Attachment 5

Argonne FASTER Test Reactor Preconceptual Design Report - Summary

ANL-ART-40

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Nuclear Engineering Division

ANL-ART-40

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FASTER Test Reactor

1 Introduction

The FASTER reactor plant is a sodium-cooled fast spectrum test reactor that provides high levels of fast and thermal neutron flux for scientific research and development. 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 FASTER reactor plant has the following main attributes (Table 1):

Table 1 – FASTER Reactor Plant Summary Characteristics

Reactor Power	300MWt / 120MWe/40% efficiency
Coolant	Sodium
Coolant Temperature, inlet/outlet	355°C / 510°C
Coolant Pressure (cover gas 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/peak burnup	34.3 GWd/t
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 reinforced concrete containment
In-vessel Fuel Handling Mechanism	Single Rotatable Plug with pantograph FHM (3)

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 (Table 2 and Table 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.

Due to the required 30 page limit of this condensed report, the report focuses on the main mission of irradiation testing capabilities of the FASTER reactor plant and its performance during transients. The main report will contain further details.

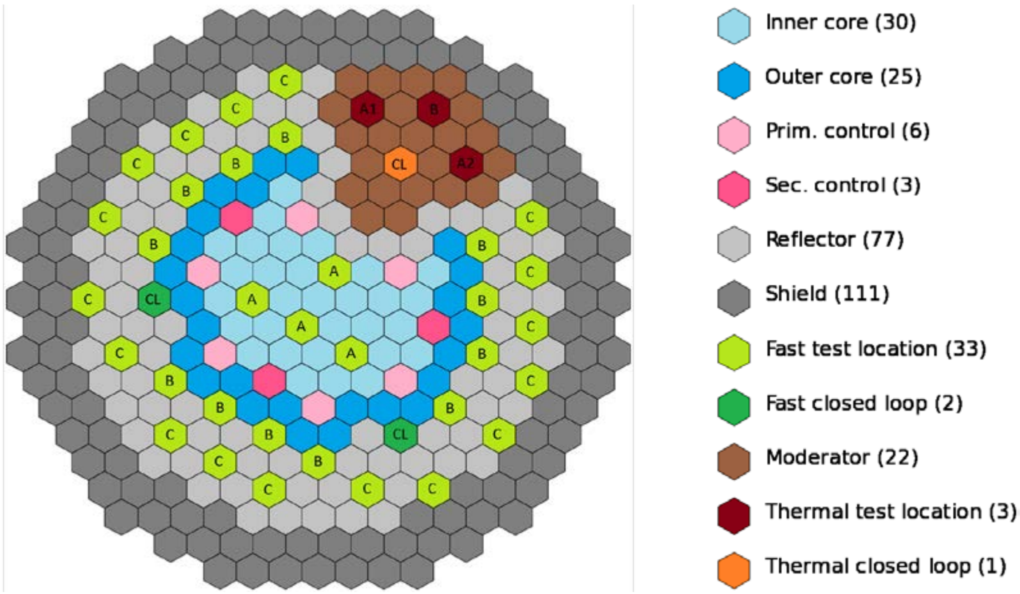

Figure 1 – FASTER Reactor Core Layout

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

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/s)
Closed loop	5.8	3.7
A1	1.9	1.2
A2	1.9	1.2
B	1.7	1.1

2 Test Reactor Objectives and Motivation for Concept 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 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 Test Reactor Point Design Description

Table 4 provides summary characteristics for the FASTER reactor plant.

Table 4 – 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
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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 assembly descriptions and dimensions are provided in Table 5.

Table 5 – Assembly Descriptions and Dimensions

Assembly type	Fuel	Reflector	Test	Control	Shield
Assembly pitch (hot)	11.870	11.870	11.870	11.870	11.870
Smear density	85	100	100	85	81
Pins/assembly	271	91	91	91	61
Pin Outside Diameter (mm)	5.405	11.072	11.072	9.512	13.462
Cladding Thickness (mm)	0.35	1.0	1.0	0.3	0.4
Pin Pitch/Diameter Ratio	1.210	1.023	1.023	1.024	1.012
Clad inner diameter (mm)	4.701	9.059	9.059	8.941	12.833
Outer Flat-to-Flat Distance (cm)	11.568	11.568	11.568	11.568	11.568
Duct Wall Thickness (mm)	3.0	3.0	3.0	3.0	3.0
Duct inside flat-to-flat distance (cm)	10.964	10.964	10.964	10.964	10.964
Duct Bundle Clearance (mm)	0.05	0.05	0.05	0.05	0.05
Inter-assembly Gap (mm)	3.0	3.0	3.0	3.0	3.0
Inlet coolant height (cm)	35.0	35.0	N/A	N/A	35.0
Lower reflector height (cm)	95.0	277.0	N/A	N/A	277.0
Fuel length (cm)	80.0	277.0	N/A	N/A	277.0
Fission gas plenum (flooded) height (cm)	12.0	277.0	N/A	N/A	277.0
Fission gas plenum (empty) height (cm)	40.0	277.0	N/A	N/A	277.0
Upper shield height (cm)	50.0	277.0	N/A	N/A	277.0
Outlet coolant height (cm)	30.0	30.0	N/A	N/A	30.0

For the purpose of the point design activity, the test locations have been modeled as entirely filled with sodium when determining the core neutronics performance characteristics, in order to maximize the neutron leakage probability and therefore the Pu wt% required in the fuel. For the safety analysis, including calculation of reactivity coefficients, the test locations have been filled will reflector assemblies in order to minimize the neutron leakage probability and not to overestimate the favorable effect resulting from sodium voiding.

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.

3.2 FASTER Reactor Performance Characteristics

The core performance characteristics at equilibrium are provided in Table 6. It should be noted that most of those characteristics will vary slightly based on the type and amount of materials loaded in the test locations, in particular those located in the active core region. In practice, a new core analysis will be required for every new core configuration, including each time a new test assembly is inserted. However, the results presented here are expected to be conservative and not to vary significantly. The reactor power level of 300 MW_{th} is the minimum that assures that the neutron flux goals can be met.

