FASTER Test Reactor Preconceptual Design

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C. Grandy[1], H. Belch[1], A. Brunett[1], S. Hayes[2], F. Heidet[1], R. Hill[1], E. Hoffman[1], E. Jin[1], W. Mohamed[1], A. Moisseytsev[1], S. Passerini[1], J. Sienicki[1], T. Sumner[1], R. Vilim[1]

[1]Argonne National Laboratory (ANL), Argonne, USA

[2] Idaho National Laboratory, Idaho Falls USA

E-mail contact of main author: cgrandy@anl.gov

Abstract. The FASTER test reactor was designed as part of the U.S. Advanced Demonstration and Test Reactor Options (ADTR) Study in 2015/2016 [1]. The ADTR study provided an assessment of advanced reactor technology options and is intended to provide a sound comparative technical context for future decisions concerning these technologies. Point designs for a select number of concepts were commissioned.

One of the two test reactor point designs was a sodium-cooled fast test reactor called FASTER [2]. FASTER is a sodium-cooled, metal alloy fueled fast reactor with a core thermal power rating of 300MW. The FASTER plant was designed with extended testing capabilities in mind while trying to keep the reactor plant as simple as possible. The main function of the FASTER plant is to provide high neutron flux irradiation capability for both fast neutron spectrum and thermal neutron spectrum applications.

The FASTER reactor plant incorporates an innovative core arrangement that also provides for irradiation testing in closed loops with different working fluids. This paper will describe the design characteristics of the FASTER plant and provide background information on the ADTR study and its objectives.

Key Words: Fast Test Reactor, Sodium-Cooled Fast Reactor.

1. Introduction

The FASTER reactor plant is a 300MWt/120MWe sodium-cooled fast spectrum test reactor that provides high levels of fast and thermal neutron flux for scientific research and development and to meet the performance objectives of the U.S. ADTR study. The 120MWe FASTER reactor plant has a superheated steam power conversion system which provides electrical power to a local grid allowing for recovery of operating costs for the reactor plant. In addition, the FASTER reactor plant could be used for isotope production or as a heat source, if desired.

The reactor power level is the minimum that assures achievement of the neutron flux goals. In its current configuration (Figure 1), the FASTER reactor provides 33 fast flux test locations, three (3) thermal flux test locations, two (2) fast flux closed loops and one (1) thermal flux closed loop [3]. Among the fast spectrum test locations, four of them are located near the core center and cannot be repositioned without affecting the core neutronics performance.

It is anticipated that the FASTER reactor plant will be utilized by domestic and international researchers with its broad appeal to many different reactor types: sodium-cooled fast reactors, lead-cooled fast reactors, gas-cooled fast reactors, and thermal spectrum reactors.

It is estimated that the FASTER test reactor will require approximately 11 to 13 years from the issuance of CD-0 to the core startup assuming funding and licensing are not limiting factors. In addition, the FASTER test reactor with the steam plant will cost approximately $2.8B (with a 30% contingency) to design (~$1.1B) and construct (with each closed loop contributing ~$100M (includes contingency) to overall estimated TPC). If it was decided to remove the steam plant and just dump the 300MWt of heat to the atmosphere, then the cost will be significantly less than $2.5B. The annual FASTER reactor plant operating costs are estimated to be less than $100M. The FASTER reactor plant annual operating costs including irradiation operations are expected to be less than $150M (using FFTF as the high end basis). All estimates are in 2016 dollars. The replacement fuel is estimated to cost about $20M/year. The FASTER reactor is expected to achieve a capacity factor of 80% or greater while putting power on the grid. The sales from this power are expected to be around $89M to $100M per year depending upon overall electrical generation capacity and power purchasing agreements, offsetting the operational and fuels costs.

2. FASTER Design Approach and Fuel Selection

The FASTER plant has been designed with extended testing capabilities in mind, while trying to keep it as simple as possible in order to make it attractive and cost efficient. The main function of the reactor is to provide neutrons for irradiation testing and thus no significant technology innovations were adopted for the FASTER reactor plant to maintain a high technology readiness level. The FASTER reactor plant will rely upon the liquid metal base technology developed in the U.S. for EBR-II, FFTF, CRBR, and the ALMR program with a special emphasis on the irradiation testing capabilities developed for EBR-II and FFTF. The FASTER reactor core design discussed here is not based on any previously existing fast reactor, but uses materials and dimensions consistent with the U.S. base technology program. The main objective of the FASTER reactor design efforts was to achieve a very high fast flux as well as a significant thermal flux while offering a large number of test locations.

Ternary metallic fuel, U-Pu-Zr, is used with HT-9 stainless steel for cladding and structural material. Although there is no mandated limit on the weight fraction of Pu that can be used in the fuel, it was decided to limit it to 20wt% based on the availability of irradiation data. Another incentive for not resorting to higher Pu wt% is the degradation of the fuel thermal conductivity as Pu content is increased. This is of particular importance for the FASTER reactor due to the high power density during operations.

In order to optimize the reactor performance and obtain a relatively compact core, the Zr wt% in the fuel is assumed to be 6wt% and the fuel smear density is assumed to be 85%. Using 6wt% instead of the more traditional 10wt% does not affect the characteristics of the ternary fuel and irradiation tests have previously been performed for such a fuel type. The decision to use an 85% smear density, instead of the 75% typically used for metallic fuel, is based on the relatively low peak burnup that will be achieved. Because of the lower fuel burnup, the internal stress applied by the fuel on the cladding, as a result of irradiation swelling, will be less important than typically observed in metallic fuel that reaches a high burnup. Furthermore, the fission gas plenum length relative to the active fuel length does not need to be as long as what is typically used in SFR core designs, because of the lower fuel burnup achieved. For the FASTER core design the fission gas plenum length is set to be 65% of the active fuel length.

3. FASTER Test Reactor Point Design Description

Table 1 provides summary characteristics for the FASTER reactor plant.

