICO-2 Incident for Which a Problem Was Detected on June 11, 2007.	67
Figure 51. Illustration of MARICO-2 Incident.	68
Figure 52. Illustration of Ex-Vessel Irradiation Rig (EXIR).	69
Figure 53. Independent Lead Channel (ILC) Irradiated in BOR-60 (Dimensions in mm).	71
Figure 54. Plan View of Independent Lead Channel (ILC) Irradiated in BOR-60 (Dimensions in mm).	72
Figure 55. Self-Contained Sodium Closed Loop Concept for BOR-60 with Electromagnetic Sodium Pump.	73
Figure 56. BAUPHIX Closed Sodium Loop Concept for Installation in PHÉNIX (Dimensions in mm).	76
Figure 57. BAUPHIX Closed Sodium Loop Concept for Installation in PHÉNIX (Dimensions in mm).	77
Figure 58. Jules Horowitz Reactor Building and Containment Under Construction in September 2015.	80
Figure 59. Illustration of Jules Horowitz Seismically Isolated Reactor Building and Containment Layout and Reactor Vessel with Surrounding Reflector.	81
Figure 60. Jules Horowitz Reactor Core, Reactor Vessel/Tank, Beryllium Reflector Outside of the Reactor Vessel, and Slot Displacement Systems for Closed Loops.	82
Figure 61. Illustration of LORELEI Closed Loop Test Article on Displacement System Inside Slot in Reflector Inside Water Pool and Supporting Systems Inside of Containment.	83
Figure 62. Illustration of Displacement Device as Well as Reactor Building and Containment Layout.	83
Figure 63. Further Illustration of Reactor Core, Reactor Vessel, and Surrounding Reflector in Water Pool.	84
Figure 64. Plan View of Reactor Core and Reflector.	84
Figure 65. Calculated Neutron Spectra Inside Core and Reflector.	85
Figure 66. Illustration of CALIPSO Test Article.	86
Figure 67. Information on CALIPSO Test Article.	87
Figure 68. CALIPSO Electromagnetic NaK Pump (Left) and Prototype for Testing (Right).	87
Figure 69. Illustration of MADISON Closed Loop.	90
Figure 70. Schematic Illustration of MADISON Closed Loop.	91
Figure 71. Information on MADISON Closed Loop.	92
Figure 72. Information on ADELIN Closed Loop.	93
Figure 73. Illustration of ADELIN Closed Loop.	94
Figure 74. ADELIN Test Article.	94
Figure 75. Illustration of LORELEI Closed Loop.	95
Figure 76. Illustration of CLOE Closed Loop.	96
Figure 77. Illustration of CLOE Sample Holder.	97
Figure 78. Illustration of Mk-III Closed Sodium Loop Utilized in TREAT.	105
Figure 79. Illustration of Instrumented Closed Loop Channel (Left) and External Closed Loop Channel (Right) Envisioned for Use in MBIR.	105

Figure 80. Core Layout Envisioned for MBIR Incorporating Three Locations for Instrumented Closed Loop Channels and Three Locations for External Closed Loop Channels.....	106
Figure 81 – Picture of Fast Flux Test Facility – white dome is FFTF containment dome	108
Figure 82 – Schematic and Elevation View of FFTF Primary and Intermediate Heat Transport Systems	109
Figure 83 – Clinch River Breeder Reactor Plant	110
Figure 84 – Schematic and Elevation View of CRBR Primary and Intermediate Heat Transport Systems	111
Figure 85 – Plan View of CRBR (Left) and FFTF (Right) Reactor Heads	112
Figure 86 - Closed Loop Module	113
Figure 87 – Closed Loop System in FFTF Containment	114
Figure 88 – FFTF Containment	116
Figure 89 – CRBR Containment.....	117

LIST OF TABLES

Table 1. Fuel Pin or Capsule Dimensions for Uninstrumented Test Assemblies	40
Table 2. Mark Designations and Purposes for Uninstrumented Test Assembly Designs.....	41
Table 3. Comparison of JHR Closed Loop Designs and Major Features	97
Table 4. Comparison of Closed Loop Designs and Major Features	103
Table 5 – Fast Flux Test Facility Design Parameters	109
Table 6 – Clinch River Breeder Reactor Plant Characteristics	110
Table 7 – Closed Loop System Components	115
Table 8 – Dose Rates for Activated Sodium and Discharge Spent Fuel Assemblies	117
Table 9 – Comparison of Relative Dose Rates for SFR and MSR Closed Loops	118

1 Introduction

A review of irradiation testing vehicle approaches and designs that have been incorporated into past Sodium-Cooled Fast Reactors (SFRs) or envisioned for incorporation has been carried out. The objective is to understand the essential features of the approaches and designs so that they can inform test vehicle designs for a future U.S. Fast Test Reactor. Fast test reactor designs examined include EBR-II, FFTF, JOYO, BOR-60, PHÉNIX, JHR, and MBIR. Previous designers exhibited great ingenuity in overcoming design and operational challenges. The various irradiation testing vehicles can be categorized as: Uninstrumented open assemblies that fit into core locations; Instrumented open test assemblies that fit into special core locations; Self-contained closed loops; and External closed loops. A special emphasis is devoted to closed loops as they are regarded as a desirable feature of a future U.S. Fast Test Reactor.

The test vehicles of interest from each fast reactor design are presented in sections covering EBR-II, FFTF, JOYO, BOR-60, PHÉNIX, and JHR. Given the emphasis on closed loops, this is followed by a special section that compares the features of different closed loops designs.

2 EBR-II Test Vehicles

2.1 Introduction

Closed loops were never installed in EBR-II. In 1976 and 1977, there was the Safety Research Modification (SRM) project at ANL [1 through 4] that investigated the technical feasibility of modifying EBR-II for extended steady state operation followed by a rapid power burst capable of causing melting of test fuel inside of a central test assembly while maintaining the safe integrity of the surrounding driver fuel. Oxide fuel was examined for both the test and driver fuel. The concept may have included a 271-pin test assembly housed inside of a sodium closed loop. However, the project never materialized.

EBR-II incorporated four different types of instrumented open test assembly facilities [5 through 7] cooled by the primary sodium coolant. Open test assembly facilities were installed in control rod locations. This approach was made possible because EBR-II was originally designed and constructed with twelve control rod locations at which control rods could be installed and removed. However, the worth of individual control rods was increased to be large enough such that all twelve control rods were not required. In part, this reflected the control rod design that incorporated a boron carbide absorber and a fuel follower fuel assembly beneath the absorber section. Ultimately, only six high worth control rods were required to operate EBR-II. This left some of the control rod locations available for instrumented test assemblies. Two to four control rod locations were utilized for instrumented tests with the remaining locations occupied by control rods [5]. Figure 1 shows a plan view of the small rotating plug with the twelve control rod locations. The original control rod and control rod drive configurations are illustrated in Figure 2.

The size of the instrumented test vehicle was limited by the available space inside of each control rod location. Each had to fit through a control rod guide bearing tube having an inner diameter of 6.985 cm (2.75 in) penetrating the small rotating plug and the cover gas above the sodium level as well as a guide bushing with an inner diameter of 6.546 cm (2.577 in) at the bottom of the guide bearing tube [8].

Figure 1. Plan View of EBR-II Small Rotating Plug Showing Control Rod Locations.

Figure 2. Illustration of EBR-II Control Rod and Control Rod Drive.

2.2 Instrumented Subassembly Test (INSAT)

The first instrumented in-core test facilities were Instrumented Subassembly Test (INSAT) facilities basically consisting of an instrumented fuel assembly, an extension tube with a gripper at the lower end, a bellows seal above the small rotating plug, a terminal box above the bellows, and an elevator drive system (Figure 3 and Figure 4). The test assembly could contain fuel or could contain non-fueled capsules with material specimens. Instrumentation for fuel assemblies included flowmeters, thermocouples, flux monitors, and fission gas pressure transducers. The fuel assembly and extension tube (Figure 5 and Figure 6) were installed through the control rod drive location. The extension tube surrounded an inner drywell liner tube; the space between the tubes was for the instrument leads that could be sheathed or individually installed in tubes

(Figure 7). The extension and drywell liner tubes were connected to a terminal box within which the instrumented leads were collected. Up to twenty-three instrument leads were installed. The interior of the terminal box was pressurized with argon to 0.0965 MPa gauge (14 psig) such that a leak in the extension tube or the drywell liner tube would be immediately detected by a drop in the terminal box pressure. Following irradiation, it was necessary to cut the instrument leads between the fuel assembly and the extension tube with a special cutter. This allowed the extension tube to be removed through the control rod guide bearing tube. The fuel assembly remained in the core and was removed with the fuel handling equipment. A special cutting tool was lowered through the extension tube and cutting was performed by rotation of the cutter. The cutter was then withdrawn upward through the extension tube. Prior to installation, the assembly and extension tube were preheated to 316 °C (600 °F) in a special purpose heater to prevent thermal shocking when the assembly and extension tube are inserted into the 304 °C (585 °F) primary sodium and to melt bond sodium inside of fuel pins or capsules from the top downward. The actual insertion included a five hour hold at five feet above the sodium level to allow for equilibration and differential thermal expansion effects of the coupling between the fuel assembly and extension tube.

Figure 3. Illustration of INSAT Installation.

Figure 4. Schematic Illustration of INSAT.

Figure 5. Extension Tube from Top of Fuel Assembly to Terminal Box.

Figure 6. Typical INSAT Experiment.

Figure 7. Illustration of Extension Tube, Drywell Liner Tube, and Instrument Leads.

2.3 In-Core Instrument Test (INCOT)

The next instrumented test facilities were In-Core Instrument Test (INCOT) facilities that contained a test assembly with instruments or materials for irradiation but not fuel. Because of the absence of irradiated fuel and fission products, it was not necessary to cut the instrument leads between the test assembly and the extension tube. Thus, following irradiation, the test assembly could be an integral part of and removed together with the supporting extension tube/thimble through the control rod guide bearing tube into a shielded container. An INCOT consisted of a thimble containing the instruments or materials at its lower end, a bellows seal, a terminal box, an elevator drive, and shielded containers for post-irradiation handling (Figure 8 and Figure 9). A special guide tube was installed in the control rod location to receive the thimble. The instruments or materials needed to fit inside of a 3.556 cm (1.4 in) sensor tube. Instruments or materials were gamma heated and heat was removed by primary sodium. A unique sodium flow control valve at the bottom of the thimble below the instruments or materials could adjust the sodium inlet flow during irradiation thereby controlling the instrument temperatures. Different configurations were designed. For example, Models 1 and 2 contained instruments that were housed in sensor tubes filled with argon or helium. Adjusting the gas composition and pressure was another way of controlling the temperatures of the instruments. In Model 3, the sensors were directly cooled by sodium. Figure 10 through Figure 12 show the Model 2 configuration. In INCOT, the terminal box operating pressure was reduced to 3 to 8 psig.

Figure 8. Illustration of INCOT Installation.