The required Pu wt% is 19.4%, which is slightly lower than the 20% limit used. The average fuel discharge burnup is about 34 GWd/t while the peak discharge burnup is less than 50 GWd/t. This is consistent with the assumptions made for the fuel smear density and plenum length, which were based on a low fuel burnup. It is important to note that the average core power density is about 560 W/cm^3 over the active fuel region. When normalized only over the fuel volume, this corresponds to an average power density of 1580 W/cm^3 and a peak value of almost 3000 W/cm^3. This is larger than what is typically observed in SFRs, and proper cooling of the core will need to be ensured. The thermal flux provided in this table is calculated for the moderated region.

Table 6 – Core Performance Characteristics

Characteristic	FASTER
Nominal power, MW_{th}	300
Required enrichment	19.41%
BOC/EOC k_{eff}	1.02672/1.00080
Average discharge burnup, GWd/t	34.3
Total HM mass, kg	2621
HM charge per cycle, kg	874
Average power density, W/cm^3	558.8
Peak power density, W/cm^3	917.0
Specific power, W/g	114.5
Average linear power, kW/m	25.2
Axial/radial leakage	11.36%/30.17%
Total flux, 10^{15} n/cm^2-s	3.74
Peak total flux, 10^{15} n/cm^2-s	6.78
Fast flux (>0.1 MeV), 10^{15} n/cm^2-s	2.70
Peak fast flux (>0.1 MeV), 10^{15} n/cm^2-s	5.19
Peak thermal flux (<0.1 eV), 10^{14} n/cm^2-s	6.19
Peak thermal flux (<0.625eV), 10^{14} n/cm^2-s	>12.0

The core power density distribution is shown in Figure 2, and since almost no energy is deposited in the moderated region it does not show up in this figure. It is also observed that the peak value is reached at the core center and not at the interface between the fast and moderated region, indicating that no thermal neutrons are returning from the moderated region.

The fast flux and thermal flux distributions are shown together in Figure 3. Different scales are used in the two regions. The thermal flux is only significant on the right-end side of the figure, corresponding to the moderated region of the core, while the fast flux is only largest in the active core region where neutrons are produced. Although not shown in Figure 3 because of the threshold value used, the fast flux remains significant in the entire reflector region. It is reduced by a factor of ~50 between its peak value at the core center and the values observed in the test assemblies located the farthest away from the core center (i.e., in the reflector).

The only region in which a significant thermal flux is observed is the moderated region. All the neutrons present in this region are neutrons leaking from the active core region and being moderated. Although this region is not fully optimized, a peak thermal flux of at least 6.0×10^{14} n/cm^2-s is achievable solely by using leaking neutrons. In this study, the energy threshold used for thermal neutrons is taken to be 0.1 eV. However, the metrics used to evaluate the test reactor performance later stated that the threshold energy for thermal neutrons should be taken as 0.625 eV. This means that the thermal fluxes claimed in this document are conservative and would be two to three times larger when using the energy threshold stated in the ATDR study.

3.3 Reactivity Coefficients and Quasi-Static Balance

The reactivity worth of the primary and secondary control rod assemblies (PCRA and SCRA) has been determined for the FASTER core in order to ensure that the core can be safely shut down from any operating condition. All nine control rods present in the core are composed of a 90 cm tall region containing a 47.58% by volume of B_4C, as well as 29.4% sodium and 23.1% HT-9. The B_4C is assumed to have an 85% smear density (i.e., the effective B_4C vol% is 39.8%). Natural boron is used in both the primary and secondary control systems. During nominal operations, the secondary control rods are fully withdrawn and only the primary control rods are used to compensate for the burnup reactivity swing.

At the beginning of the equilibrium cycle (BOEC) the PCRAs need to be inserted by about 34 cm to make up for the initial excess reactivity, and by the end of the equilibrium cycle (EOEC) PCRAs are fully withdrawn from the core. Although the requirement is that any of the control systems must be able to safely shut down the core with the highest worth control rod being stuck in the operating condition, the total worth of each system has also been evaluated when no rod is stuck. The results are shown in Table 7.

Table 7 – Reactivity Worth of the Primary and Secondary Control Systems

Worth [$]	BOEC	EOEC
Primary	21.1	22.0
Primary –rod stuck	17.3	17.9
Secondary	10.6	11.2
Secondary –rod stuck	5.7	6.1

The methodology used to determine the shutdown margins for each reactivity control system is not detailed here, but it is the methodology typically used for fast reactor analysis. In particular, it accounts for the reactivity change between the “resting state of the core” and its operating condition, for a single rod reactivity fault (ejection), considering uncertainties and a number of other factors. The shutdown margins, with the highest worth control rod being stuck, are provided in Table 8. Given that the core has the most excess reactivity at BOEC, this is when the shutdown margins are the smallest. The margins obtained are very satisfactory and could allow for reducing the B_4C volume fraction used, if needed.

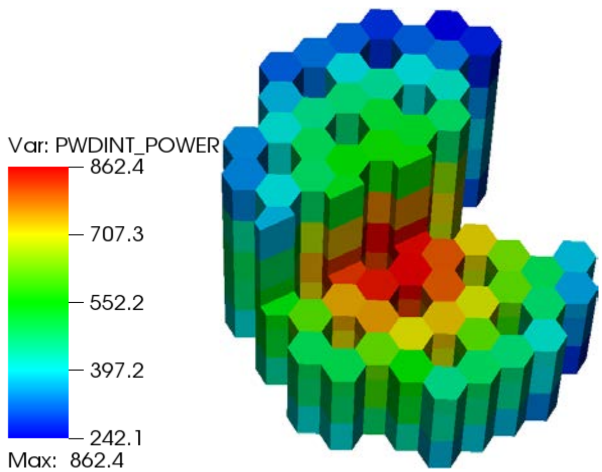

Figure 2 – Power Density Distribution (in W/cm^3)