Table 1 – FASTER Reactor Plant Summary Characteristics

Reactor Power	300MWt / 120MWe/40% efficiency
Coolant	Sodium
Coolant Temperature	510°C / 355°C
Coolant Pressure (cover gas pressure pressure)	Cover Gas pressure – few inches of water
Fuel, Cladding, Duct	U-Pu-Zr metal fueled core, HT-9, HT-9
Cycle Length	100 days
Average burn-up	34.3 GWd/ton
Power density (average, peak)	558.8 W/cc, 917 W/cc
Plant Life	30 years with expectation of life extension
PHTS Configuration	Pool plant geometry
Reactor vessel structural materials	Austenitic stainless steel
Primary and Secondary Pumps	Mechanical centrifugal pumps (2)
Intermediate Heat Exchanger	Tube-and-Shell heat exchanger (4)
Reactor Vessel Support	Conical Ring – Top Support
Emergency Decay Heat Removal	Direct Reactor Auxiliary Heat Exchanger in cold pool (3)
Primary Purification System	Conventional cold and nuclide trap technology
Power Conversion System	Superheated steam cycle
Containment	Steel concrete reinforced containment
In-vessel Fuel Handling Mechanism	Single Rotatable Plug with pantograph FHM (3)

3.1 Core Layout and Assemblies Description

The 300 MW_{th} FASTER core, shown in Figure 1, is composed of 55 fuel assemblies, each with the same Pu wt fraction. The fuel, coolant and structural material volume fractions are 30.93%, 37.36%, and 23.65%, respectively. The active fuel height is 80 cm. Six primary control rod assemblies and three secondary control rod assemblies composed of B_4C rods ensure the safe shutdown of the core. There are 33 fast neutron flux test locations, in addition to the two closed loops also being exposed to a fast neutron flux. The fuel assembly positions have been chosen to enhance neutron leakage probability toward the moderated zone (brown in Figure 1). The purpose of the moderator is to take advantage of the neutrons leaking out of the active core region and thermalize them in order to provide thermal spectrum testing capabilities. With the current design, fast neutrons are thermalized by the moderator and do not return into the active core region because of the reflector layer between the two regions. This design approach prevents a number of potential issues. There are three thermal test locations and one closed loop having a thermal neutron flux. Canned beryllium is used as the moderator and zircaloy is used as the structural material in that region to avoid parasitic absorption of thermal neutrons in iron. The moderated region does not contain any fuel and is cooled with sodium.

The innovative fast and thermal core performance characteristics are described in a companion FR17 conference paper [3].

The reactor is to be operated in a three fuel batch management scheme with a cycle length of 100 effective full power days (EFPD). At the end of a cycle, one third of the fuel assemblies, having the highest burnup, are discharged and replaced with fresh fuel assemblies. The fuel assemblies remaining in the core are not shuffled.

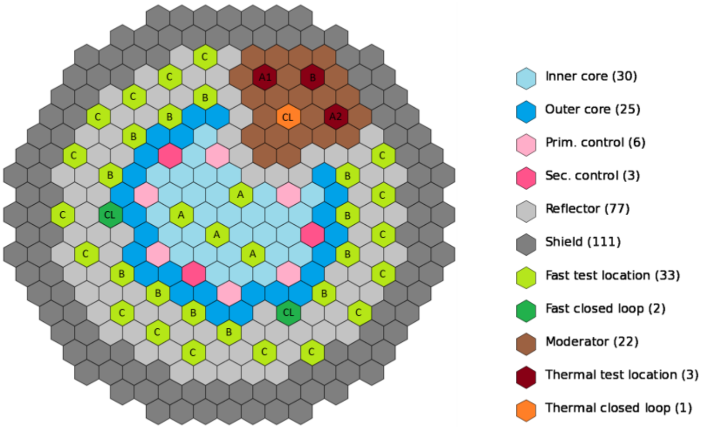

Figure 1 – FASTER Reactor Core Layout

3.2 FASTER Plant Design

Within the primary pool plant geometry, the primary heat transport system (PHTS) includes the primary pumps (2), the reactor core, the intermediate heat exchangers (4), and various structures and connections between these components (Figure 2). The primary pumps are mechanical centrifugal pumps that provide 758.3 m3/s flow rate at 704 kPa discharge head. The pumps are 90% efficient and are 9.2m long and 0.9m in diameter (pump casing)

The IHXs are conventional sodium-to-sodium tube-and-shell heat exchangers that allow the primary (hot) sodium to flow through the shell side of the IHX and provide sensible heat to the secondary sodium that flows through the tube side of the IHX. The IHXs each provide 75MWt heat transfer capacity and are design with a 25% thermal margin. There are 1,200 tubes with an effective tube length of 3.85m. The tube material is 9Cr-1Mo steel.

The intermediate heat transport system (IHTS) (**Figure 3**) consists of centrifugal (2) mechanical pumps, two helical coil steam generators (HCSGs), the tube side of the IHX, and interconnected piping. The IHTS is protected from overpressure by a sodium-water reaction protection system in case of a steam generator tube leak.

The normal shutdown heat removal path is through the PHTS, through the IHTS, and through the steam plant bypassing the turbine and dumping the steam to the main condenser. This heat removal path can provide for all heat removal capabilities needed when electrical power exists.

Primary and secondary sodium coolant is purified in separate cold trap systems. In addition, the primary sodium system has a nuclide trap for the specific removal of cesium and other radionuclides that may result from cladding breach testing. The cover gas purity is maintained by an argon cover gas supply and purification system, for both the PHTS and IHTS.