Figure 9. Elevation View of INCOT.

Figure 10. Upper Portion of INCOT Thimble for Model 2 Sensor Test Assembly.

Figure 11. Middle Portion of INCOT Thimble for Model 2 Sensor Test Assembly.

Figure 12. Lower Portion of INCOT Thimble for Model 2 Sensor Test Assembly.

2.4 Breached Fuel Test Facility (BFTF)

The INCOTs were followed by the Breached Fuel Test Facility (BFTF) facilities. Each facility consisted of a Run Beyond Cladding Breach (RCBC) type of fuel assembly housed in a special thimble support tube, a double-walled thimble assembly containing a removable central instrument stalk with sodium flowmeters, thermocouples, and a deposition sampler for plateout of fission products released from fuel, a delayed neutron detector assembly to monitor fuel failure, a bellows seal, a terminal box, and an elevator drive (Figure 13). Instrumentation is not present inside of the fuel assembly such that there are no instrument leads connecting the fuel assembly and thimble. The thimble locks to the support tube surrounding the fuel assembly. This approach was employed to prevent the thimble from diffusion-bonding/self-welding to the fuel assembly and inadvertently lifting the fuel assembly when the thimble is raised. Thus, during refueling, the thimble can be raised to permit refueling operations using the fuel handling equipment. The fuel assembly can be left in the core, if desired, and the thimble reseated atop it when refueling operations are completed. Some of the sodium flowing through the assembly escapes from entering the thimble assembly through the interface between the fuel assembly and the thimble. In particular, 8 % bypasses the thimble under nominal conditions. This was a deliberate design feature such that released fission gas that bypasses the thimble would find its way to the normal EBR-II system that detects and identifies tagged fission gas. The BFTF utilized the existing INCOT elevator drive assembly, bellows seal, and support platform such that the facility was sometimes referred to as INCOT-BFTF.

Figure 13. Illustration of BFTF.

Figure 14. Breached Fuel Test Facility.

Figure 15. Typical BFTF Deposition Sampler.

2.5 Fuel Performance Test Facility (FPTF)

The final instrumented test facilities were the Fuel Performance Test Facility (FPTF) facilities. The facilities consisted of a fuel assembly or an assembly of capsules housed inside of a special guide tube, thimble, instrument carrier, a bellows seal, a terminal box, and an elevator drive (Figure 16). A main feature was a sodium flow control valve inside of the thimble to reduce the sodium flow through the fuel assembly to simulate loss-of-flow as well as other scenarios resulting in temperature transients in the fuel. The fuel assembly is surrounded by a guide tube installed in the control rod location. The lower part of the subassembly locks to the guide tube with a bayonet coupling to prevent inadvertent lifting of the fuel assembly if diffusion bonding/self-welding were to occur at the interface of the fuel assembly and the thimble. The thimble is seated atop the fuel assembly with a relatively leak proof ball-seat coupling fabricated of Inconel 718 designed to minimize the likelihood of diffusion bonding. Instrumentation included flowmeters, a delayed neutron detector, thermocouples, sodium leak detectors, and acoustic microphones to detect sodium boiling. The FPTF also utilized the existing INCOT elevator drive assembly, bellows seal, and support platform such that the facility was sometimes referred to as INCOT-FPTF.

The test assemblies were Mark R subassemblies as discussed below under Uninstrumented Test Assemblies. Examples are shown in Figure 17 through Figure 19. The Mark R-19C subassembly contained nineteen capsules. The space between the two hexcans is filled with gas to insulate the capsules from heat losses to flowing sodium surrounding the outer hexcan thereby enabling the capsules to attain temperatures as high as 649 °C (1200 °F). The Mark R-19D subassembly contained nineteen fuel pins. The Mark R-61A subassembly contained sixty-one fuel pins.

Figure 16. Illustration of Fuel Performance Test Facility.

Figure 17. Mark R-19C Subassembly.

Figure 18. Mark R-19D Subassembly.

Figure 19. Mark R-61A Subassembly.

2.6 Large Diameter Irradiation Test Assembly (LITA)

A concept was developed for a Large Diameter Irradiation Test Assembly (LITA) facility that could irradiate test sections containing fuel or materials up to 13.65 cm (5.375 in) in diameter in seven EBR-II subassembly positions (Figure 20). If it were centered on a control rod location, then the sodium effluent could be collected inside of a thimble and monitored or controlled similarly to a BFTF or FPTF test. EBR-II incorporated a nozzle (the X-nozzle) large enough to accommodate the installation and removal of the assembly. A detailed design was never developed.

Figure 20. Illustration of LITA.

2.7 (Nuclear Instrument Test Facilities (NITF))

For irradiation of instruments in lower fluxes than inside of the reactor core, EBR-II also provided two Nuclear Instrument Test Facilities (NITFs) located outside of the reactor vessel and inside of the primary tank (Figure 21 through Figure 24). Neutron fluxes at the O-1 and J-2 locations were 6.4×10^8 and 8.0×10^8 n/(cm²·s), respectively. Gamma fluxes were 5.5×10^4 and 1.2×10^6 R/hour, respectively. The thimbles were filled with flowing air. The temperature could be controlled by controlling the air flowrate to remove heat from the surrounding primary tank sodium and the instruments. The O-1 facility was smaller and could operate at temperatures between 368 °C (695 °F) and 38 °C (100 °F). The former temperature is representative of sodium in the primary tank. The J-2 facility was larger and could operate at temperatures up to 459 °C (858 °F) through the use of an internal oven-sleeve assembly.

Figure 21. Schematic Illustration of J-2 and O-1 NITFs.

Figure 22. Locations of O-1 and J-2 Thimbles.

Figure 23. Schematic Illustration of O-1 Instrument Thimble.

Figure 24. Schematic Illustration of J-2 Instrument Thimble.

2.8 Uninstrumented Test Assemblies

EBR-II also incorporated uninstrumented test assemblies in driver fuel locations. Many specific designs were developed for a variety of special purpose irradiations of fuels or materials that could be carried out with fuel pins exposed to the primary sodium coolant or using sealed capsules containing materials filled with sodium or gases. The designs were standardized and documented such that other experimenters could take advantage of existing approved and proven designs for new irradiations.

Table 1 indicates forty-five Mark numbers designating the various designs. The meaning of the fourteen Mark designations is shown in

Table 2. The Mark R designs are particular interesting because they were intended for insertion in a guide tube installed in a control rod location and use with a FPTF.

Table 1. Fuel Pin or Capsule Dimensions for Uninstrumented Test Assemblies

Capsule or Element			Spacer Wire			Subassembly Mark No. (underlined models have pressure-pulse protection)
OD (in.)	Length (above grid in.)		Dia. (in.)	Normal Pitch (in.) ^b	Grid Bar Shape	
	Nominal	Max. ^a				
0.174	23.814	24.5	0.049	6	Square	D-1, R-61A, <u>R-19D</u>
0.220	60.593	61.0	0.051	6	Round	B-61
0.230	29.218	29.8	0.042	6	Round	R37B, D-61, D-61A
0.230	39.718	40.0	None	None	Round	<u>L-37A</u> , L-61
0.230	39.718	43.2	0.030	12	Round	<u>R-19E</u>
0.230	39.718	40.0	0.042	12	Round	B-61B
0.230	39.718	40.3	0.056	12	Round	J-37D, R-19C
0.230	39.653	40.0	0.054	6	Round	R-37A
0.230	39.653	40.0	0.058	6	Round	J-37E
0.230	60.593	61.0	0.042	12	Round	B-61A
0.250	39.718	40.1	0.040	6	Round	J-37C, N-37
0.250	60.593	61.0	0.040	6	Round	H-37A
0.250	60.593	61.0	0.062	12	Round	F-37A
0.290	60.593	61.0	0.056	6	Round	B-37, <u>T-19</u>
0.290	24.653	29.8	0.056	6	Round	D-37
0.295	39.718	40.0	0.030	6	Round	N-19A
0.295	39.718	42.0	None	None	Round	A-19D
0.310	39.718	46.5	None	None	Round	A-19B
0.310	39.718	40.1	0.059	6	Round	J-19D, J-19E
0.310	39.718	40.1	0.080	6	Round	N-19
0.370	38.218	40.7	0.100	6	Round	D-19
0.370	39.718	40.5	0.067	6	Round	L-19, L-19A
0.375	39.625	42.1	None	None	Square	A-19A
0.375	45.525	49.1	None	None	Square	A-19C
0.375	39.718	40.4	0.030	6	Square	J-19B
0.375	60.593	61.0	0.097	6	Round	B-19
0.375	47.653	49.0	0.026	6	Square	R-19F
0.778	57.437	60.0	0.020	6	Square	B-7B
0.805	57.437	60.0	None	None	Square	B-7A
Unit						
Assembly	None	None	None	None	None	M-1, M-1A, M-1B
Grid						
Type, etc.	None	None	None	None	None	S-1, S-1A, S-1B

^a Maximum allowable end-of-life element length (includes irradiation growth).

^b Most subassembly designs will accept a complete bundle of elements of any uniform pitch.

Table 2. Mark Designations and Purposes for Uninstrumented Test Assembly Designs

Mark	Purpose
A	Irradiation of capsules inside of dimpled shroud tubes
B	Irradiation of fuel pins or capsules of maximum length
D	Irradiation of structural materials with preheated sodium heated by fuel pins
E	Irradiation of fuel pins or pressurized capsules likely to fail with multiple barriers to prevent failure propagation to adjacent assemblies as well as particle collection devices above and below the core
F	Irradiations investigating different fuel pin and wire wrap dimensions
H	Similar to F but with smaller diameter wire wraps
J	Irradiations at higher mixed mean sodium outlet temperatures up to 649 °C (1200 °F)
L	Irradiation of fuel pins or material specimens under isothermal conditions
M	Irradiation of large structural material specimens up to the available space inside of a hexcan
N	Similar to F or H but with shorter fuel pins
Q	Similar to E and J but with protection against pressure pulses and stainless steel swelling
R	Designed for use with a FPTF in control rod location incorporating a lower adapter with a bayonet opening to lock into the surrounding guide tube when subassembly is rotated
S	Irradiation of fuel pins with spacer grids
T	Similar to B but with pressure pulse protection

2.9 References

1. D. Meneghetti, A. J. Foltman, F. C. Franklin, K. N. Grimm, D. A. Kucera, and H. A. Larson, "Physics Design of an EBR-II Fast-Transient Facility," American Nuclear Society 1976 International Meeting, Washington, D.C., November 14-17, 1976.
2. J. L. Gillette and J. E. Sullivan, "Thermal Analysis of Driver Subassemblies in the EBR-II Transient Facility," American Nuclear Society 1976 International Meeting, Washington, D.C., November 14-17, 1976.
3. A. J. Foltman, F. C. Franklin, K. N. Grimm, and D. Meneghetti, "Physics Design of a Safety Research Modification of EBR-II," American Nuclear Society 1977 Winter Meeting, San Francisco, CA, November 27-December 2, 1977.
4. J. G. Gale and W. K. Lehto, "Seismic Response of the EBR-II to the Mount Borah Earthquake," International Topical Meeting on Fast Reactor Safety, Knoxville, TN, April 21-25, 1985.
5. "EBR-II Division Experiment Guide Part II Second Edition," Rev. 0, September 7, 1990.
6. "Guide for Irradiation Experiments in EBR-II," EBR-II Project, From March 1, 1975 with Updates.
7. J. A. Pardini, R. C. Brubaker, D. J. Veith, G. C. Giorgis, D. E. Walker, and O. S. Seim, "The Experimental Breeder Reactor (EBR-II) Fuel-Performance Facility (FPTF)," American Nuclear Society Fast, Thermal, and Fusion Reactor Experiments Conference, Salt Lake City, UT, April 12-15, 1982.
8. O. Seim, E. Hutter, T. Sullivan, R. Olp, J. Pardini, E. Filewicz, and R. Brubaker, "Conceptual System Design Description of the EBR-II In-Core Instrument Test Facility," ANL/EBR-004, Argonne National Laboratory/EBR-II Project, June 1969.