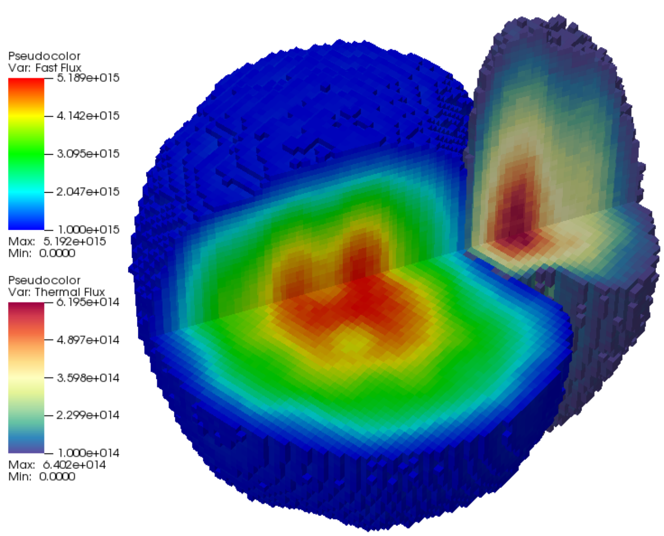

Figure 3 – Fast and thermal fluxes distribution (in n/cm^2-s)

Table 8 – Reactivity Shutdown Margins of the Primary and Secondary Control Systems

Shutdown margin [$]	BOEC	EOEC
Primary	3.21	15.14
Secondary	3.59	5.04

The reactivity coefficients required for the safety analysis have been determined at BOEC and EOEC using the PERSENT [1] code coupled with DIF3D [2]. Although not detailed here, the assumptions used for determining these coefficients are commonly used for fast reactor analysis. The various reactivity coefficients calculated are summarized in Table 9. It is important to note that the sodium void worth is negative during the entire cycle despite using Pu-based fuel. This is due to the very large neutron leakage probability and high quality Pu being used. Another noteworthy reactivity coefficient is the radial expansion which is observed to be very negative due the active fuel region having a relatively small height-to-diameter ratio and very large neutron leakage probability. Other reactivity coefficients are typical of a SFR fueled with metal fuel.

Table 9 – FASTER Reactivity Coefficients

		BOEC	EOEC
β_{eff}		0.00332	0.00333
Prompt lifetime	µs	0.349	0.373
Generation time	µs	0.348	0.373
Radial expansion	cent/°C	-0.445	-0.451
Axial expansion	cent/°C	-0.126	-0.118
Sodium void worth	$	-2.242	-2.420
Sodium density	cent/°C	-0.063	-0.067
Doppler	cent/°C	-0.109	-0.119
Sodium voided Doppler	cent/°C	-0.089	-0.097
Fuel density	cent/°C	-0.820	-0.800
Structure density	cent/°C	0.009	0.010

The quasi-static reactivity balance [3] has been performed for the FASTER core, using the reactivity coefficients previously discussed. The results are provided in Table 10. The single rod reactivity fault, $\Delta\rho$, is about 1.13$ at BOEC and nearly 0.0$ at EOEC. Although the results indicate that all three required conditions are met and that the core is expected to be inherently safe, it might still be necessary to use control rod stops in order to reduce the single rod reactivity fault and limit the maximum temperatures reached during unprotected transients.

Table 10 – Quasi-static Reactivity Balance Coefficients and Conditions

Coefficients		
A - Power/flow reactivity decrement (cents)	-33.9	-34.2
B - Power/flow coefficient (cents)	-141.4	-106.4
C - Inlet temperature coefficient (cents/°C)	-1.4	-0.9
Quasi-static reactivity conditions		
A/B <= 1	0.24	0.32
1 <= CDT_c/B <= 2	1.51	1.34
$\Delta\rho/\|B\|$ <= 1	0.80	0.03

3.4 Orificing Strategy and Temperature Distribution

In order to achieve the desired flux levels, the FASTER core has been designed with a very high power density and therefore adequate coolability of the core needs to be ensured. The multi-assembly steady state thermal hydraulic code SE2-ANL [4] is used to determine orifice groups and calculate the required coolant flow rates and temperature distributions. The inlet coolant temperature is 355°C and the bulk outlet coolant temperature is 510°C, corresponding to an average coolant temperature rise of 155°C. These temperatures are typical for SFRs with metallic fuel.

In order to limit the peak coolant and peak mid-wall cladding temperatures to acceptable values it was found that using four orifice groups for the fuel assemblies is sufficient. The temperature distribution has also been determined when using a larger number of groups, but this resulted in a reduction of the peak temperatures of only a few degrees. The proposed orifice groups are represented in Figure 4.

Seven orifice groups are used for the non-fuel assemblies, but the flow rates in those assemblies are significantly smaller than in the fuel assemblies, meaning it is possible to use more or fewer groups without significantly affecting the bulk outlet coolant temperature. Additionally, test assemblies and closed loop locations are currently modeled with reflector assemblies and will likely require specific orifice sizes depending on the desired test conditions.

The coolant flow rate, velocity, number of assemblies, and other characteristics are provided in Table 11 for each orifice group shown in Figure 4. The temperatures are provided in °C. The coolant velocity required in the orifice group containing the highest power assemblies is about 10 m/s. This is only the coolant velocity required inside the active fuel region, and is different from the velocity of the coolant exiting the core and going to the heat exchanger. The peak 2σ mid-wall temperature for the non-fuel assemblies is larger than for the fuel assemblies, but this is not a concern since no fuel-cladding interaction will occur.