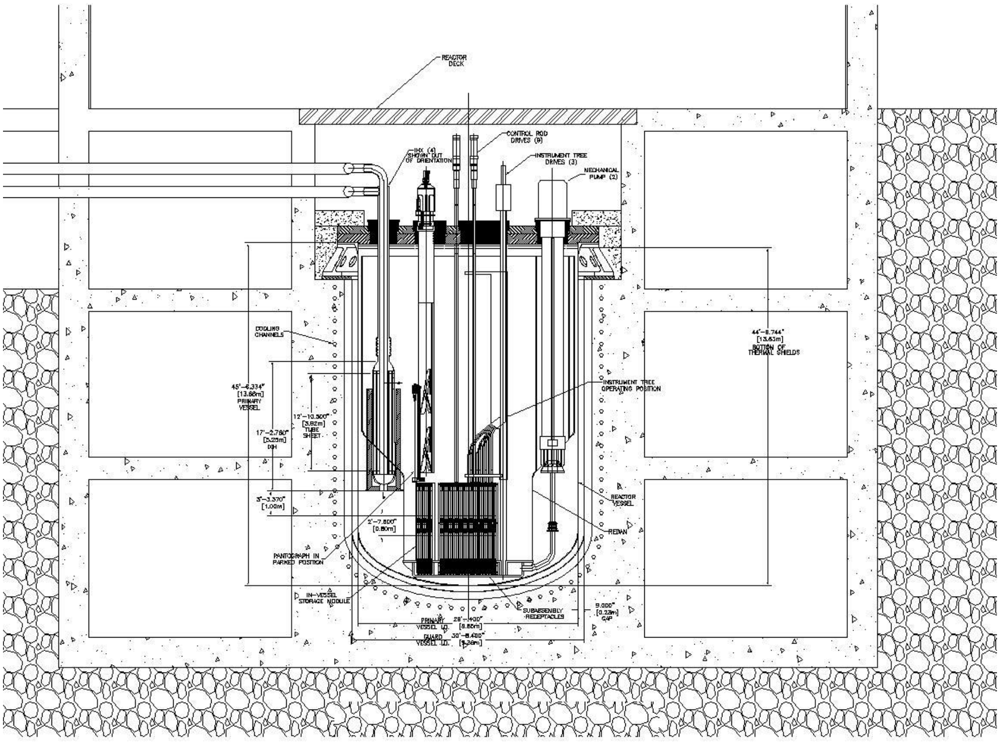

Figure 2 – Elevation View of PHTS

The containment design is a low leakage steel reinforced concrete containment that is designed for all internal and external threats while minimizing the release of radionuclides to the environment during design basis and beyond design basis accidents.

Emergency decay heat removal is provided through three 750 kW (each) independent direct reactor auxiliary cooling system (DRACS) loops that allow for the passive removal of emergency decay heat from the primary heat transport system. The DRACS heat exchanger (3) (a tube-and-shell HX) is submerged in the FASTER reactor vessel cold pool. It is connected via piping to an air dump heat exchanger (ADHX) located outside of containment. Dampers on the ADHX minimize the parasitic losses from the emergency decay heat removal system during normal operation and will open fully upon a protective signal or loss of power. Two of the three DRACS heat exchangers are required for emergency decay heat removal.

The balance of plant consists of a conventional superheated steam cycle attached to the (2) once-through sodium heated steam generators to put power on the grid. Conditions are calculated with the GateCycle software (**Figure 4**).

3.3 Fuel Handling

There are three sets of in-vessel transfer machines (IVTM) that perform the refueling function within the reactor vessel pool. In order to move fuel from an in-core location to the storage position, the upper internal structure segment is rotated from above the core and placed in a parked position so that the IVTM can reach its 120° sector of the core. There are three in-vessel storage locations associated with each 120° sector of the core. The ex-vessel transfer machine (EVTM) is designed to remove spent core assemblies from the core and transfer them to a transfer position. The EVTM is also designed to maintain the spent core assembly at the correct temperature with active cooling and is designed to handle fresh core assemblies and insert them into the reactor vessel.