3 Fast Flux Test Facility (FFTF) Test Vehicles

3.1 Introduction

The Fast Flux Test Facility (FFTF) was designed to incorporate a variety of irradiation testing capabilities including closed sodium loops. A large 400 MWt core with a height of 0.91 m and diameter of 1.20 m was selected that enabled the concurrent incorporation of a large number of test assemblies. To accommodate the closed loops that could remain in the core for up to three cycles, the core layout was selected to locate the closed loops in particular core positions and three fuel handling machines were provided to enable refueling around the closed loops.

Information on the FFTF test vehicles was obtained from References [9 through 14].

3.2 Fuels Open Test Assembly (FOTA)

The Fuels Open Test Assembly (FOTA) shown in Figure 25 was a highly instrumented fuel assembly with an overlying stalk penetrating the upper head and a spool piece above the upper head. The FOTA was open to the primary sodium coolant with the sodium flowrate controlled with an inlet orifice. The sodium exit temperature could be 67 °C (120 °F) greater than that of surrounding driver assemblies. There could be fifty-one leads for temperature, pressure, fission gas pressure, etc. inside of the fuel assembly. The stalk could interfere with an In-Vessel Handling Machine (IVHM) during refueling. Thus, it was necessary to sever/cut the instrument leads at the fuel assembly interface and remove the stalk. The fuel assembly could be removed by the IVHM or left in the core. The design lifetime of the Open Test Assemblies (OTAs) was 300 full power days (three cycles).

Figure 25. Illustration of Fuels Open Test Assembly (FOTA).

3.3 Post-Irradiation Open Test Assembly (PIOTA)

A Post-Irradiation Open Test Assembly (PIOTA) illustrated in Figure 26 is a FOTA that remains in the core after the instrument leads have been severed. The stalk is installed above the lower fuel bundle portion and still provides sodium flowrate and temperature measurements.

Figure 26. Illustration of Post-Irradiation Open Test Assembly (PIOTA).

3.4 Vibration Open Test Assembly (VOTA)

The Vibration Open Test Assembly (VOTA) installed in 1979 following FFTF startup was instrumented to measure assembly flow-induced vibrations, sodium flowrate, core restraint loads, neutron flux, gamma flux, and gamma heating.

3.5 Absorber Open Test Assembly

This was another variation on the OTA theme.

3.6 Materials Open Test Assembly (MOTA)

The Materials Open Test Assembly (MOTA) illustrated in Figure 27 contained only non-fissionable materials for irradiation such that there would not be a decay heat source following irradiation as with irradiated fuel. Due to the lack of decay heat, a MOTA could be withdrawn at the end of a cycle to permit fuel handling operations and then reinserted at the beginning of the next cycle. The MOTA utilized a standard hex duct with the sodium flowrate controlled with an inlet orifice. Each MOTA had nine levels of specimens; one below the core, five in the active core, and three above the core (Figure 28 and Figure 29). There were seven uninstrumented canisters at each level in the MOTA-1 series and up to six instrumented canisters at each level in the MOTA-2 series. Each canister was 12.7 to 16.2 cm (5.0 to 6.4 in) high by 2.8 cm (1.1 in) in diameter. There were typically forty-eight specimen canisters (the below core canister was empty) in the MOTA-2 series of which thirty were closed and had independent online temperature control through control of the Argon-Helium gas composition and thermal conductivity inside a heat transfer gas gap between the heated specimen compartment and surrounding sodium. Other canisters were “weepers” admitting sodium for interaction with specimens. Each canister had baskets containing test specimens. Canisters were hinged to the central stalk for disassembly making it easier to remove the insulating end cap and specimens. In some MOTAs, the canisters were instrumented with as many as eighty-two leads for gas control, temperature, pressure, etc. Nine MOTAs (Figure 30) were irradiated from 1983 to 1992; eight for about one year and one for four months.

Figure 27. Illustration of Materials Open Test Assembly (MOTA).

Figure 28. MOTA Canisters

MOTA NEUTRON FLUX DISTRIBUTION

Figure 29. Neutron Flux Distribution Along Height of MOTA Canisters.

Figure 30. MOTA-1 Series and MOTA-2 Series Canister Configurations and Positions in FFTF Core.

3.7 Closed Loops and Closed Loop In-Reactor Assemblies

The original FFTF design could have simultaneously incorporated four closed loops. Each closed loop system incorporated a Closed Loop In-Reactor Assembly (CLIRA) designed for 306 effective full power days or three cycles. Each closed loop system is designed for a 2.3 MWt heat rejection to air. Two closed loop primary sodium modules were built and one was installed in a cell inerted with nitrogen inside of the FFTF containment. Although they represent the most capable of all of the closed loop designs, none of the closed sodium loops at FFTF were actually used. In addition, the FFTF closed loops, which were designed to remain in the reactor during refueling operations, forced the FFTF designers to perform refueling around the closed loop locations inside the core. Their solution was to incorporate three separate In-Vessel Handling Machines (IVHMs). Figure 31 and Figure 32 illustrate one of the closed loop systems and a CLIRA.

CLOSED LOOP SYSTEM

FOR TESTING OF FAST REACTOR FUELS AND MATERIALS IN FFTF

Figure 31. Illustration of FFTF Individual Closed Loop System.

Figure 32. Illustration of FFTF Closed Loop In-Reactor Assembly (CLIRA).

Figure 33 and Figure 34 present schematic illustrations of the closed loop primary and intermediate sodium systems, respectively. The closed loop primary sodium system is extensive. It incorporates a cold trap to purify the sodium as well as sodium sampling facilities. Two electromagnetic (EM) pumps were provided for redundancy to reduce the likelihood of a loss of closed loop primary sodium coolant flow. The closed loop intermediate sodium system is located outside of the containment and below grade with the exception of the intermediate sodium-to-air heat exchanger. The closed loop primary sodium system was designed to be compact to minimize the amount of space required inside of the inerted containment cell as shown in Figure 35 and Figure 12. The sodium-to-air dump heat exchanger was a large component as shown in Figure 37.

Figure 33. Schematic Illustration of FFTF Closed Loop Primary Sodium System.

Figure 34. Schematic Illustration of FFTF Closed Loop Intermediate Sodium System.

Figure 35. Illustration of FFTF Closed Loop Primary Sodium System.

Figure 36. FFTF Closed Loop Primary Sodium System Turned on Its Side.

Figure 37. Closed Loop System Sodium-to-Air Heat Exchanger (Dump Heat Exchanger).

Figure 38 shows a Closed Loop In-Reactor Assembly. The pressure tube is the outer boundary. Sodium flows downward in an annulus between the pressure tube and flow tube, and then upward through the test section inside a 985 cm (2.75 inch) diameter circle. The CLIRA is instrumented with 26 leads. It could accommodate nineteen or thirty-seven fuel pins for steady state and transient irradiation testing. The maximum test section sodium exit temperature was 760 °C (1400 °F). The maximum CLIRA sodium exit temperature was 649 °C (1200 °F). This temperature limit was observed by mixing sodium from the test section exit with a cooler sodium bypass flow.

The CLIRA pressure tube was designed to withstand the maximum pressure from a postulated vapor explosion involving the melted oxide fuel and sodium. The CLIRA also incorporated a meltdown cup with heat sinks and heat transfer to surrounding sodium to contain a postulated meltdown of the oxide fuel.

Figure 38. Closed Loop In-Reactor Assembly (CLIRA).

3.8 FFTF Test Assembly Locations

The FFTF incorporated eight special test assembly locations shown in Figure 39 providing four locations for open test assemblies plus four locations originally intended for closed loops. Two open test assembly plus four closed loop positions are located on lines that avoid interference with the three In-Vessel Handling Machines during refueling.

1-9: ROW NUMBER FOR CORE ASSEMBLY. ROWS 1-6 ARE THE FUELED ZONE; ROWS 7-9 ARE THE REFLECTOR ZONE.

CL - CLOSED LOOP POSITION.

SCR - SECONDARY CONTROL ROD ABSORBER POSITION. SIX ASSEMBLIES WERE INSTALLED INITIALLY.

OT - OPEN TEST ASSEMBLY POSITION.

PC - POSITIONS IN WHICH CONTROL RODS MAY BE INSTALLED.

PFS - PERIPHERAL FIXED SHIM ABSORBERS.

PCR - PRIMARY CONTROL RODS. THREE RODS WERE INSTALLED INITIALLY.

Figure 39. FFTF Test Assembly Locations.

3.9 References

9. C. P. Cabell, "A Summary Description of the Fast Flux Test Facility," Hanford Engineering Development Laboratory, HEDL-400, 1980.
10. D. J. Oakley, "Closed Loop In-Reactor Assembly (CLIRA) A Fast Flux Test Facility Vehicle," American Society of Mechanical Engineers Winter Annual Meeting, San Francisco, HEDL-SA-1605FP, 1978.
11. D. C. Corrigan, L. J. Julyk, C. W. Hoth, J. C. McGuire, and W. R. Sloan, "FFTF Utilization for Irradiation Testing," Second International Conference on Liquid Metal Technology in Energy Production, Richland, WA, April 20-24, 1980.
12. R. E. Bauer, "Instrumented In-Reactor Test Capabilities in FFTF," Journal of Nuclear Materials, Vol. 103 and 104, pp. 1563-1566, 1981.
13. A. M. Ermi, L. R. Greenwood, and H. L. Heinisch, "Irradiation Parameters for the FFTF Materials Open Test Assemblies from 1983 through 1992," WHC-SD-FF-TD-010, Revision 0, August 1994.
14. A. Farabee and R. P. Omberg, "FFTF Closed Loops," Presentation Slides, Fourth U.S.-Russia Civil Nuclear Energy Working Sub-Group.