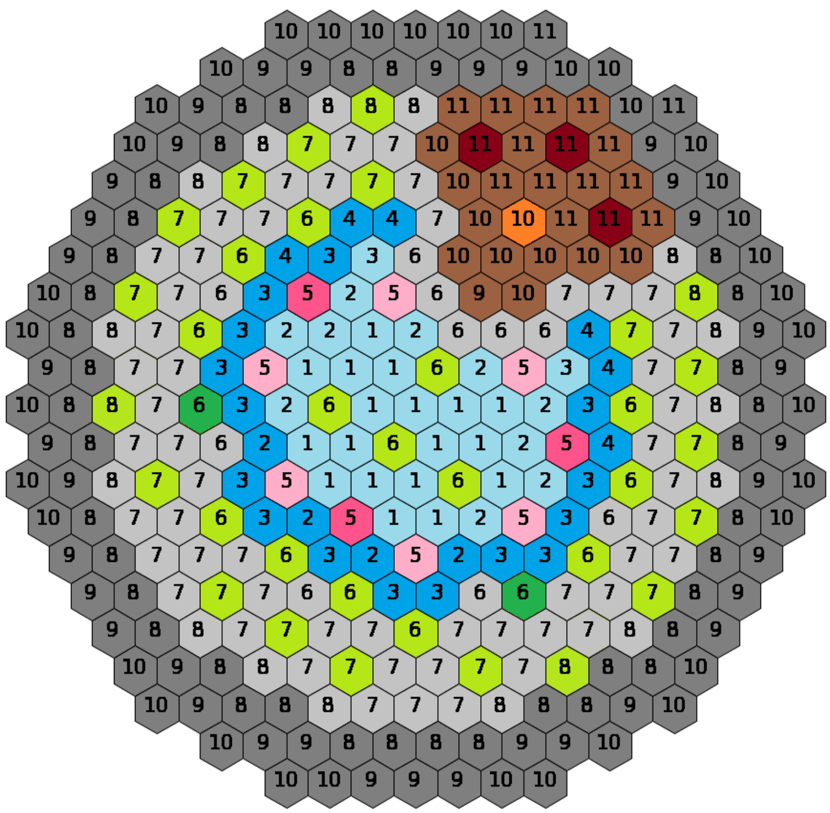

Figure 4 – Orifice Groups

Table 11 – Coolant Flow Characteristics for each Orifice Group

	Group	Assembly #	Flow rate /assembly, kg/s	Total flow rate, kg/s	Velocity, m/s	Average outlet coolant Temp., °C	Peak outlet coolant Temp., °C	Peak 2σ mid-wall Temp., °C
Fuel	1	18	34.53	621.5	10.3	510	549	587
	2	14	28.07	392.9	8.4	510	548	587
	3	17	21.43	364.3	6.4	510	549	587
	4	6	16.77	100.6	5.0	513	550	587
Non-fuel	5	9	0.55	4.95	<1	506	744	n.a
	6	26	0.55	14.3	<1	514	779	1039
	7	67	0.22	14.9	<1	499	667	909
	8	56	0.068	3.82	<1	495	589	786
	9	37	0.018	0.68	<1	480	553	739
	10	46	0.006	0.28	<1	473	525	620
	11	17	0.001	0.02	<1	489	533	636

3.5 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 5). The primary pumps are mechanical centrifugal pumps with the characteristics shown in Table 12.

Table 12 – Primary Pump Design Characteristics

Flow rate, m^3/s	758.3
Pump head, kPa	704
Power, kW	950
Efficiency, %	90
Pump length, m	9.18
Pump diameter (case), m	0.889

The IHXs are conventional 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 design characteristics of the IHX are provided in Table 13.

Table 13 – IHX Design Parameters

Heat transfer capacity, MW	75
Thermal Design Margin, (for average case)	±%25
Heat transfer area, m^2	231
Primary sodium temperature, inlet, °C/outlet, °C	355 / 510
Primary sodium mass flowrate, kg/s	379.15
Secondary sodium temperature, inlet, °C/outlet, °C	279 / 499
Secondary side sodium mass flowrate, kg/s	265
Tube outer diameter, cm	1.59
Tube wall thickness, mm	0.889
Tube pitch, cm	2.5
Effective tube length, m	3.85
Number of tubes	1,200
Tube sheet thickness, mm	100
Downcomer piping-OD, cm	32.385
Downcomer piping-thickness, mm	9.525

Downcomer piping-length, m	TBD
Outlet piping –OD, cm	45.72
Outlet piping –thickness, mm	9.525
Outlet piping –length, m	TBD
Shell (primary) side pressure drop, kPa	15.96
Tube (secondary) side pressure drop, kPa	30.76
Shell height, m	4.96
Shell outside diameter, m, main body/maximum	1.038/1.145
Shell thickness, mm	19
Tube material	9 Cr-1Mo

The intermediate heat transport system (IHTS) (Figure 6) 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 components are connected via the IHTS piping.

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 traps located in the primary and intermediate coolant systems. These cold traps will remove oxygen, hydrogen, and other impurities via conventional crystallization techniques. 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.