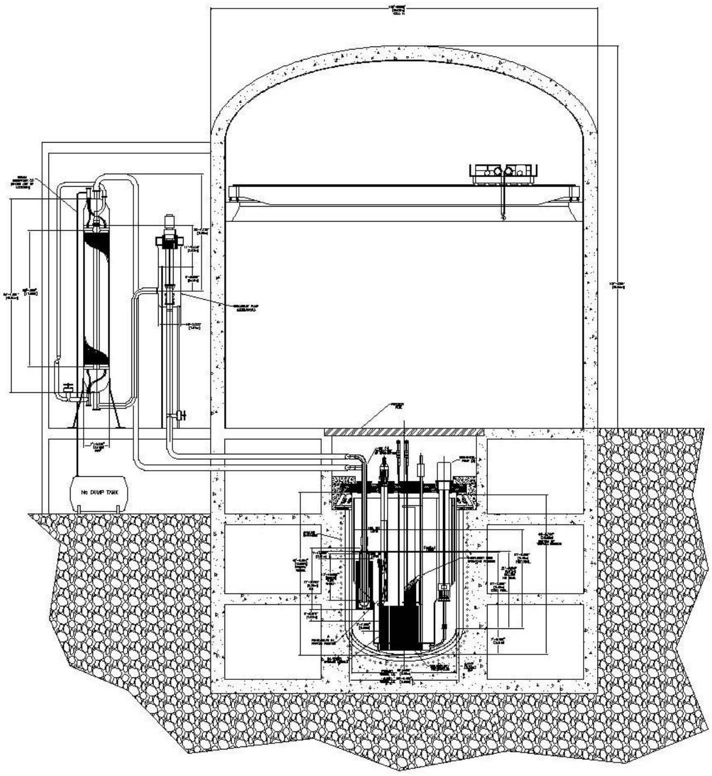

Figure 3 – FASTER NSSS – Elevation View

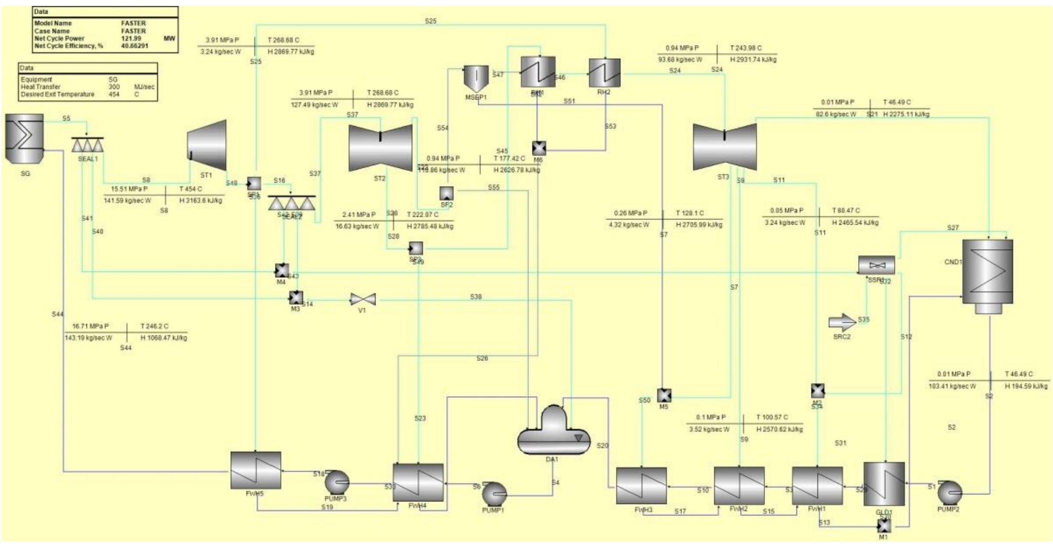

Figure 4 – FASTER Thermodynamic Cycle and Balance of Plant

3.4 Test Assembly Flux Levels and Volumes

In its current configuration (Figure 1), the FASTER core provides 33 fast flux test locations, three (3) thermal flux test locations, two (2) fast flux closed loops and one thermal flux closed loop. Among the fast spectrum test locations, four of them are located near the core center and cannot be repositioned without affecting the core neutronics performance. The other 29 fast flux test locations are located in the radial reflector region and their position can be changed without significantly affecting the core performance. In fact, any of the reflector assembly locations could be used as a test location without having any significant impact on the core performance. In a similar way, the number of thermal flux test locations could be increased by replacing reflector assemblies with moderator and thermal test assemblies. This would result in a reduction of the number of fast flux test locations. It is important to note that the closed loop and instrumented irradiation positions are fixed because the fuel handling machines and instrumentation trees have been designed around these fixed core positions.

The core assembly length is estimated to be ~2.77 m. The actual test length will depend on the test assembly design; in particular, the length of the lower adaptor and core handling socket. The likely resulting effective test length will be around two meters, corresponding to an available test volume of ~24 liters in each test location. The total test volume in the current core configuration is about 0.95 m^3. The flux level achieved in a test assembly depends on its distance from the core center, as well as on its composition. Given that the materials to be tested are currently undetermined, the flux levels provided here were obtained when test locations are filled with a reflector assembly (80% steel, 20% coolant).

The normalized axial fast flux profile is shown in **Figure 5** for a test assembly located in the active core region and for a test assembly located in the reflector region. The characteristics of the fast flux test assemblies based on their flux values and their characteristics are summarized in **Table 2**. In order to provide a measure of the total irradiation capacity available, the total fast fluxes are multiplied by the test volumes. This captures the fact that the fast flux near the extremities of the test location is significantly smaller than near the center and that increasing the test length without increasing the active core length will not significantly increase the irradiation capacity.

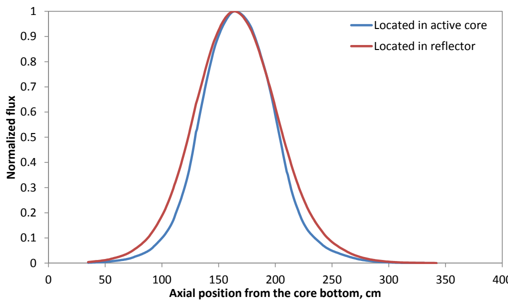

Figure 5 – Normalized axial fast flux distribution in test locations

Table 2 – Summary of Fast Flux Conditions in the Test Assemblies

Group	Number of assemblies	Peak fast flux range (10^{15} n/cm^2·s)	Fast flux*Volume range (10^{19} n·cm/s)	Total fast flux*Volume (10^{19} n·cm/s)
A	4	4.7-5.2	4.0-4.9	17.9
Closed loops	2	2.3	1.7	3.4
B	10	1.9-2.7	1.4-2.1	15.8
C	19	0.3-1.3	0.2-0.9	10.3

In the thermal flux test assemblies and thermal closed loop, the fast flux level is not relevant and the thermal flux level is provided instead. It is important to note that the thermal neutrons were defined as all neutrons having an energy lower than 0.1 eV. By using the energy threshold later established as part of the ATDR study framework (0.625 eV), these thermal flux values would be two to three times larger.

The peak thermal flux values calculated in the closed loop and three test assemblies located in the moderated region are provided in **Table 3** for each location individually. The peak value is typically achieved near the side of the assembly that is facing the active core region (i.e., where the neutrons are coming from). The thermal flux is radially reduced by a factor of ~2 across an assembly, for a given axial position. The normalized axial thermal flux distribution is shown in **Figure 6**. The rough aspect of the curve is due to the uncertainties of the calculations performed with MCNP.

Table 3 – Summary of Thermal Flux Conditions in the Test Assemblies

Location	Peak thermal flux (10^{14} n/cm^2·s)	Thermal flux*Volume (10^{18} n/cm^2·s)
Closed loop	5.8	3.7
A1	1.9	1.2
A2	1.9	1.2
B	1.7	1.1

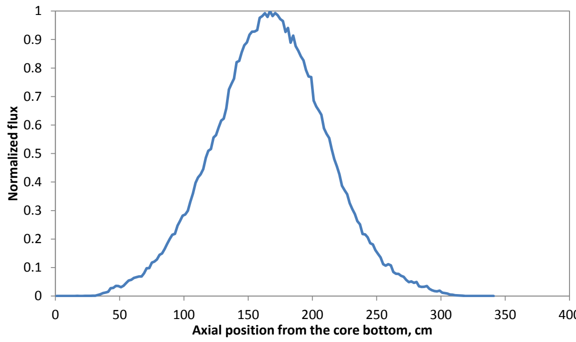

Figure 6 – Normalized axial thermal flux distribution in the test locations

Closed Loop Systems

The three closed loop systems (CLS) are an important capability of FASTER and part of the ATDR study scoring metrics. They enable FASTER to be utilized to irradiate and test fuels and materials in a prototypical flowing coolant environment with different coolants for different reactor types. The closed loop (CL) testing capability goes beyond just fuels and materials testing. Each CLS with a different coolant is a demonstration of that particular coolant and its technology inside of an operating nuclear reactor. Thus, one has an integrated demonstration of fuel, core materials, coolant, coolant chemistry control, and optionally coolant cleanup technologies under prototypical conditions in either a fast or thermalized neutron spectrum, as appropriate. For a different reactor coolant than sodium, this can be a test and demonstration as well as an approach to increasing the TRL level for the fuel, materials, and coolant technologies for far less cost than designing, building, and operating a separate nuclear reactor with those fuels, materials, and coolant technologies. The closed loop approach might reveal unanticipated problems with a different reactor technology for a far less expense than designing, building, and operating a separate reactor.