4 JOYO Test Vehicles

4.1 Introduction

JOYO has a significantly smaller core than FFTF. The JOYO Mk-III core (Figure 40) has a height of 0.5 m and a diameter of 0.8 m relative to a 0.91 m height and 1.20 m diameter for FFTF. JOYO had a power level of 140 MWt relative to 400 MWt for FFTF. As a result, the number of testing locations available in JOYO is less than that in FFTF. JOYO will be derated to 100 MWt when JAEA gains authorization from their regulator for JOYO restart. This reduced core power rating will reduce the neutron flux below what was previously achievable. Like FFTF, JOYO rejects heat from the intermediate sodium coolant to air (Figure 41). Information on the JOYO test vehicles was obtained from References [15 through 23].

The different test vehicles available for irradiations are summarized in Figure 42. Temperatures in the test vehicles can be as high as 750 °C in the core.

Figure 40. JOYO Mk-III Core Layout.

Figure 41. Illustration of JOYO.

Figure 42. Different Test Vehicles Available for Irradiations in JOYO.

4.2 Un-instrumented Test Assemblies

Different types of un-instrumented test assemblies are available analogous to the different types of un-instrumented test assemblies developed and irradiated in EBR-II. There are four types of un-instrumented fuel irradiation rigs (Figure 43) for irradiation of fuel pins. Fuel irradiated under conditions that might result in failure and fission gas release can be irradiated inside of capsules that can contain the fission gas pressure as shown in Figure 44. There are materials irradiation rigs for core, structural, and absorber materials as shown in Figure 45.

Figure 43. Four Types of Un-instrumented Fuel Irradiation Rigs Providing Differences in Environment.

Features

- Capsule can withstand the pressure increase in the event of fuel failure
- Allow irradiation of fuels with insufficient irradiation data
- Flow rate can be set individually
- Allow interim examination and re-irradiation

License

Fuel Forms	Oxide, Carbide, Nitride, Metal
MA Contents	$\leq 50 \%$
Melting Area of Pellet	Oxide: $\leq 20 \%$ Others: No Melt
Burn-up	$\leq 200 \text{ GWd/t}$

Figure 44. Un-instrumented Capsule Fuel Irradiation Rig.

Figure 45. Un-instrumented Materials Irradiation Rigs.

4.3 Instrumented Test Assembly (INTA)

The Instrumented Test Assembly (INTA) illustrated in Figure 46 is analogous to the Fuels Open Test Assembly (FOTA) in FFTF. Similar to a FOTA, there is a lower fuel assembly part and an upper stalk part. Similar to a FOTA, it is necessary to sever the instrument leads between the fuel assembly and stalk with a cutter and raise the stalk such that the INTA does not interfere with refueling operations.

Figure 46. Illustration of Instrumented Test Assembly (INTA).

4.4 Upper Core Structure Irradiation Plug Rig (UPR)

The Upper Core Structure Irradiation Plug Rig (UPR) illustrated in Figure 47 was an in-reactor test for the Self Actuated Shutdown System (SASS) Curie point electromagnet.

13

Figure 47. Illustration Upper Core Structure Irradiation Plug Rig (UPR).

4.5 Materials Testing Rig with Temperature Control (MARICO)

The Materials Testing Rig with Temperature Control (MARICO) illustrated in Figure 48 is analogous to the Materials Open Test Assembly (MOTA) in FFTF. It contains five levels of capsules with three capsules per layer. Similar to the MOTA, the specimen temperature is controlled by controlling the proportions and pressure of Argon and Helium in a gas gap controlling heat rejection to primary sodium. Capsules are hinged to a central tube to facilitate the removal of specimens similar to a MOTA (Figure 49). However, unlike a MOTA that is integral with and can be removed together with the overlying stalk, the MARICO test assembly must be removed by the fuel handling equipment. As a consequence, it is necessary to sever the instrument lines between the test assembly and stalk with a cutter. It was the MARICO-2 test assembly that was involved in the incident first detected on June 11, 2007 illustrated in Figure 50 and Figure 51. Following cutting of the instrument leads, the test assembly was moved from its irradiation position to a location in the In-Vessel Storage Rack. However, the handling head did

not disconnect properly from the test assembly and the test assembly and handling head were subsequently deformed.

Material Testing Rig with Temperature Control (MARICO)

14

Figure 48. Schematic Illustration of the Materials Testing Rig with Temperature Control (MARICO).

Figure 49. Specimens for Installation Inside of Canisters Mounted with Hinges on MARICO Test Assembly.

- The test subassembly wasn't disconnected from the holding mechanism.
- Test subassembly was bent on the in-vessel storage rack.

Figure 50. Illustration of MARICO-2 Incident for Which a Problem Was Detected on June 11, 2007.

3D Image of Obstacle in the Reactor Vessel

Figure 51. Illustration of MARICO-2 Incident.

4.6 Ex-Vessel Irradiation Rig (EXIR)

The Ex-Vessel Irradiation Rig (EXIR) illustrated in Figure 52 enables irradiations at low neutron fluxes immediately outside of the reactor vessel inside of the gap between the reactor and guard vessels. The neutron flux capability is 10^{10} n/(cm²·s). Temperatures are controlled online with an electric heater. Loadings can be applied with gas pressure.

Figure 52. Illustration of Ex-Vessel Irradiation Rig (EXIR).

4.7 References

15. T. Soga, W. Itagaki, Y. Kihara, and Y. Maeda, "Endeavor to Improve In-Pile Testing Techniques in the Experimental Fast Reactor Joyo," Presentation Slides, IAEA TM In-Pile Testing and Instrumentation for Development of Generation-IV Fuels and Materials, Halden, Norway, August 21-24, 2012.
16. "Outline of Joyo Restoration Work," Experiment Fast Reactor Joyo, Oarai R&D Center, Japan Atomic Energy Agency.
17. "Irradiation Technology in Experimental Fast Reactor Joyo Irradiation Rig Assembling Facility (IRAF)," Pamphlet, Japan Atomic Energy Agency, O-arai Research and Development Center.
18. H. Kataoka, T. Yasu, H. Takatsudo, and S. Miyakawa, "Development of Material Irradiation Rig with Precision Temperature Control in Experimental Fast Reactor JOYO," Journal of Nuclear Materials, Vol. 258-263, pp. 677-681, 1998.
19. K. Mukai, "Development of Sodium-Cooled Fast Reactors in Japan," Presentation Slides, 1st International Topical Seminar on "Coolant and Innovative Reactor Technologies," Cadarache, France, November 27-28, 2006.
20. T. Soga, "Improvement of the Irradiation Capability in the Experimental Fast Reactor Joyo," Presentation Slides, IAEA Technical Meeting, November 2008.
21. T. Sekine, T. Ashida, K. Imaizumi, M. Takamatsu, A. Nagai, and Y. Maeda, "Restoration Work for Obstacle and Upper Core Structure in Reactor Vessel of Experimental Fast Reactor Joyo," Slides, Japan Atomic Energy Agency, December 7, 2009.

22. Y. Maeda, “Retrieval of Damaged Components from Experimental Fast Reactor Joyo Reactor Vessel,” Slides, Japan Atomic Energy Agency, June 8, 2010.
23. See https://www.jaea.go.jp/04/o-arai/joyo_users_guide/equipment/index.html

5 BOR-60 Closed Loops

5.1 Introduction

A comprehensive review of the irradiation testing vehicles used in BOR-60 shall not be given here. Instead the focus is upon closed loops installed and irradiated in BOR-60 or envisioned for irradiation.

5.2 Independent Lead Channel (ILC)

The Independent Lead Channel (ILC) shown in Figure 53 and Figure 54 was a self-contained lead closed loop for irradiation of BREST-OD-300 nitride fuel pins in flowing lead (Pb) coolant and a fast neutron spectrum [24 and 25]. The self-contained closed loop was designed, fabricated, and tested in BOR-60 irradiations. The closed lead flow was driven by an axial flow centrifugal pump driven by magnetic clutch by an external motor. Control of the concentration of oxygen dissolved in the lead was achieved by bubbling gas mixtures through the lead. Electrical heating was provided to melt the lead and prevent it from refreezing. Heat was rejected to in-reactor vessel sodium. Following two irradiation campaigns of an ILC in BOR-60, it was necessary to improve the design including the lead pump and dissolved oxygen control due to unanticipated problems. Those problems were not uncovered during out-of-pile testing prior to the first irradiation campaign.

5.3 Sodium Closed Loops

Self-contained sodium closed loops have also been considered for installation in BOR-60. Figure 55 presents an illustration of such a concept with its own electromagnetic (EM) sodium pump.

Figure 53. Independent Lead Channel (ILC) Irradiated in BOR-60 (Dimensions in mm).

Figure 54. Plan View of Independent Lead Channel (ILC) Irradiated in BOR-60 (Dimensions in mm).

RIAR. 1995.

Figure 55. Self-Contained Sodium Closed Loop Concept for BOR-60 with Electromagnetic Sodium Pump.

5.4 References

24. V. N. Leonov et. al., "Pre- and In-Pile Tests of Fuel Element Mock-ups for the BREST-OD-300 in the Independent Lead-Cooled Channel of the BOR-60 Reactor," ICONE11-36409, 11th International Conference on Nuclear Engineering, Tokyo, Japan, April 20-23, 2003.
25. V. N. Efimov et. al., "The BOR-60 Loop-Channel Design for Testing the BREST Reactor Fuel," Power Reactors and Sub-Critical Blanket Systems with Lead and Lead-Bismuth as Coolant and/or Target Material, IAEA-TECDOC-1348, May 2003.

6 PHÉNIX Closed Loop Concept

6.1 Introduction

A comprehensive review of the irradiation testing vehicles used in PHÉNIX shall not be given here. Instead the focus is upon a closed loop conceived for installation in PHÉNIX.

6.2 BAUPHIX Closed Sodium Loop

The BAUPHIX closed sodium loop design shown in Figure 56 and Figure 57 was a self-contained sodium closed loop for irradiation of up to nineteen PHÉNIX and SUPERPHÉNIX fuel pins in flowing sodium and a fast neutron spectrum under abnormal temperature and flow conditions coolant and a fast neutron spectrum [26 and 27]. It did not progress beyond the concept stage and closed loops were never installed in PHÉNIX. The sodium flow was driven by an electromagnetic (EM) sodium pump. A hot sodium trap was provided for sodium purification. An oxygen sensor measured the dissolved oxygen level in the sodium to determine the sodium purity. Sodium temperatures in the closed loop circuit could vary from 430 to 600 °C. Heat was rejected to in-vessel primary sodium. The heat rejection capability was 0.6 MWt.

Figure 56. BAUPHIX Closed Sodium Loop Concept for Installation in PHÉNIX (Dimensions in mm).

Figure 57. BAUPHIX Closed Sodium Loop Concept for Installation in PHÉNIX (Dimensions in mm).

6.3 References

26. P. Coulon and L. Martin, "PHÉNIX in the Future – New Prospects," pp. 119-124, Proceedings LMR: A Decade of LMR Progress and Promise, Washington, DC, November 11-15, 1990.
27. P. Coulon, "Loop BAUPHIX," pp. 148-182, Specialists' Meeting on Plant Operating Experience of JOYO/KNK-II/PHENIX/PFR, O-arai Engineering Center, O-arai, Japan, Power Reactor and Nuclear Fuel Development Corporation, PNC TN9430 90-004, October 16-20, 1989.