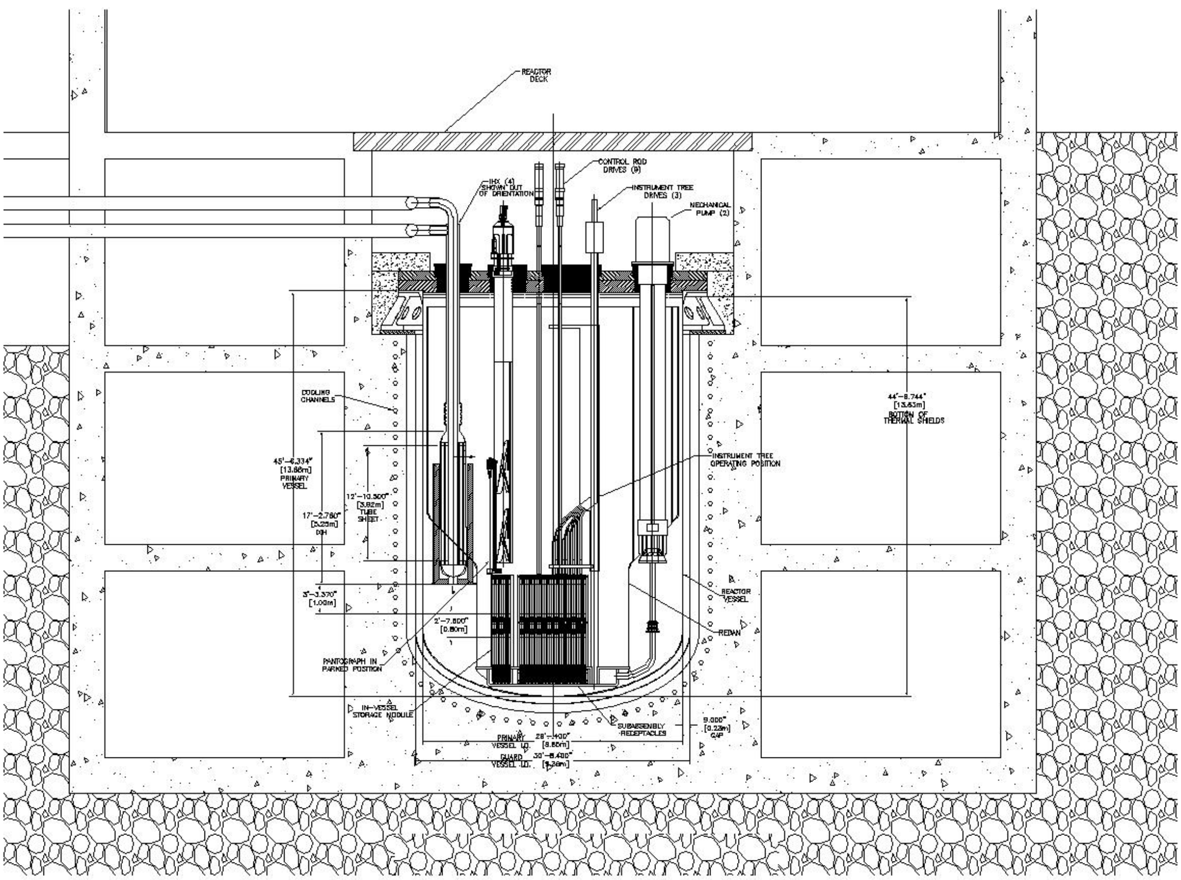

Figure 5 – Elevation View of PHTS

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. The containment and those parts of the nuclear island (NI) containing sodium piping and components incorporate design features to mitigate the effects of postulated sodium leakages and sodium fires such that temperature and pressure loadings following sodium leakage remain small or negligible. These features include sodium leak detectors, detecting a sodium leak when it is small before it can grow, shutting down the pump and draining the sodium from a failed sodium loop into the loop sodium dump tank to limit the amount of sodium released upon detection of a sodium leak or sodium burning, automatic isolation of ventilation lines serving a compartment upon detection of aerosols in the outlet duct, compartmentalization to reduce the volume of an air-filled partially-sealed compartment housing sodium piping or components thereby reducing the amount of available oxygen such that a sodium fire will rapidly consume the available oxygen and burn itself out, use of sodium catch pan fire suppression decks to significantly reduce the sodium burning rate from a sodium pool and protect the underlying concrete, confining sodium released from a failed pipe inside the gap between the pipe and surrounding thermal insulation and draining it through drain pipes to eliminate or significantly reduce the potential for formation of sodium jets or sprays, and the use of steel liners on compartment inner surfaces to further protect concrete. These mitigation approaches were previously developed and tested as part of the CRBR, PRISM, and SAFR design activities.

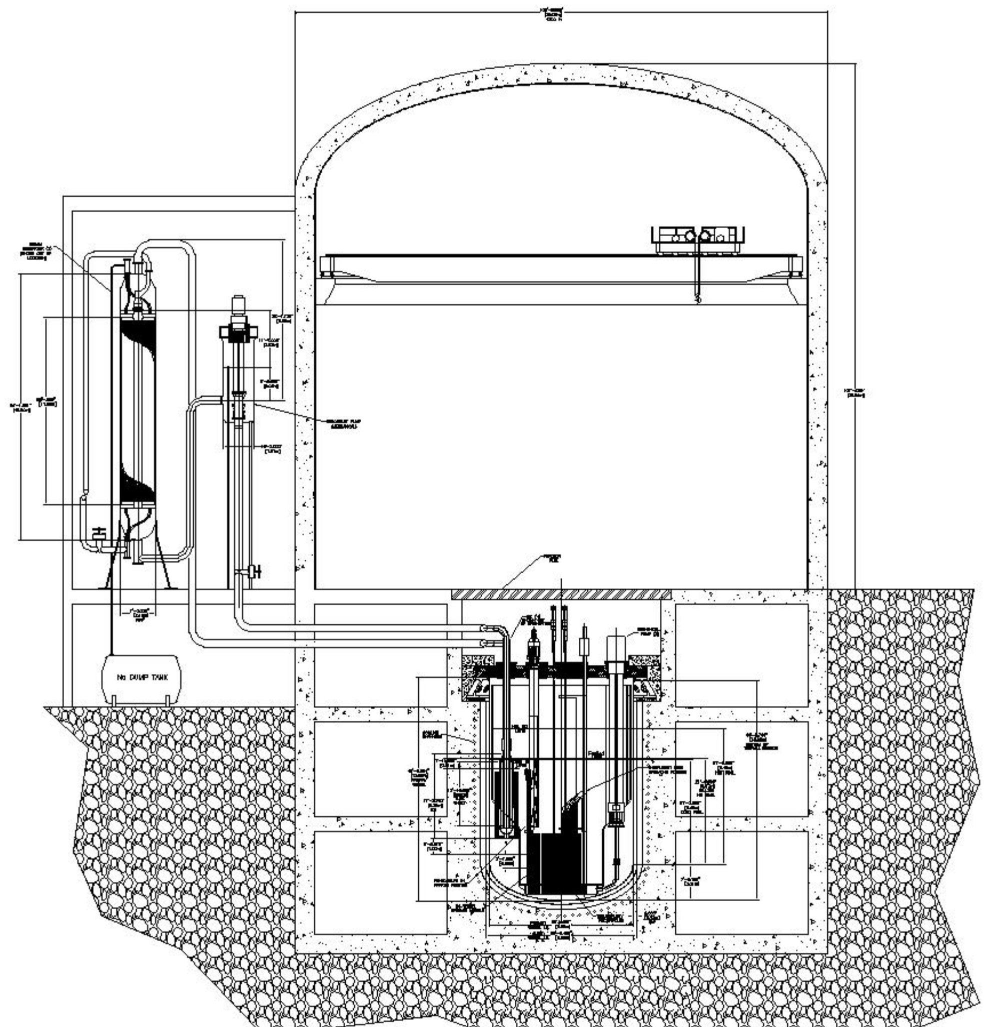

Figure 6 – FASTER NSSS – Elevation View

Emergency decay heat removal is provided through three 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.