CLSs incorporating sodium were an integral part of the FFTF design [4] that could have simultaneously incorporated four such CLs. Two compact integrated closed loop primary modules were actually built and one was installed in a cell inerted with nitrogen inside of the FFTF containment. None of the CLs at FFTF were actually used, however, during its 10 year operating life. For irradiation and testing with flowing coolants at different conditions other than the main primary coolant flow, the closed loop approach is essential. Details of FFTF closed loops are discussed in another FR17 paper [4]

For FASTER, heat removal requirements for different coolants and reactor configurations were first investigated assuming that each CLS can accommodate a test section inside of a flow tube having an inner diameter of 6.985 cm (2.75 inch) and a closed loop heat rejection rate capability of 2.3 MW_t per loop, similar to the CLS designs for FFTF. Heat removal rate and coolant flowrate requirements for different coolants for different example reactor designs are shown in **Table 4**. For nominal steady state temperature and velocity conditions, the heat removal rate capability of 2.3 MW_t is sufficient. A single possible exception is the Pebble Bed Fluoride-Salt-Cooled High-Temperature Reactor (PB-FHR) for which it might be necessary to slightly reduce the size of the core mockup to reduce the power deposition below the indicated 2.2 MW_t. The 2.3 MW_t heat rejection rate generally provides some margin for transient testing that can include greater power deposition rates than at nominal steady state.

Table 4 - Heat Rejection Rate and Flowrate Requirements for Closed Loops for Different Reactor Coolants and Example Reactor Designs

Coolant	Sodium	Sodium	Lead, Pb	Liquid Salt, FLiBe, 2LiF-BeF_2	Liquid Salt, FLiBe, 2LiF-BeF_2	Pressurized Helium	Pressurized Water	Pressurized Water
Reactor	PGSFR for Nominal Conditions	PGSFR for Unprotected Transient Overpower Conditions	LFR with High Core Outlet Temperature	ORNL AHTR	UCB Pebble Bed FHR	GA Prismatic HTGR	WEC AP1000	High Flux Isotope Reactor (HFIR)
Flow Direction	Up	Up	Up	Up	Up	Down	Up	Down
Flow Area Fraction Inside Reactor Core	0.38	0.38	0.599	0.15	0.60	0.187	0.531	0.50
Coolant Inlet Pressure, MPa	Near Atmospheric	Near Atmospheric	Near Atmospheric	Near Atmospheric	Near Atmospheric	6.39	15.5	2.24
Coolant Outlet/Inlet Temperatures, °C	547/395	738/395	650/400	700/650	700/600	750/322	321/281	67.8/57.2
Coolant Inlet	5.52	5.52	2.0	1.94	2.0	20.2	4.85	15.5

Velocity, m/s								
Coolant Mass Flowrate, kg/s	6.91	6.91	48.5	2.18	9.14	0.0737	7.53	29.4
Coolant Volume Flowrate, m3/s (gpm)	0.00804 (127)	0.00804 (127)	0.00459 (72.8)	0.00111 (17.6)	0.00460 (72.9)	0.0145 (229)	0.00986 (156)	0.0298 (472)
Power Removed by Coolant, MWt	1.33	2.98	1.75	0.263	2.21	0.164	1.66	1.30

Next, the feasibility of designing closed loop in-reactor assemblies for different coolants and reactor configurations was examined. It is assumed that the pressure boundary of the in-reactor assembly is a double-walled pressure tube. The incorporation of a double-walled pressure tube is viewed as a necessary and sufficient approach to incorporate coolants other than sodium inside of a SFR. Required wall thicknesses for each of the two pressure tubes were calculated using the formulae and tables in the ASME Boiler and Pressure Vessel Code Section III, “Rules for Construction of Nuclear Facility Components,” Division 1-Subsection NH, “Class 1 Components in Elevated Temperature Service,” 2001 Edition. The lifetime of each in-reactor assembly is assumed to be 10,000 hours which is a 4 % margin over the duration of four FASTER operating cycles. The outer tube outer diameter of 11.26 cm (4.44 inches) is assumed identical to that of the hexcan duct-to-duct inner distance for a FASTER fuel assembly. The outer tube outer diameter is the largest value that can fit inside of an assembly location in the FASTER core with clearances filled with sodium between the outer tube and the hexcans of the six neighboring core assembles. For the low pressure coolants (sodium, lead, and pressurized water under HFIR conditions), the design pressure is taken equal to the same value for the in-reactor assemblies in FFTF (2.5 MPa = 363 psig). The case of liquid salt coolant is not analyzed because a suitable structural material has not yet been codified in the ASME code. For helium and pressurized water under PWR conditions, the design pressure is assumed to be 10 % greater than the values assumed in **Table 4**. The required pressure tube dimensions for a design temperature of 649 °C (1200 F) are shown in Table 5. For the low pressure coolants, the required wall thicknesses of the outer pressure and inner pressure tubes are 2.51 mm (0.0986 in) and 2.25 mm (0.0887 in), respectively. To ensure against concerns about potential buckling of the pressure tubes under external pressure, effects of irradiation, and other uncertainties, the wall thicknesses are increased to a minimum of 6.35 mm (0.25 in). The inner tube inner diameter of 8.09 cm (3.18 in) provides plenty of space for a flow tube to separate downward and upward flows and a test section inside of the flow tube. For pressurized helium coolant, the inner tube inner diameter of 8.16 cm (3.21 in) also provides ample space. For pressurized water under PWR conditions, there is space for a flow tube and test section but the number of fuel pins would need to be reduced below that implied by the assumptions in **Table 4**.