7 Jules Horowitz Reactor (JHR) Closed Loops

7.1 Introduction

The Jules Horowitz Reactor (JHR) under construction by AREVA at the CEA Cadarache site has been designed to irradiate fuels and materials in fast and thermal neutron fluxes as well as to produce ^{99}Mo for 25 to 50 % of current European needs. Startup is currently planned for 2021. There is an extensive literature on JHR. References that the author has found to be more useful are listed below [28 through 41]. Unfortunately, the open literature contains little detailed information.

The Jules Horowitz Reactor (Figure 58 through Figure 64) is a 100 MWt loop-type pressurized water reactor operating at low pressure (1.2 MPa). The fuel is uranium-silicide, U_3Si_2 , in an aluminum matrix. The uranium is enriched in ^{235}U to 27 %. While the U.S. research and test reactor program has been advocating the use of aluminum alloy-clad uranium-molybdenum fuel, it was felt that this fuel type was not yet ready when the fuel was selected for JHR. There are thirty-six fuel assemblies. The core is under-moderated to produce a fast neutron spectrum. The peak fast flux level inside of the 0.6 m diameter by 0.6 m high core is stated to be 1×10^{15} n/(cm²·s) for energies greater than 0.1 Mev and 5×10^{14} n/(cm²·s) for energies greater than 1 Mev (Figure 65). Outside of the reactor vessel/tank, there is a 0.3 m thick beryllium reflector that thermalizes neutrons resulting in a peak thermal neutron flux of 5×10^{14} n/(cm²·s). Because of the loop-type design, the core essentially occupies the inner diameter of the reactor vessel. The reflector resides inside of a large water pool. The core incorporates three large diameter and seven small diameter fixed test locations while the reflector incorporates about twenty fixed locations and six displacement slots. The reactor is surrounded by a large PWR-type containment likely similar to that of the European Pressurized Reactor (EPR) and the reactor building is seismically isolated; there is a seismic fault in the vicinity of the Cadarache site. The containment design basis is a postulated severe accident involving melting of the largely aluminum fuel, intermixing with water, and a steam explosion. The primary water coolant transfers heat to a secondary water coolant circuit through three heat exchangers. Heat is rejected from the secondary water circuit.

The Jules Horowitz Reactor incorporates test articles with static coolant (water or NaK). They are not discussed here. The focus is upon the existing closed loop concepts and designs. The JHR closed loop approach does not involve a single multipurpose closed loop design. Instead there are several closed loop designs with each closed loop designed for more specific purposes.

Figure 58. Jules Horowitz Reactor Building and Containment Under Construction in September 2015.

Figure 59. Illustration of Jules Horowitz Seismically Isolated Reactor Building and Containment Layout and Reactor Vessel with Surrounding Reflector.

Figure 60. Jules Horowitz Reactor Core, Reactor Vessel/Tank, Beryllium Reflector Outside of the Reactor Vessel, and Slot Displacement Systems for Closed Loops.

Figure 61. Illustration of LORELEI Closed Loop Test Article on Displacement System Inside Slot in Reflector Inside Water Pool and Supporting Systems Inside of Containment.

Figure 62. Illustration of Displacement Device as Well as Reactor Building and Containment Layout.

Figure 63. Further Illustration of Reactor Core, Reactor Vessel, and Surrounding Reflector in Water Pool.

Figure 64. Plan View of Reactor Core and Reflector.

Figure 65. Calculated Neutron Spectra Inside Core and Reflector.

7.2 CALIPSO Closed NaK Loop

The core can incorporate self-contained closed loops such as the CALIPSO test article for irradiation of material samples under similar temperature conditions in a fast neutron flux at high temperatures (Figure 66 through Figure 68). CALIPSO incorporates flowing NaK (sodium-potassium eutectic) as the closed loop primary coolant to minimize temperature variations between samples to maximum of 8 °C. The NaK melting temperature is -12.6 °C which is below the reactor pressurized water coolant core inlet temperature of 30 °C. The closed loop incorporates a self-contained electrical heater, electromagnetic pump, and recuperator heat exchanger. Heat is rejected to upward flowing reactor water coolant surrounding the test article.

Figure 66. Illustration of CALIPSO Test Article.

→ Investigation of physical properties of material under high dpa

- Thermodynamic loop integrated within the test device
 - Heat Exchanger (HE) / Electrical Heater (EH)
 - Innovative electromagnetic pump (L 450 mm, D 80 mm) → NaK flow (2 m³/h)
- Improvement of the sample temperature mastering
 - From 250 up to 450°C (setting of HE & EH parameters)
 - $\Delta\theta < 8^{\circ}\text{C}$ (Tmax – Tmin all along the samples stack)

- On-going qualification of the design with a CALIPSO prototype
 - First successful tests of the electromagnetic pump

Figure 67. Information on CALIPSO Test Article.

Figure 68. CALIPSO Electromagnetic NaK Pump (Left) and Prototype for Testing (Right).

7.3 Closed Water Loops Inside of the Reflector

The beryllium reflector can incorporate self-contained and external closed water loops in the thermal neutron fluxes inside of the six slot locations. The JHR configuration with a reflector inside of a water pool outside of the reactor vessel is a remarkably simple solution to the logistical problems of irradiating test articles that are part of a closed loop. The beryllium reflector is outside of the reactor vessel and the reactor core. The test article is mounted on a displacement vehicle that can be moved under water inside of the slot closer to or farther from the core to select the desired thermal neutron flux. The test article can be moved during an irradiation campaign to maintain an unvarying flux, for example. There is no need for the test article to be installed in and removed from the reactor vessel or core through an upper head while maintaining leak tightness. Blocks/stops are installed in the displacement slots to mechanically limit the motion toward the core thereby limiting the possible power. A limitation of this approach is that only a thermal flux is available outside of the reactor vessel. The MADISON, ADELIN, LORELEI, and CLEO closed loops (Figure 70 through Figure 77) are being developed for use with the displacement slots. The test articles are coupled by flexible piping inside of the water pool to supporting closed loop components and analysis equipment in compartments/cubicles inside of the containment.

The reflector and water pool outside of the reactor vessel are also part of the reactor. The primary coolant boundary of each closed loop should be maintained leak tight. To prevent the release of radioactivity in the event of a postulated closed loop primary coolant boundary leak, the closed loop components and analysis equipment are located inside of the containment.

7.4 MADISON Closed Water Loop

The MADISON (Multi-rod Adaptable Device for Irradiations of experimental fuel Samples Operating in Normal conditions) closed water loop illustrated in Figure 69 through Figure 71 is for irradiation of LWR fuel pins under normal operating conditions. Irradiation may be long term. The closed loop coolant is pressurized water closed loop coolant up to 16 MPa. Up to 8 fuel pins can be accommodated. The closed water loop incorporates pumps, a preheater before the test section, a cooler, water chemistry monitoring, water sampling, water purification, and water treatment. The test article is moved inside the displacement slot as needed to select the neutron flux. Flexible piping inside the water pool connects the test article to water loop components inside an experiment cubicle compartment inside of the containment.

7.5 ADELIN Closed Water Loop

The ADELIN (Advanced Device for Experimenting up to Limits Nuclear fuel Elements) illustrated in Figure 72 through Figure 74 is for irradiation of single LWR fuel pins under abnormal but less than accident conditions that can result in cladding failure.

7.6 LORELEI Closed Water Loop

The LORELEI (Light water, One Rod Equipment for LOCA Experimental Investigations) illustrated in Figure 75 is for irradiation of single pre- and re-irradiated LWR fuel pins under LOCA conditions through quenching including measurement of fission product releases. Re-irradiation is carried out to build up short-lived fission products.

7.7 CLOE Closed Water Loop

The CLOE (Corrosion LOop Experiment) illustrated in Figure 76 and Figure 77 is for irradiation of LWR material samples in flowing coolant to investigate irradiation-assisted stress corrosion cracking. Comparison of irradiated samples will be made with samples in an autoclave without irradiation inside of a cubicle in the containment.

Figure 69. Illustration of MADISON Closed Loop.

Figure 70. Schematic Illustration of MADISON Closed Loop.

Description A loop device for irradiation of LWR fuel samples in normal conditions of power reactors <ul style="list-style-type: none"> - In JHR reflector - On displacement system - Heavy components in cubicle 	Type of fuel sample All type of LWR fuel sample <ul style="list-style-type: none"> - PWR / BWR and VVER fuel samples - UO₂ fuels (up to 7% in U⁵) - MOX fuels (up to 15% in Pu/(U+Pu)) - Fresh fuels - High burn up fuels (120GW.j/t) 	Carrying capacity Flexible loop with a large carrying capacity <ul style="list-style-type: none"> - 4 fuel rods of LHGR $\leq$ 400W/cm and clad diameter $\leq$ 10 mm - 3 fuel rods of LHGR $\leq$ 400W/cm and clad diameter $\geq$ 10 mm - 7 fuel rods of LHGR $\leq$ 200W/cm
Type of experiment Characterization and qualification of fuel samples <ul style="list-style-type: none"> - Fuel behaviour vs BU and LHGR - Long-term irradiations - (creep, corrosion, crack propagation...) - Comparison investigations - Re-irradiation before ramps 	Measurements <ul style="list-style-type: none"> - Temperature, pressure, water flow - Precise thermal balance (5%) - Clad elongation (LVDT) - Other measurements possible (Fission Products, Non destructive tests...) 	Fluid environment Representative of power reactors <ul style="list-style-type: none"> - Thermal-hydraulics of PWR, BWR or VVER - Chemistry of PWR, BWR or VVER 	Performances Good homogeneity between any 2 identical fuel rods <ul style="list-style-type: none"> - 3-5% max. heterogeneity (in case of 4 samples) of linear power - For all type of fuels / Burn up - Use of thin neutron screens

Figure 71. Information on MADISON Closed Loop.

Description

A one rod loop device for up to limit irradiations of LWR fuel samples

- In reflector
- On displacement system
- Heavy components in cubicle
- 3 or 4 ramp tests per reactor cycle using an underwater transfer station (which allows loading and retrieving the sample holder without disconnecting the ADELIN device)

Type of experiment

Characterization and qualification of fuel samples

- Power ramp tests
- Post failure behaviour
- Rod over-pressure threshold
- Melting limits
- Lift off
- Water contamination in case of failure

Chemistry

Representative of LWR power reactors

Type of fuel sample

All type of LWR fuel samples

- REP (et VVER) & REB ($5,5 \text{ mm} < \varnothing < \sim 14 \text{ mm}$) fuel samples
- Fuel rod length : up to 600 mm
- UO₂ fuels (up to 12% in U₅)
- MOX fuels (up to 20% in Pu/(U+Pu))
- Fresh or High burn up fuels (BU max : 120 GW j/t)

Thermal hydraulics

Representative of LWR power reactors

- PWR (155 bars)
- BWR (slight overpressure: 90 bars, T_{sat} (75 bars), low void fraction)
- Inlet temperature > 250°C
- $\Delta T > 20^\circ\text{C}$ (Thermal balance)
- Clad surface T : 345 +/- 10°C

Neutron flux

A device able to reach high power level and high power ramp

- Up to 620 W/cm for low enriched fuel
- Power ramps : up to 700W/(cm.min)

Experimental tests

Several interfaces with JHR system

- Direct piping to Fission Product laboratory
- Tight interface with alpha fuel cell for post-irradiation examination
- Gamma spectrometry, X radiography and neutron radiography possible

On-line measurements

Numerous and precise sensors

- Temperature, pressure
- Thermal balance
- Fission gas release
- Clad failure
- ...