Table 14 – DRACS HX - each

Heat transfer capacity, KW	750
Heat transfer area, m^2	7.64
Primary sodium temperature, inlet, C°/outlet, C°	510/355
Primary sodium mass flowrate, kg/s	3.793
Secondary NaK temperature, inlet, C°/outlet, C°	328/484
Secondary NaK mass flowrate, kg/s	5.47
Tube outer diameter, cm	2.22
Tube wall thickness, mm	0.9
Tube pitch, cm	3.79
Effective tube length, m	2.489
Number of tubes	44
Shell OD, cm	32.26
Shell wall thickness, mm	9.525
Material	9Cr-1Mo

The balance of plant consists of a conventional superheated steam cycle attached to the (2) once-through sodium heated steam generators. Conditions are calculated with the GateCycle software [5] (Figure 7).

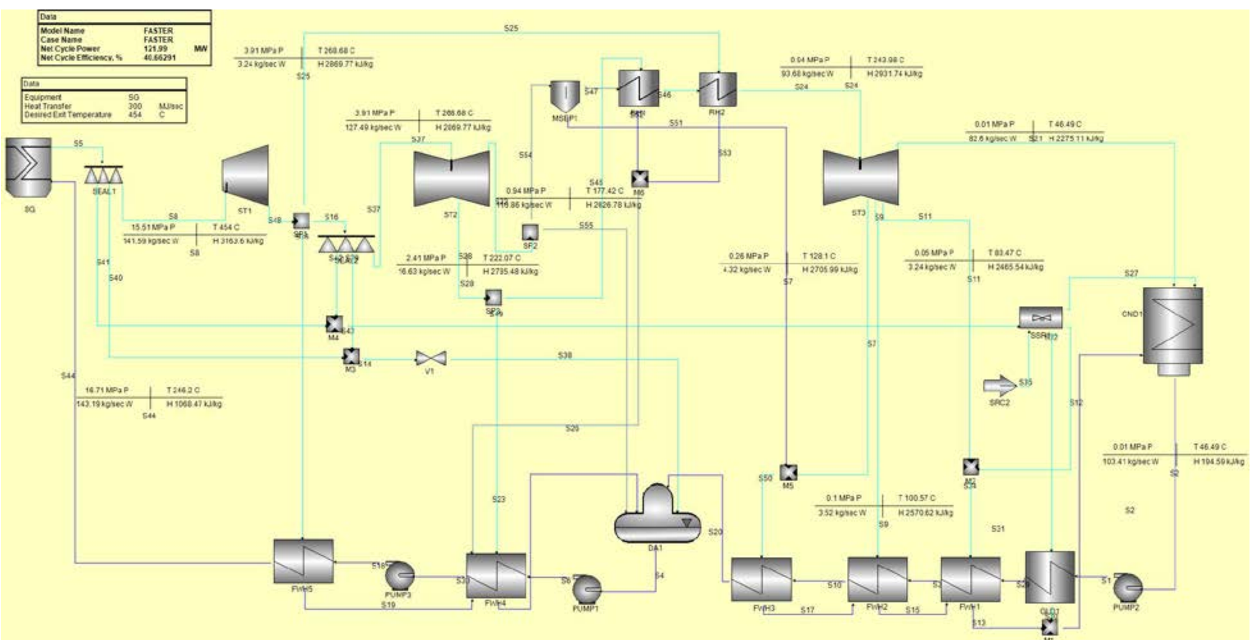

Figure 7 – FASTER Thermodynamic Cycle and Balance of Plant

3.6 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.

3.7 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 8 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 15. 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.

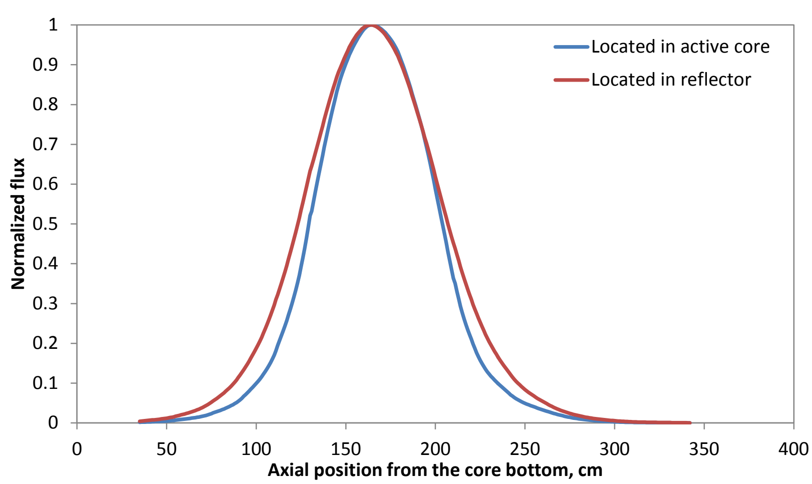

Figure 8 – Normalized axial fast flux distribution in test locations

Table 15 – 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 16 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 9. The rough aspect of the curve is due to the uncertainties of the calculations performed with MCNP.