Table 5 – Required Pressure Tube Dimensions for 649 °C (1200° F) Design Temperature

Coolant	Sodium, Lead, or Low Pressurized Water	Sodium, Lead, or Low Pressurized Water with 0.25 in Wall Thicknesses	Pressurized Helium	Highly Pressurized Water
Pressure Tube Material	316	316	800H	316
Design Gauge Pressure, MPa (psig)	2.50 (363)	2.50 (363)	7.82 (1019)	17.05 (2473)
Design Temperature, °C (F)	649 (1200)	649 (1200)	649 (1200)	649 (1200)
Design Lifetime, hours	10,000	10,000	10,000	10,000
Outer Pressure Outer Diameter, cm (in)	11.26 (4.443)	11.26 (4.443)	11.26 (4.443)	11.26 (4.443)
Outer Pressure Tube Wall Thickness, cm (in)	0.251 (0.0986)	0.635 (0.25)	0.677 (0.266)	1.850 (0.728)
Outer Pressure Tube Inner Diameter, cm (in)	10.76 (4.236)	9.990 (3.933)	9.907 (3.900)	7.560 (2.976)
Gap Between Pressure Tubes, cm (in)	0.318 (0.125)	0.318 (0.125)	0.318 (0.125)	0.318 (0.125)
Inner Pressure Tube Outer Diameter, cm (in)	10.12(3.986)	9.355 (3.683)	9.272 (3.650)	6.925 (2.726)
Inner Pressure Tube Wall Thickness, cm (in)	0.225 (0.0887)	0.635 (0.25)	0.557 (0.219)	1.14 (0.448)
Inner Pressure Tube Inner Diameter, cm (in)	9.673 (3.808)	8.085 (3.183)	8.157 (3.212)	4.649 (1.830)

For liquid salt and pressurized helium coolant, it is desirable to achieve higher temperatures. For a design temperature of 704°C, ample space is still available with the low pressure and pressurized helium coolants (Table 6). There still remains space when the design temperature is further increased to 760°C as shown in Table 7. The test sections in the FFTF closed loop in-reactor assemblies were designed for a sodium outlet temperature of 760°C while the double-walled pressure tube and other closed loop hardware was designed for 649°C. This was achieved by bypassing part of the upward sodium flow around the test section in the annular space between a cylindrical thermal baffle surrounding the test section and the flow tube separating the downward and upward sodium flows inside of the pressure tube. An alternate approach that permits more space for a test section is to design the entire in-reactor assembly for a greater temperature and mix the outlet coolant with a cooler coolant bypass stream inside of a mixing component outside of the reactor.

Table 6 – Required Pressure Tube Dimensions for 704 °C (1300 F) Design Temperature

Coolant	Sodium, Lead, or	Sodium, Lead, or	Pressurized	Highly

	Low Pressurized Water	Low Pressurized Water with 0.25 in Wall Thicknesses	Helium	Pressurized Water
Pressure Tube Material	316	316	800H	316
Design Gauge Pressure, MPa (psig)	2.50 (363)	2.50 (363)	7.82 (1019)	17.05 (2473)
Design Temperature, °C (F)	704 (1300)	704 (1300)	704 (1300)	704 (1300)
Design Lifetime, hours	10,000	10,000	10,000	10,000
Outer Pressure Outer Diameter, cm (in)	11.26 (4.443)	11.26 (4.443)	11.26 (4.443)	11.26 (4.443)
Outer Pressure Tube Wall Thickness, cm (in)	0.456 (0.179)	0.635 (0.25)	1.08 (0.427)	3.61 (1.42)
Outer Pressure Tube Inner Diameter, cm (in)	10.35 (4.074)	9.990 (3.933)	9.091 (3.579)	4.033 (1.588)
Gap Between Pressure Tubes, cm (in)	0.318 (0.125)	0.318 (0.125)	0.318 (0.125)	0.318 (0.125)
Inner Pressure Tube Outer Diameter, cm (in)	9.714 (3.824)	9.355 (3.683)	8.456 (3.329)	3.398 (1.338)
Inner Pressure Tube Wall Thickness, cm (in)	0.393 (0.155)	0.635 (0.25)	0.814 (0.321)	1.09 (0.429)
Inner Pressure Tube Inner Diameter, cm (in)	8.928 (3.515)	8.085 (3.183)	6.827 (2.688)	1.217 (0.4793)

Table 7 - Required Pressure Tube Dimensions for 760 °C Design Temperature

Coolant	Sodium, Lead, or Low Pressurized Water	Pressurized Helium
Pressure Tube Material	316	800H
Design Gauge Pressure, MPa (psig)	2.50 (363)	7.82 (1019)
Design Temperature, °C (F)	760 (1400)	760 (1400)
Design Lifetime, hours	10,000	10,000
Outer Pressure Outer Diameter, cm (in)	11.26 (4.443)	11.26 (4.443)
Outer Pressure Tube Wall Thickness, cm (in)	0.905 (0.356)	1.85 (0.726)
Outer Pressure Tube Inner Diameter, cm (in)	9.450 (3.720)	7.569 (2.980)
Gap Between Pressure Tubes, cm (in)	0.318 (0.125)	0.318 (0.125)
Inner Pressure Tube Outer Diameter, cm (in)	8.815 (3.470)	6.934 (2.730)
Inner Pressure Tube Wall Thickness, cm (in)	0.709 (0.279)	1.14 (0.447)
Inner Pressure Tube Inner Diameter, cm (in)	7.398 (2.913)	4.662 (1.835)

The CLS for each alternative (non-sodium) coolant incorporates an in-reactor assembly with a test section, a primary loop with the particular coolant for irradiation and testing, a secondary loop with an appropriate secondary coolant for heat transport, a primary coolant-to-secondary coolant IHX, a secondary coolant-to-air DHX for heat rejection to the atmospheric heat sink, and interconnecting piping. Six different CL primary coolants have been included thus far in the FASTER design; others can be added in the future. The six primary coolants and the major features of the CLS for each are shown in **Table 8**. For sodium, lead or lead-bismuth eutectic (LBE), liquid salt, and helium, each primary CL in-reactor assembly is designed for a maximum temperature of 760 °C (1400 °F). For sodium, lead or LBE, liquid salt, and helium primary coolants, sodium is used as the secondary coolant to reject heat to air. A single secondary coolant, sodium, is utilized because it is a low pressure coolant and because of its low freezing temperature, excellent heat transfer properties, excellent compatibility with stainless steel and other alloys, and to avoid the cost of designing and installing a secondary loop and secondary DHX for a different fluid. Sodium is not used for the pressurized water primary coolants to provide separation between sodium and water components and piping, and because heat rejection for primary water coolant can occur at temperatures below or above but near the sodium freezing temperature.