Purification system

Purification on a reduced water flow

- Jet pump for flow amplification (x5)
- Irradiation on failed fuels

Figure 72. Information on ADELIN Closed Loop.

Figure 73. Illustration of ADELIN Closed Loop.

Figure 74. ADELIN Test Article.

Figure 75. Illustration of LORELEI Closed Loop.

Figure 76. Illustration of CLOE Closed Loop.

Figure 77. Illustration of CLOE Sample Holder.

7.8 Comparison of JHR Closed Loops

Designs and features of the closed loops under development and design for JHR are compared in Table 3.

Table 3. Comparison of JHR Closed Loop Designs and Major Features

Closed Loop	MADISON (Multi-rod Adaptable Device for Irradiations of experimental fuel Samples Operating in Normal conditions)	ADELIN (Advanced Device for Experimenting up to Limits Nuclear fuel Elements)	LORELEI (Light water, One Rod Equipment for LOCA Experimental Investigations)	CLOE (Corrosion LOOP Experiment)	CALIPSO (in-Core Advanced Loop for Irradiation in Potassium Sodium)
Location	Displacement slot in reflector in water pool outside of reactor tank in thermal neutron spectrum	Displacement slot in reflector in water pool outside of reactor tank in thermal neutron spectrum	Displacement slot in reflector in water pool outside of reactor tank in thermal neutron spectrum	Displacement slot in reflector in water pool outside of reactor tank in thermal neutron spectrum	Core inside of reactor tank in fast neutron spectrum
Status	Under design by CEA. No details in open literature. Planned to be ready at JHR startup	Under design by CEA. No details in open literature. Planned to be ready at JHR startup	Under development and design by CEA and IAEA (Israel). No details in open literature. Planned to be ready a few years after JHR startup	Under development and design by CEA and DAE (India). No details in open literature.	Under design. No details in open literature.
Closed Loop Coolant	Pressurized water	Pressurized water	Pressurized water	Pressurized water	NaK (Sodium-potassium eutectic)
Number of loops in reactor	1 as needed	1 as needed	1 as needed	1 as needed	1 as needed
Purpose	Irradiation of LWR fuel pins under normal operating conditions. Irradiation may be long term	Irradiation of single LWR fuel pin under abnormal but less than accident conditions that can result in cladding failure	Irradiation of single pre- and re-irradiated LWR fuel pin under Loss-of-Coolant Accident conditions through quenching including measurement of fission product releases	Irradiation of LWR material samples in flowing coolant to investigate irradiation-assisted stress corrosion cracking. Comparison with samples in autoclave without irradiation inside cubicle	Irradiation of material samples under similar temperature conditions in fast neutron flux and high temperatures. Flowing NaK is utilized to minimize the temperature variation between samples to a maximum of 8 °C
Closed Loop Type and Features	Closed water loop with pumps, preheater before test section, cooler, water chemistry monitoring, water sampling, water purification, and water treatment. Test article moved inside displacement slot as needed to select neutron flux. Flexible piping inside water pool connects test article to water loop components inside experiment cubicle compartment inside of containment.	Closed water loop with pumps connected through heat exchanger to secondary water loop with heat rejection. Flexible piping inside water pool connects test article to water loop components inside experiment cubicle compartment inside of containment.	Closed natural circulation water loop. Heat exchange through double-wall pressure boundary to pool water flowing downward through annular channel driven by water pump on displacement device. Helium cover gas to control pressure and water level. Steam formed by vaporization of water at bottom of pin. Electrical heater to control fuel pin heatup rate. Equipment to capture released fission products for analysis inside experiment cubicle.	Closed water loop. Pump, heater, pressurizer, water cleanup, and water chemistry equipment located in cubicle connected to test article by flexible piping inside water pool.	Test article with self-contained electrical heater, electromagnetic pump, and recuperator heat exchanger. Heat rejection to upward flowing reactor water coolant surrounding test article with 30 °C inlet temperature.
Number of Fuel Pins in Test Section	Up to 8	1	1	Material samples	Material samples
Coolant Pressure, MPa	16.0	15.5	7.0	19.0	Low
Coolant Temperature, °C	296/329 inlet/outlet typical of EPR	270/330 pre-transient; Higher temperatures during transient	270/330 pre-transient; Up to 1300 during LOCA heatup; 10 °C/s heatup and quenching rates	320 to 390	Maybe 250 inlet in first phase. Up to 450 in first phase and up to 600 in second phase.
Coolant	1.63×10^{-3}	2.56×10^{-4}	TBD	Up to 1.02×10^{-2}	5.56×10^{-4}

Flowrate, m ³ /s				at 390 °C	
Coolant Flowrate, kg/s	1.2	0.2	TBD	Greater than 1.0	0.5
Heat Removal Rate, MWt	0.23	0.067 pre-transient	TBD	TBD	TBD
Space Available for Test Section	TBD	Circle 2.6 cm diameter	TBD	TBD	Circle 2.4 cm diameter
Area for Test Section, cm ²	TBD	5.3	TBD	TBD	4.5

7.9 References

28. See <http://www.cad.cea.fr/rjh/irradiation-devices.html> and <http://www.cad.cea.fr/rjh/index.html>
29. J. Estade et. al., “Jules Horowitz Reactor: Preparation of the Commissioning Phase and Normal Operation,” European Research Reactor Conference, RRFM-IGORR 2016, Berlin, Germany, March 13-17, 2016.
30. S. Gaillot et. al., “Jules Horowitz Reactor, France, Development of an Experimental Loop Integrating an Optimized Irradiation Process,” European Research Reactor Conference, RRFM-IGORR 2016, Berlin, Germany, March 13-17, 2016.
31. D. Moulin et. al., “Out-of-Pile Testing of the CALIPSO Irradiation Device for the Jules Horowitz Reactor,” European Research Reactor Conference, RRFM-IGORR 2016, Berlin, Germany, March 13-17, 2016.
32. G. Bignan et. al., “The Jules Horowitz Reactor Research Project: A New High Performance Material Testing Reactor Working as an International User Facility – First Developments to Address R&D on Material,” 2nd International Workshop Irradiation of Nuclear Materials: Flux and Dose Effects, CEA – INSTN Cadarache, France, November 4-6, 2015.
33. C. Colin et. al., “Test Devices in Jules Horowitz Reactor Dedicated to the Material Studies in Support to the Current and Future Nuclear Power Plants,” Fontevraud 8 – Contribution of Materials Investigation and Operating Experience to LWRs’ Safety, Performance and Reliability, Avignon, France, September 15-18, 2014.
34. L. Ferry et. al., “The LORELEI Test Device for LOCA Experiments in the Jules Horowitz Reactor,” Proceedings of WRFPM 2014, Sendai, Japan, September 14-17, 2014.
35. G. Bignan and J. Estrade, “The Jules Horowitz Reactor: A New High Performances European MTR (Material Testing Reactor) with Modern Experimental Capacities: Toward an International User Facility,” Available at http://www.igorr.com/home/liblocal/docs/IGORR2013/05_1003.pdf

36. C. Colin et. al., “Experimental Devices in Jules Horowitz Reactor Dedicated to the Material Studies in Support to the Current and Future Nuclear Power Plants,” 14th Meeting of the International Group on Research Reactors, IGORR 14, Prague, Czech Republic, March, 2012.
37. D. Parrat et. al., “The Future Jules Horowitz Material Test Reactor: A Major European Research Infrastructure for Sustaining the International Irradiation Capacity,” 9th International Conference on WWER Fuel Performance, Modeling and Experimental Support, Helena Resort near Burgas, Bulgaria, September 17-24, 2011.
38. P. Roux et. al., “The MADISON Experimental Hosting System in the Future Jules Horowitz Reactor,” Paper and Presentation, 13th Meeting of the International Group on Research Reactors, IGORR 13, Knoxville, Tennessee, USA, September 19-23, 2010.
39. D. Moulin et. al., “Thermal Assessment of the CALIPSO Irradiation Device for the Jules Horowitz Reactor,” Paper and Presentation, 13th Meeting of the International Group on Research Reactors, IGORR 13, Knoxville, Tennessee, USA, September 19-23, 2010.
40. G. Bignan et. al., “The Jules Horowitz Reactor: A New High Performances European MTR Open to International Community,” Presentation, 13th Meeting of the International Group on Research Reactors, IGORR 13, Knoxville, Tennessee, USA, September 19-23, 2010.
41. G. Bignan et. al., “Fuel and Material Irradiation Hosting Systems in the Jules Horowitz Reactor,” 13th Meeting of the International Group on Research Reactors, IGORR 13, Knoxville, Tennessee, USA, September 19-23, 2010.

8 Comparison of In-Reactor Closed Loops

8.1 Discussion

In-reactor closed loops have been sought for irradiation and testing throughout the history of fast reactors.

Table 4 compares the features of several notable closed loop concepts and designs for installation inside of SFRs or related to testing of fast reactor fuel and cladding. Closed loops incorporating sodium were an integral part of the FFTF design that could have simultaneously incorporated four such closed loops. Two closed loop primary sodium modules were built and one was installed in a cell inerted with nitrogen inside of the FFTF containment. Although they represent the most capable of all of the closed loop designs, none of the closed sodium loops at FFTF were actually used. In addition, these FFTF closed loops, which were designed to stay in the reactor during refueling operations, forced the FFTF designers to perform refueling around these installed closed loops. Pressurized water closed loops are also an integral part of the Advanced Test Reactor (ATR) thermal reactor design and are utilized for irradiation and testing.

In general, there have been two approaches to closed loops. The first is a test article with a self-contained closed coolant loop with a pump inside of the test article. Heat is rejected from the test article coolant to the surrounding reactor coolant or structures. In

Table 4, this approach is followed for the Independent Lead Channel (ILC) closed lead loop that was installed and operated in BOR-60 [24 and 25] shown in Figure 1 and Figure 54, the BAUPHIX closed sodium loop concept for PHÉNIX [26 and 27] shown in Figure 56 and Figure 57 but which was never design and installed, the Mk-III closed sodium loop [42] shown in Figure 16 used for numerous experiments with SFR fuel pins in the TREAT thermal reactor, and each of the up to three instrumented closed loop channels (Figure 68 and Figure 80) envisioned for MBIR [43 through 45] that is currently under construction in Dimitrovgrad. Typically, each self-contained closed loop test article is designed to fit inside of a core location similar to that of a single fuel assembly. This approach can, in principle, facilitate a design in which the test article can be installed and removed, each refueling cycle, using the refueling equipment. The TREAT Mk-III loop fit into two side-by-side core locations.