Table 16 – 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

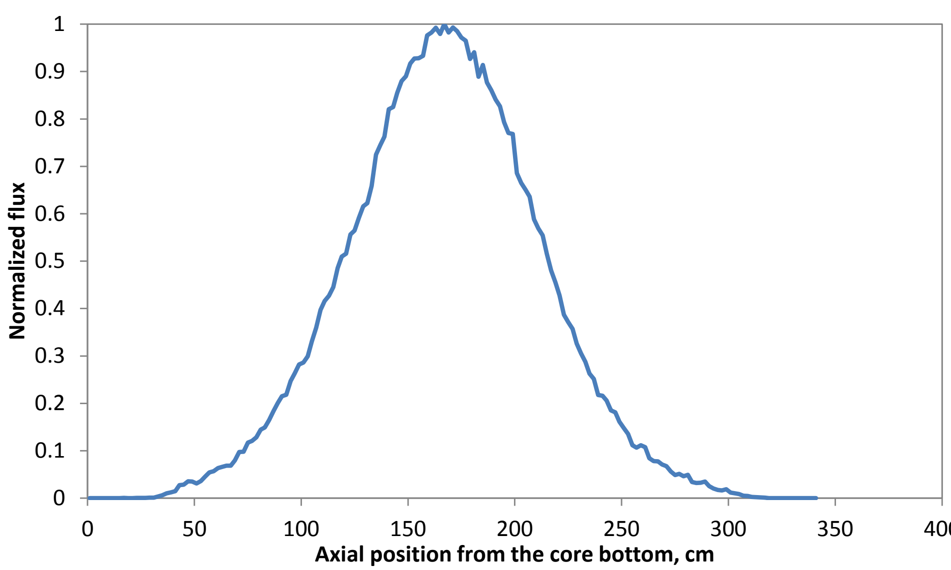

Figure 9 – Normalized axial thermal flux distribution in the test locations

3.8 Closed Loop Systems

The three closed loop systems (CLS) are an important capability of FASTER and part of the study 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 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 fuel, 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 [6], [7] 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. Pressurized water CLs are also an integral part of the Advanced Test Reactor (ATR) design and are utilized for irradiation and testing [8].

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 [6], [7]. Heat removal rate and coolant flowrate requirements for different coolants for different example reactor designs are shown in Table 17 for test sections simulating a small portion of each reactor core inside of the flow tube. 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 17 - 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, $2LiF\text{-}BeF_2$	Liquid Salt, FLiBe, $2LiF\text{-}BeF_2$	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 Velocity, m/s	5.52	5.52	2.0	1.94	2.0	20.2	4.85	15.5

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 17. The required pressure tube dimensions for a design temperature of 649 °C (1200 F) are shown in Table 18. 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 insure 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 17.

Table 18 – 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 (1300 F), ample space is still available with the low pressure and pressurized helium coolants (Table 19). There still remains space when the design temperature is further increased to 760 °C (1400 F) as shown in Table 20.

The test sections in the FFTF closed loop in-reactor assemblies were designed for a sodium outlet temperature of 760 °C (1400 F) while the double-walled pressure tube and other closed loop hardware was designed for 649 °C (1200 F). 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 19 – Required Pressure Tube Dimensions for 704 °C (1300 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)	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 20 - Required Pressure Tube Dimensions for 760 °C (1400 F) 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 21. 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 21 - 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 to Preclude Leakage	Double-Wall Flow Tube with Monitored Gap to Preclude Leakage	Double-Wall Flow Tube with Monitored Gap to Preclude Leakage	Double-Wall Flow Tube with Monitored Gap to Preclude Leakage and for Thermal Insulation	Double-Wall Flow Tube with Monitored Gap 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 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 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 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.

Details of the closed loop modules are provided in the Table 21 and in the main report.

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) [7]. 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 incorporated a tungsten funnel to collect fuel, a TZM molybdenum alloy cup with six inwardly projecting fins to contain molten fuel, and a tungsten washer followed by a borated graphite shield block beneath the TZM cup. For each individual CLS primary coolant type and reactor type core simulation, an assessment will to be carried out of the accident phenomena and an approach to contain the test section materials as well as mitigate the release of radionuclides from the in-reactor assembly will be engineered. This type of analysis could not be carried out during the time frame of this study.

3.9 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 main capabilities of the instrumented assemblies envisioned for FASTER are hereunder described. Considering, as a minimum requirement, the same capabilities offered by the FFTF design, instrumented assemblies loaded with fuel elements allow for monitoring of parameters such as sodium temperature within the fuel bundle, cladding temperature, duct temperature, fuel centerline temperature, and fission gas pressure within the pin. Transducers can also be used to measure differential pressure within the fuel bundle. In FFTF, up to fifty-one leads for pressure, temperature, and electrical connections were used. Such a number represents a lower bound for a FASTER instrumented assembly. In addition, in FFTF, open test assemblies could be removed from the core after first severing the instrumentation leads, and then reinserted into the core for a post-irradiation open test after examination in a hot cell.

Additionally, instrumented assemblies loaded with encapsulated structural experiments allow for the evaluation of irradiation behavior of structural materials. These assemblies will provide detailed temperature control and measurement during irradiation as well as density and dimensional measurements on test specimens made ex-core during reactor shutdown. In FFTF, the open test assemblies loaded with material specimens could contain up to forty-eight canisters, thirty of which with independent temperature control, and a total of 2500 cm^3 of in-core irradiation volume per assembly. In addition, up to eighty-two leads for pressure, temperature and electrical connections were available. Again, this is to be considered a lower bound for FASTER instrumented assemblies. Instrumented assemblies loaded with structural material specimens are also designed to be removed from the core at the end of an operating cycle, the test specimens then examined, and the experiment then reinserted into the core for the start of the next cycle.

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 insert 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.

3.9.1 Identification of prototypic or scalable aspects of fuel performance testing.

The instrumented assembly will provide a one-of-a-kind capability for validation of advanced simulation tools. Fuel pin wire wraps induce complex three-dimensional mixing that historically has not been accurately represented in design calculations. Designers understood the limitations of their models and correspondingly introduced uncertainty factors to account for lack of fidelity. In turn this leads to a derating of the core. Ongoing research and development of multi-physics simulations for improving fast reactor economics is specifically targeting improved prediction of subassembly temperatures by better modeling the effects of wire-wrapped induced mixing. The usefulness of these models for future design work and recovery of thermal margin is critically dependent on validating these models against prototypic experiment data.

The instrumented assembly offers a unique opportunity to generate prototypic experiment data for validation of these advanced computational design tools. With the multi-lead capability provided by the instrumented assembly, it is possible to provide a spatially rich set of temperature measurements within the fuel assembly. The needed measurements can be acquired using two different measurement capabilities. The first uses thermocouples embedded in wire wraps to sense local coolant temperature. The second approach replaces the wire wrap with a stainless steel capillary of the same outside diameter through which is threaded an optical fiber. This so-called *Bragg fiber* is capable of generating temperature data to the sub-centimeter level along its length yielding here-to-fore unrealized spatial resolution of coolant temperature within a fueled assembly. The development of these fibers for higher temperature applications is ongoing. It is expected that soon there will be a fiber capable of operating at SFR temperatures for the time needed at full power to acquire the needed steady-state temperature data.