Table 8 - Closed Loop System Primary Coolants and Major Features

Primary Coolant for In-Reactor Irradiation and Testing	Sodium	Lead, Pb, or Lead-Bismuth Eutectic, 45 wt % Pb-55 wt % Bi	Liquid Salt, FLiBe, $2LiF\text{-}BeF_2$	Pressurized Helium	Pressurized Water for NPP Conditions	Pressurized Water for Research and Test Reactor Conditions
Secondary Coolant	Sodium	Sodium	Sodium	Sodium	Pressurized Water	Pressurized Water
Primary Materials	316H, 316	ALD-Coated 316H and 316	Hastelloy N	800H	Low Alloy and Carbon Steel with Stainless Steel Cladding	Low Alloy and Carbon Steel with Stainless Steel Cladding
Secondary Materials	316H, 316	316H, 316	316H, 316	316H, 316	Low Alloy and Carbon Steel with Stainless Steel Cladding	Low Alloy and Carbon Steel with Stainless Steel Cladding
Intermediate Heat Exchanger	Single-Walled Tube Helical Coil Similar to FFTF Closed Loop System Design	Double-Walled Straight Tube to Preclude Leakage	Double-Walled Straight Tube with Hastelloy N Tubes to Preclude Leakage	Double-Walled Straight Tube to Preclude Leakage	Single-Walled Tube Helical Coil	Single-Walled Tube Helical Coil
In-Reactor Assembly	Single-Wall Flow Tube	Double-Wall Flow Tube with Monitored Gap	Double-Wall Flow Tube with Monitored Gap to	Double-Wall Flow Tube with Monitored Gap to	Double-Wall Flow Tube with Monitored Gap	Double-Wall Flow Tube with Monitored Gap

		to Preclude Leakage	Preclude Leakage	Preclude Leakage	to Preclude Leakage and for Thermal Insulation	to Preclude Leakage and for Thermal Insulation
Primary Coolant Pumps	Electromagnetic	Electromagnetic	Electromagnetic	Centrifugal/Radial Pump	Canned Rotor	Canned Rotor
Primary Coolant Chemistry Control and Cleanup	Cold Trap, Plugging Meter Measurements	Intermixing with Hydrogen to Reduce Oxygen Content, Oxygen Sensor Measurements	Redox Potential Control, Tritium Stripping and Capture	Makeup for Coolant Leakages, Minimal Chemistry Control	pH Control, Mixed Bed Demineralizers, Cation Bed Demineralizer, Control of Radiolysis Reactions	pH Control, Mixed Bed Demineralizers, Cation Bed Demineralizer, Control of Radiolysis Reactions
Primary Coolant Loop Cell Volume Normalized by FFTF Closed Loop Primary Cell Volume	1	1	3	1	3	3

It is necessary to prevent leakages of other primary coolants into the primary sodium. Lead, LBE, or liquid salt leaking into sodium could attack structural materials such as 316SS. To preclude leakages, the pressure tube of the in-reactor assembly is made double-walled with a gap between the two walls that is monitored for leakage. The primary coolant-to-sodium CLS IHX is a double-walled straight tube (DWST) HX to preclude leakage. For helium primary coolant, a double-walled pressure tube with a gap is provided to preclude leakage of helium into sodium that might result in the formation of bubbles that could enter the core with reactivity effects and to preclude a blowdown of high pressure helium into the reactor vessel sodium. A DWST CLS IHX is utilized to preclude a blowdown of high pressure helium into secondary sodium. For pressurized water primary coolant, a double-walled pressure tube is needed to preclude water/steam leakage into reactor vessel sodium or a blowdown of high pressure water/steam into surrounding sodium and sodium-water reactions. The gap between the two walls will also incorporate a vacuum to reduce heat transfer from the hotter surrounding sodium to water. In particular water at research and test reactor conditions will be significantly cooler than the surrounding reactor vessel sodium. The gap between the two walls will be monitored for leaks.

The CLS design for each coolant type and the fast reactor containment design must accommodate the effects of postulated CLS accidents resulting in the inability to remove heat from the in-reactor assembly. For the FFTF CLS design, the in-reactor assembly was designed to accommodate a Test Section Meltdown Accident (TSMDA). A meltdown cup was provided below the bottom cup end of the pressure tube. The meltdown cup was designed to contain 0.75 liter (46 inch^3) of molten UO_2 fuel. It is expected that FASTER will incorporate similar capabilities.

3.8 FASTER Testing Under Prototypical Conditions

Specific core locations and their associated *instrumented assemblies* provide an online monitoring and measurement capability for irradiation experiments. This meets the basic requirement of an irradiation testing facility, that it provide for irradiation and testing of fuels, materials and specimens under prototypical reactor conditions with continuous monitoring of quantities of interest (e.g., temperature and flow rate). The monitoring capability is enabled by dedicated instrument lines which reach each assembly through dedicated experimenters' leads from the center island of the reactor head. Seven locations for independently instrumented assemblies are envisioned for the FASTER design. Instrumented assemblies use a standard fuel duct with an attached stalk to guide the instrumented lines. Flow is controlled with an inlet orifice. Instrumented assemblies were also part of the FFTF design (there they were referred to as *open test assemblies*) which represents a good starting point as the base technology for the FASTER instrumented assemblies. In FASTER, the instrumented subassemblies will support three types of experiments:

1. Encapsulated Fuel Element Experiments: These types of experiments are meant to characterize and test materials that are first introduced in reactor for testing and whose behavior under irradiation has not been fully characterized yet. Therefore, those experiments need to be enclosed in ad-hoc capsules to avoid any release of material or reaction with the coolant. This category includes capsules that: a) contain fissile materials; b) were intentionally pressurized during assembly; c) contain absorber materials; d) contain non-fissile materials that may generate significant quantities of gas during irradiation; and e) contain non-fissile materials whose compatibility with the primary coolant is unknown.