The second approach is an in-reactor assembly connected to an external primary coolant loop. The primary coolant loop typically rejects heat to an independent external intermediate coolant loop. In the case of FFTF, the intermediate sodium loop rejected heat to the air atmospheric heat sink. In FFTF, each in-reactor assembly occupied a core location similar to that of a single fuel assembly. Up to three external closed loops (Figure 79 and Figure 80) are envisioned for MBIR [43 through 45] for which each in-reactor assembly occupies seven hexagonal core positions. The hexagonal assembly flat-to-flat distance for MBIR is significantly less than that of FFTF.

As observed from

Table 4, the self-contained closed loop approach is generally limited in the flowrate that can be provided by the self-contained internal pump. The heat rejection capability from the closed loop coolant to the reactor coolant or surroundings is also limited. Typically, the space available for a test section is limited as well. The approach in which the coolant flow is driven by an external primary coolant loop can be designed to provide a significantly greater capability in terms of coolant flowrate, heat rejection capability, and space available for a test section.

The self-contained closed loop approach can be more challenging to the designer because of the need to fit a viable pump, perhaps a coolant chemistry control system, and other capabilities inside of a limited volume. For example, following two irradiation campaigns of an Independent Lead Channels (Figure 53 and Figure 54) in BOR-60, it was necessary to improve the design including the lead pump and dissolved oxygen control due to unanticipated problems. Those problems were not uncovered during out-of-pile testing prior to the first irradiation campaign. The external closed loop approach permits the use of existing proven technology for the closed primary and intermediate coolant loop systems as well as the heat rejection system that is less risky.

The external closed loop approach is more costly, however. The components are significantly more extensive increasing the cost. Additional equipment to install and remove the closed loop in-reactor assemblies is required. In FFTF, the primary sodium closed loops were designed to be installed inside of inerted cells inside of the containment. This has an effect of increasing the required containment volume which can further increase the test facility cost. However, a similar approach is being followed in the MBIR design because of the external closed loop testing capability.

The Independent Lead Channels that were installed and irradiated in BOR-60 are significant in that they represent the first time that a closed loop with another coolant than sodium (i.e., lead) was installed inside of a SFR and fuel and cladding of four fuel pins were irradiated in a flowing alternative coolant in a fast neutron spectrum. The MBIR design is being developed for the installation of both up to three self-contained and up to three external closed loops with alternative coolants including sodium, lead, lead-bismuth eutectic, gas, and liquid salt. It is unknown if and when the external closed loops will be designed, developed, and installed in MBIR. The external closed loop technology will be a new technology for the Russian designers and will need to be designed into the MBIR plant before construction to ensure that its requirements can be met.

Table 4. Comparison of Closed Loop Designs and Major Features

Reactor Plant	BOR-60	PHÉNIX	TREAT	MBIR	MBIR	FFTF
Closed Loop Design	Independent Lead-Cooled Channel (ILC) for BOR-60 (2002)	BAUPHIX Closed Sodium Loop for PHÉNIX(1998)	TREAT Mk-III Closed Sodium Loop	Instrumented Closed Loop Channel for MBIR	External Loop Channel for MBIR	Closed Loop for FFTF
Status	Closed loop was designed and tested in BOR-60 irradiations	Closed loop was just a concept	Deployed in TREAT	Under development	Status unknown	Designed into FFTF. Two primary system modules were fabricated, one was installed in cell. In-Reactor Closed Loop Assembly was never installed in FFTF.
Closed Loop Coolant	Lead (Pb)	Sodium	Sodium	Alternative coolants; Example of lead-bismuth eutectic (45 wt % Pb – 55 wt % Bi)	Alternative coolants; Example of sodium	Sodium
Number of loops in reactor	1	1	1	Up to 3	Up to 3	Up to 4
Purpose	Irradiation of BREST-OD-300 fuel pins in flowing lead and fast neutron spectrum	Irradiation of PHÉNIX and SUPERHÉNIX fuel pins in flowing sodium and fast neutron spectrum under abnormal temperature and flow conditions	Testing SFR fuel pins in flowing sodium under simulated accident conditions of power, temperature, or flow	Irradiation of LFR fuel pins in flowing lead-bismuth eutectic and fast neutron spectrum	Irradiation of SFR fuel pins in flowing sodium and fast neutron spectrum	Irradiation of SFR fuel pins in flowing sodium and fast neutron spectrum under normal or abnormal conditions of, temperature and flow
Closed Loop Type and Features	Test article with self-contained lead loop with lead flow driven by axial flow centrifugal pump driven by magnetic clutch by external motor. Lead dissolved oxygen control by bubbling gas mixtures through lead. Electrical heating to melt lead and prevent refreezing. Heat rejection to in-reactor vessel sodium.	Test article with self-contained sodium loop driven by electromagnetic pump. Hot sodium trap for sodium purification. Heat rejection to in-reactor vessel sodium.	Test article with self-contained sodium loop driven by annular linear induction pump in parallel leg to test section.	Test article with self-contained lead-bismuth eutectic loop with flow driven by axial flow centrifugal pump driven by magnetic clutch by external motor. Lead-bismuth dissolved oxygen control by bubbling gas mixtures through lead-bismuth. Electrical heating to melt lead-bismuth and prevent refreezing. Heat rejection to in-reactor vessel sodium.	Test article connected to external sodium loop with external heat rejection.	Closed Loop In-Reactor Assembly (CLIRA) connected to external primary sodium loop connected through intermediate heat exchanger to external intermediate sodium loop rejecting heat to air atmosphere through Dump Heat Exchanger (DHX). Primary and intermediate sodium coolant purification through cold trapping.
Number of Fuel Pins in	4	Up to 19	Up to 7	TBD	TBD	Up to 37

Test Section						
Coolant Inlet/Outlet Temperature, °C	490/575	430/600	354/538	320/482	430/600	538/649
Coolant Flowrate, m ³ /s	3.61×10 ⁻⁴	5.00×10 ⁻³	1.24×10 ⁻³	5.82×10 ⁻⁴	3.41×10 ⁻³	1.92×10 ⁻² maximum
Coolant Flowrate, kg/s	3.78	4.25	1.08	6.00	2.9	16.5 maximum
Heat Removal Rate, MWt	0.053	0.600	0.252	0.140	0.622	2.30
Space Available for Test Section	Circle 3.38 cm diameter	Circle 4.13 cm diameter	Circle 3.02 cm diameter	Circle 4.1 cm diameter	Circle 6.0 cm diameter	Circle 6.985 cm diameter
Area for Test Section, cm ²	8.97	13.4	7.15	13.2	28.3	38.3

MARK III LOOP THERMOCOUPLE LOCATIONS

Figure 78. Illustration of Mk-III Closed Sodium Loop Utilized in TREAT.

Figure 79. Illustration of Instrumented Closed Loop Channel (Left) and External Closed Loop Channel (Right) Envisioned for Use in MBIR.

Outside Loop Channels

Fuel assembly (93)	Material test assembly (14)	Instrumented Closed Loop Channel (3)
Control rod (8)	Blanket assembly (278)	In-reactor fuel storage (38)
		Shield assembly (74)

Figure 80. Core Layout Envisioned for MBIR Incorporating Three Locations for Instrumented Closed Loop Channels and Three Locations for External Closed Loop Channels.

8.2 References

42. C. Baker, "TREAT Sodium Loop: Status Update," Presentation, October 7-8, 2015.

43. A. Tuzov, “MBIR International Research Center: Current Progress and Prospects,” Presentation, IAEA Technical Working Group on Fast Reactors Meeting, Obninsk, Russia, May 2014.
44. A. N. Popov, “Design Solutions for the Full-Functional Complex of MBIR Facility,” Presentation, Saint Petersburg, Russia, 2014.
45. Yu Dragunov et. al., “Experimental Potentialities of the MBIR Reactor,” IAEA-CN-199-246, International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios FR13, Paris, France, March 4-7, 2013.

9 Impact of Closed Loops on Reactor Plant Design

As mentioned in the previous sections, vehicles to perform irradiation testing come in a number of different functionalities:

- Non-instrumented test assemblies
- Instrumented test assemblies (removed after 1 cycle)
- Flowing loop(removed after 1 cycle)
- Flowing loop (maintained in core for full duration – 3-4 cycles)

The following describes how the different irradiation test vehicles impact the design of the primary heat transport system and other parts of the fast reactor plant.

Non-instrumented test assemblies – Non-instrument test assemblies are roughly the same exterior dimensions as a typical core assembly and are thus handled by the fuel handling system. There is no impact to the design of the primary heat transport system as a result of this irradiation vehicle type.

Instrumented test assemblies (removed after 1 cycle) – need to have the ability for insertion of the test assemblies into the reactor through the reactor head. Instrument leads are cut and normal fuel handling equipment handles the core assembly. The upper internal structure and reactor head need to be designed with appropriate features to accommodate the equipment penetrations and special tools need to be developed to cut the instrumented assembly test leads. The handling of the irradiated test assembly follows the standard path of refueling.

Flowing Closed Loop Assembly (removed after 1 cycle) – This type of test assembly would need to have the ability for insertion of the test assemblies into the reactor through the reactor head. The upper internal structure and reactor head need to be designed with appropriate features to accommodate the equipment penetrations. The closed loop is removed from primary vessel or retracted from the core a large transfer system located ex-vessel.

Flowing Closed Loop Assembly (maintained in core for full irradiation cycle – 3-4 cycles) – This capability was designed into the FFTF reactor plant as flowing closed loop technology for performing irradiations in independent flowing coolant conditions. This closed loop technology,

as described above, was never deployed because of cost and schedule implications per discussions with Al Farabee.

A good comparison to understand the impacts of incorporating flowing closed loops and instrumented irradiation vehicles that stay in the reactor more than a single cycle is to compare the U.S. Fast Flux Test Reactor and the Clinch River Breeder Reactor designs – especially the designs of the primary heat transport system.

The Fast Flux Test Facility is a 400MW thermal sodium-cooled fast reactor built at the Hanford reservation in the state of Washington, Figure 81 and Figure 82. FFTF has the characteristics shown in Table 5:

Figure 81 – Picture of Fast Flux Test Facility – white dome is FFTF containment dome

Table 5 – Fast Flux Test Facility Design Parameters

Reactor Core Power Level	400MWt
Fuel type	U-Pu-Oxide fuel – MOX
Primary Plant Geometry	Loop plant (with 3 loops)

Sodium Flow Rate	43,500 gallons/minute
Primary sodium temperatures (inlet/outlet)	680F / 980F
Core Assemblies	199
Fueled Core Assemblies	74
Number of Closed Loops accommodated?	4 at 2.3MWth capacity each
In-vessel refueling system	Fixed arm in-vessel refueling (3 independent machines)
In-vessel storage	Yes – three locations

The FFTF first went critical in February 1980 and then achieved full power operations in December 1980. The reactor was shut down in December 1993. The FFTF reactor provided a verification of the CRBRP fuel design and some of its technologies such as scaled down versions of the pumps and intermediate heat exchangers.

The FFTF provided irradiation testing to investigate irradiation behavior of advanced, low-swelling fuel cladding materials (D-9 and HT-9) and the verification of large-scale component designs (pumps, heat exchangers). The FFTF mission was also extended to safety testing and performed natural circulation shutdown heat removal testing and passive power reduction in unprotected loss-of-flow sequences.

Figure 82 – Schematic and Elevation View of FFTF Primary and Intermediate Heat Transport Systems

The Clinch River Breeder Reactor (Figure 83 and Figure 84) was a joint project between the U.S. AEC/ERDA/DOE and the Electric Power Industry. The project was authorized in 1970 with funding initiated in 1972. The CRBR plant was to be a prototype and a demonstration plant for

the U.S. Liquid Metal Fast Breeder Reactor program. The CRBR had the following characteristics from Table 6:

Table 6 – Clinch River Breeder Reactor Plant Characteristics	
Reactor Core Power Level	975 MWth core – 380MWe
Fuel type	U-Pu-Oxide fuel – MOX – same as demonstrated in FFTF
Primary Plant Geometry	Loop plant (with 3 loops)
Sodium Flow Rate	100,500 gallons/minute
Primary sodium temperatures (inlet/outlet)	730F / 995F
Core Assemblies	691
Fueled Core Assemblies	198
Number of Closed Loops accommodated?	None
In-vessel refueling system	Straight pull in-vessel refueling machine with rotatable plugs (single machine)
In-vessel storage	Some, but appears limited

Figure 83 – Clinch River Breeder Reactor Plant

Figure 84 – Schematic and Elevation View of CRBR Primary and Intermediate Heat Transport Systems

A good way to understand the impacts of closed loops designed for installation in a reactor vessel is to compare the size of the reactor vessels with core power level. CRBR had virtually the same reactor vessel size as FFTF's reactor vessel, CRBR (21.5 feet reactor vessel ID) compared with FFTF (20.25 reactor vessel ID). However, CRBRP had a core power level of 975MWth compared with FFTF's 400MWth. So, in a reactor vessel with similar size, we are incorporating a core power level of more than twice the size. In fact, FFTF has 199 core assemblies comprising its core, shields, and reflectors while CRBR had 691 core assemblies comprising its core, shields, reflectors, and control assemblies. Figure 85 shows the plan view of both the CRBRP and FFTF reactor heads.

Figure 85 shows the layout of FFTF's three in-vessel transfer machines and the three instrument trees. The instrument trees provide the same functionality as the upper internal structure of the CRBRP. The instrument trees provide core instrumentation over the fueled core assemblies to monitor the health of the individual core assemblies during reactor power operation. In addition, the three instrument trees (FFTF) and upper internal structure (CRBR) provide for alignment and support of the control rod drive lines that penetrate through the reactor head and that raise and lower the control rods. FFTF has three instrument trees and three in-vessel refueling systems because of the installed closed loop technologies. When the in-vessel closed loop assemblies are installed in the reactor vessel, they provide physical interferences that need to be instrumented and refueled around and prevent the use of simpler single in-vessel transfer machines and instrument tree. Therefore, the core is divided into a tri-sector core with three instrument trees and three refueling systems each covering one-third of the core. Having this tri-sector core to accommodate the installed closed loops prevents the optimization of the equipment inside the

core and forces a larger reactor vessel to accommodate the motions needed to perform refueling and instrumented core power operations.

Figure 85 – Plan View of CRBR (Left) and FFTF (Right) Reactor Heads

The FFTF closed loop technology also requires various systems and equipment located outside of the reactor vessel but inside containment to function correctly. This additional equipment includes an independent primary heat transport system that provides the pumping power to circulate the closed loop primary coolant, a separate intermediate heat exchanger to transfer the heat generated (2MWt) to an independent intermediate heat transport loop and heat rejection system, a purification system to purify the primary coolant as needed and maintain the coolant within specification, and various valves and piping to connect everything together. This closed loop system was modularized in an integrated primary closed loop module for FFTF as shown in Figure 86 and **Error! Reference source not found.** The very compact module significantly reduced the space requirements for the closed loop module. The primary closed loop module was installed inside the FFTF reactor containment building in a nitrogen inerted cell to ensure control of the closed loop radioactive materials in case of a leak.

Figure 86 - Closed Loop Module

CLOSED LOOP SYSTEM

FOR TESTING OF FAST REACTOR FUELS AND MATERIALS IN FFTF

Figure 87 – Closed Loop System in FFTF Containment

Table 7 – Closed Loop System Components

Primary Sodium Loop	Secondary Sodium Loop
Closed Loop In-Reactor Assembly	Electro-Magnetic (EM) Pump
Branch Arm Piping	Cold Trap
Main Electromagnetic Pump	Vent/Surge Tank
Auxiliary Pump	Dump Heat Exchanger
Surge Tank	EM Sampling Pump
Vapor Trap	Aerosol Filter
Drain Tank	Magnetic Flow Meter
Auxiliary Tank	Mixing Component
Intermediate Heat Exchanger	
Cold Trap	
Sample Pump	

As mentioned above, the closed loop primary flow exits the reactor vessel and flow to the primary module inside containment. This primary module takes up space that then increases the containment size to accommodate this closed loop system. This is evident in reviewing the containment size of FFTF (400MWt) to that of CRBR (975MWt). The containment diameter of FFTF is 135 feet in diameter and the containment diameter of CRBR is 186 feet even though the core power is over double that of FFTF and the intermediate heat exchangers and primary pumps are significantly larger.

Figure 88 – FFTF Containment

Figure 89 – CRBR Containment

Other impacts on the containment design that would be impacted by flowing closed loops that are different from sodium loops, would be the increase in radiation from fluid fuel closed loops such as a molten salt closed loop. Radiation sources in primary sodium of SFR are gammas from activated sodium isotopes (Na22 and Na24), while the radiation sources of molten salt consist of gammas from FPs, actinides, activation products and spontaneous neutrons. The dose rate at the pipe inner surface in MSR is dependent on core design parameters, but the ballpark dose rate was evaluated using a similar source terms. Table below shows volumetric gammas and spontaneous neutrons in a typical 400MWth SFR. The magnitude of volumetric dose rate is 6 orders of magnitude difference between the primary coolant and used fuel assembly at discharge burnup.

Table 8 – Dose Rates for Activated Sodium and Discharge Spent Fuel Assemblies

Photons/sec/cm ³		Spontaneous neutrons/sec/cm ³
Primary sodium	Discharge fuel assembly	Discharge fuel assembly
2.87E+08	1.1E+14	2.3E+01

If the same volumetric flow rate in the closed loop is assumed, the dose rates of SFR and MSR are proportional to the volumetric activities of primary coolant and fuel assembly. For MSR, however, it is expected that the power density and discharge burnup are smaller compared to SFR. By assuming that the power density and discharge burnup are factor of 10 and 3 smaller, respectively, the volumetric activity of the MSR is a factor of 30 smaller compared to SFR. By counting the dose conversion ratio per gamma energy, the estimated dose rate at the pipe inner surface is provided in table below. The dose rate of an MSR is about a factor of 40,000 higher than the dose rate of typical SFR.

Table 9 – Comparison of Relative Dose Rates for SFR and MSR Closed Loops

	SFR	MSR	
Volumetric sources	2.87E+08 photons/sec/cm ³	~3.7E12 photons/sec/cm ³	7.8E-01 ns/sec/cm ³
Dose, mrem/hr	~1.0E7	~4.0E11	~8.4E-01

Therefore, incorporating an MSR closed loop into the design of a test reactor would require very heavy shielding in the design in addition to other protective measures that will impact that containment design.

Overall, incorporating closed loops and instrumented assemblies in a fast test reactor will impact the reactor plant design in the following way:

- The primary heat transport system and reactor vessel will be abnormally large compared with fast reactor that is designed for power generation.
- The number of invessel refueling machines will be increased to accommodate the handling of fuel around the closed loops
- The number of in-vessel upper internal structures will be increased to provide indication of flow and temperature of the sodium existing the fueled section of the core.
- The number of in-vessel storage locations will increase to accommodate because of the sector core
- The reactor containment will be abnormally large to accommodate the out-of-pile equipment for the closed loops
- The radiation hazards will increase in the reactor containment building as a result of the out-of-pile equipment for the primary closed loop coolant and will greatly increase if molten salt closed loops are installed in the fast test reactor.
- A very large ex-vessel handling machine will be need to accommodate the insertion and removal closed loop in-reactor assemblies

10 Summary

A review of irradiation testing vehicle approaches and designs that have been incorporated into past Sodium-Cooled Fast Reactors (SFRs) or envisioned for incorporation has been carried out. Fast test reactor designs examined include EBR-II, FFTF, JOYO, BOR-60, PHÉNIX, JHR, and MBIR. Previous designers exhibited great ingenuity in overcoming design and operational challenges. The various irradiation testing vehicles can be categorized as: Uninstrumented open assemblies that fit into core locations; Instrumented open test assemblies that fit into special core locations; Self-contained closed loops; and External closed loops. External closed loops offer the greatest capability especially for alternate closed loop coolants other than sodium. However, external closed loops also involve the greatest cost and will distort the design of the fast test reactor when compared with a demonstration reactor.

Acknowledgements

Argonne National Laboratory's work was supported by the U. S. Department of Energy Advanced Reactor Technologies (ART) Program under Prime Contract No. DE-AC02-06CH11357 between the U.S. Department of Energy and UChicago Argonne, LLC. The work presented here was carried out under the Fast Reactor Technology area of the ART Program. The authors are grateful to Bob Hill (ANL/NE), the National Technical Director, as well as Alice Caponiti (U.S. DOE), Headquarters Program Manager for the project. The authors are especially thankful to Dr. Earl Feldman (Argonne National Laboratory/Nuclear Engineering Division) for discussions on EBR-II and for loaning his original copy of the "Guide for Irradiation Experiments in EBR-II," Dr. Tomonori Soga (Japan Atomic Energy Agency/O-arai Research and Development Center) for providing information on the irradiation testing vehicles in JOYO, Dr. Olivier Gastaldi (CEA/Cadarache) for providing an update on the status of JHR and its closed loops, and Dr. Ron Omberg for providing information on FFTF closed loop technology.

Nuclear Engineering Division

Argonne National Laboratory
9700 South Cass Avenue, Bldg. 208
Argonne, IL 60439

www.anl.gov

Argonne National Laboratory is a U.S. Department of Energy
laboratory managed by UChicago Argonne, LLC