2. Un-encapsulated Fuel Element Experiments: Fuel-like specimens that have passed irradiation tests performed inside capsules (under 1 above) can then be further investigated without the need for an additional barrier. This category includes fissile and control materials encased in their own cladding, but not encapsulated within another boundary. Experiment procedure stated that several experimental fuel elements had to be extensively tested in the encapsulated configuration before being accepted for testing in the un-encapsulated configuration.

3. Encapsulated Structural Material Experiments: Structural materials whose behavior is known or that do not need special treatment like fuel can be tested in ad-hoc standardized capsules. This category includes capsules not intentionally pressurized prior to irradiation and which contain materials that: 1) were known to be compatible with the primary coolant; and 2) did not generate significant quantities of gas under irradiation. These experiments also included "weeper" capsules which allow intentional ingress of primary coolant sodium into the capsule.

The operation of instrumented assemblies must be limited so that the exit coolant temperature from an open test position experiment subassembly does not differ by more than 40°C from the average exit coolant temperature for adjacent driver fuel or blanket subassemblies. This requirement is based on maximum allowable alternating stresses in the upper structure of the reactor resulting from sodium mixing effects.

The testing capability offered by instrumented assemblies is not limited simply to fuels and materials irradiation testing but can be extended to advanced instrumentation test capability. There is the opportunity for online monitoring of quantities of interest not just at the channel inlet or outlet but along the length of the assembly. In particular, open test assemblies can be used for online and direct measurements of parameters of interest (such as temperature and pressure); such assemblies could then be engineered to host traditional instrumentation and advanced instrumentation for a head-to-head comparison of performances under irradiation and harsh environmental conditions. The types of probes that could be tested include ones adopting innovative physical principles for either the measurement itself (for example, thermoacoustic sensors or fiber optic temperature sensors) or for data acquisition and transmission. In addition in-core tests could also be focused on self-powered instrumentation (through either heat or radiation) to be used under accident conditions such as those during a station blackout. Such sensors could be of vital importance to be able to perform long term plant diagnosis during beyond design basis accidents when power supply to traditional instrumentation lines may not be available for extended periods of time.

Lastly, rabbit tubes to provide for the insertion and retrieval of specimen can be located at the instrumented test assembly locations and in the closed loop locations. The rabbit tubes will be inserted through the head of the reactor vessel down to the core and grid plate structure. The rabbit tubes will be filled with inert gas (argon) to facilitate rapid insert and retrieval of irradiation specimens.

3.7 FASTER Test Reactor Safety Analysis

The safety goals in nuclear power reactor design and operation are to ensure the health and safety of the public, to protect the plant operating staff from harm, and to prevent plant damage. Traditionally, these goals have been fulfilled by a "defense-in-depth" approach that 1) minimizes risk by maximizing safety margins in design and operation, 2) reduces the likelihood of potentially harmful events by providing safety systems to deal with anticipated events, and 3) provides additional design features to mitigate the harmful consequences of low probability events.

Best estimate simulations of Unprotected Loss of Flow (ULOF), Unprotected Transient Overpower (UTOP), and Unprotected Loss of Heat Sink (ULOHS) transients were performed to determine the margins to sodium boiling and fuel melting, with an assumed fuel melting temperature of 1071°C. Additionally, low enough temperatures in the primary system must be maintained to ensure prolonged structural stability of the major components. Of all the structures, maintaining the integrity of the reactor vessel is the most important as it provides the boundary for the primary sodium heat transport system. The maximum allowable temperature for the reactor vessel and sodium pool is assumed to be 732°C, which is the Service Level D limit used in the SAFR PSID.

Results from the ULOF, ULOHS, and UTOP transient simulations are summarized in the **Table 9** and **Table 10** below. Adequate safety margins are maintained during each of the analyzed transients.

Table 9 - Margins and Peak Temperatures for Unprotected Transient Scenarios at BOC Conditions

	Sodium Boiling Margin (°C)	Peak Cladding Temperature(°C)	Peak Fuel Temperature(°C)	Peak Reactor Vessel Temperature(°C)
Nominal	399	568	712	355
ULOF	234	720	741	462
ULOHS	391	569	712	562
UTOP	292	688	889	415

Table 10 - Margins and Peak Temperatures for Unprotected Transient Scenarios at EOC Conditions

	Sodium Boiling Margin (°C)	Peak Cladding Temperature(°C)	Peak Fuel Temperature(°C)	Peak Reactor Vessel Temperature(°C)
Nominal	398	560	652	355
ULOF	268	682	694	431
ULOHS	396	561	653	496

4. Summary or Conclusions

The FASTER test reactor was designed as part of the U.S. Advanced Demonstration and Test Reactor Options (ADTR) Study in 2015/2016 [1]. The ADTR study provided an assessment of advanced reactor technology options and is intended to provide a sound comparative technical context for future decisions concerning these technologies. Point designs for a select number of concepts were commissioned. The FASTER test reactor concept meets or exceeds all of the ATDR requirements for a test reactor. As the U.S. progresses to re-establish its fast reactor testing capability, the FASTER test reactor provides an excellent starting point for the design development of a fast irradiation test reactor.

5. Acknowledgements

Argonne National Laboratory's work was supported by the U. S. Department of Energy Advanced Reactor Technology (ART) Program under Prime Contract No. DE-AC02-06CH11357 between the U.S. Department of Energy and UChicago Argonne, LLC.

6. References

[1] PETTI, D., et al., "Advanced Demonstration and Test Reactor Options Study", Argonne National Laboratory, Oak Ridge National Laboratory, Idaho National Laboratory, USA, INL/EXT-16-37867 Revision 2, July 2016.

[2] GRANDY, C., et al., "FASTER Test Reactor Preconceptual Design Report Summary", Argonne National Laboratory, ANL-ART-40, Argonne, IL (2016).

[3] HEIDET, F., et al., "Neutron Thermalization in the FASt TEst Reactor, Proceedings of FR17, Paper 144, Russia, June 2017.

[4] WOOTAN, D. W., OMBERG, R. P., GRANDY, C., FARABEE, O., "Design and Fabrication of Closed Loop Systems (CLSs) for the Fast Flux Test Facility (FFTF), Proceedings of FR17, Paper 39, Russia, June 2017.