3.10 Plant Security and Safeguards

The plant is separated into nuclear and nonnuclear areas, referred to as the nuclear island (NI) protected area and the balance-of-plant (BOP) controlled area, respectively. The NI contains all safety-grade components and systems, while all non-safety-grade components and systems are located in the BOP area or the owner-controlled area. This results in a zoned approach to security with fewer facilities and personnel within a small NI-protected area.

In addition, the plant has inherent and unique operating characteristics and safety features that reduce the scope of the security system. These include:

- Physical separation of the NI from the BOP with no safety related components and systems in the intermediate heat transport system (IHTS) or BOP controlled area. This is a feature of the FASTER SFR design that does not need the IHTS or BOP for emergency heat removal
- Inherent, passive reactivity feedbacks
- Multiple passive decay heat removal systems where only 2 out of 3 are required
- Long grace period before corrective action is needed.

The FASTER security system is designed to protect special nuclear material (SNM) and vital equipment in the most efficient way so that cost, operational impact, and security force size are all minimized. The design integrates the often competing objectives of assuring nuclear plant safety and physical security while not adversely affecting the safety of plant personnel. Thus, the design reflects the interface between the safety and security requirements.

The security system is designed to:

- Allow authorized personnel and material access to the facility
- Keep out all unauthorized personnel and material
- Detect and verify all unauthorized activities
- Delay unauthorized activities
- Deter potential adversary actions
- Prevent theft of special nuclear material
- Prevent sabotage of vital equipment.

The physical security system for the NI-protected area has the following primary components:

- Multiple perimeter barriers and isolation zones
- Redundant intrusion detection systems
- Surveillance and assessment equipment
- Hardened guardhouse facilities
- A computer-based card-key access control system
- Positive personnel identification systems
- Central and secondary alarm stations
- Redundant and uninterruptible power systems
- Hardened, access-controlled boundaries for vital areas and equipment
- On-site guard force backed by off-site response forces.

These systems provide detection, assessment and delay of, and response against the threats of radiological sabotage and theft. Detection is accomplished using sensors located at the NI-protected area boundary. Assessment of sensor alarms is remotely accomplished using a closed-circuit television (CCTV) system. A delay to intrusion is provided by the perimeter barriers to permit CCTV assessment. A more substantial delay to allow adequate time for an effective response is achieved at the exterior building envelope. Delay times and specific features of the barrier system depend on tradeoffs involving the response force, such as the number of guards and the time it takes for guards to engage the intruders.

The objectives of a safeguards system are to provide:

- A physical protection system that prevents theft of SNM and that prevents sabotage of vital equipment whose failure, destruction, or misuse could result in a radiological hazard to the public
- A materials control and accounting system to maintain records on the amount and location of SNM in the plant and to monitor the SNM to verify that it has not been stolen or diverted.

The safeguards system for FASTER is relatively straightforward, The fuel will come into the facility via transfer casks and will exit the facility using the same transfer casks. Spent fuel will be stored on site as part of the FASTER reactor plant operations. Fuel specimens for examination will be sent to hot cell facilities located on site, but are not part of this study.

3.11 Decommissioning and Waste Generation Aspects

At the end of the FASTER reactor plant life, the fuel will be off-loaded from the core and the dummy 316SS core assemblies will be installed in the core. The cold traps and nuclide traps will be cut/removed from the system and handled separately. The primary and secondary sodium will be removed and processed on site or shipped off site for treatment and disposal. The residual sodium will be deactivated using moist CO_2 and moist N_2 and ultimately flushed with water. Once the sodium in a system has been deactivated, normal decommissioning techniques will be used for decommissioning the plant.

4 FASTER Test Reactor Safety

4.1 FASTER Reactor Safety Basis

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 an 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. This approach is usually identified as “defense in depth.”

The basic principle of “defense in depth” is to provide multiple levels of protection against release of radioactive material. One part of defense-in-depth is physical barriers, like the multiple barriers to release of radioactivity provided by the fuel cladding, the primary coolant system boundary, and the reactor containment building. Generally, active or passive safety systems are provided to protect the physical barriers. These include the reactor shutdown systems and the reactor cooling systems. Inherent characteristics of the design, such as negative reactivity feedback and long flow coast-down, may provide an additional level of protection. Emergency planning provides an additional layer of defense-in-depth, should the other barriers be threatened. However, in all instances, the “defense in depth” strategy depends on the independence of the protective measures, so that no single event can breach more than one protective level.

The FASTER safety design approach implements the “defense in depth” strategy by adopting the traditional three levels of safety. In addition, the FASTER design features have been selected to provide significant safety margin enhancements by inherent passive safety responses to upset conditions and equipment failures.

At the first level, FASTER is designed to operate with a high level of reliability, so that accident initiators are prevented from occurring. The first level of safety is ensured in part by selection of fuel, cladding, coolant, and structural materials that are stable and compatible, and provide large margins between normal operating conditions and limiting failure conditions. An arrangement of components was adopted that allows monitoring, inspection, and testing for performance changes or degradation. Finally, the FASTER design provides for repair and replacement of components as necessary to ensure that safety margins are not degraded.

The selection of liquid sodium coolant and metallic fuel with a pool-type primary system arrangement provides a highly reliable reactor system with large operational safety margins. The coolant thermophysical properties provide superior heat removal and transport characteristics at low operating pressure with a large temperature margin to boiling. The metallic fuel operates at a relatively low temperature, below the coolant boiling point, due to its high thermal conductivity. The pool-type primary system confines all significantly radioactive materials within a single vessel, allows for easy removal and replacement of components as well as shutdown heat removal by natural circulation.

At the second level of safety, FASTER is designed to provide protection in the event of equipment failure or operating error. This level of protection is provided by engineered safety systems for reactor shutdown, reactor heat removal, and emergency power. Each of these safety-grade backup systems functions in the event of failure in the corresponding operating system, and are subjected to continuous monitoring and periodic testing and inspection.

The FASTER design provides an independently powered and instrumented secondary reactor shutdown system that operates automatically to reduce reactor power rapidly in the event that the primary shutdown system fails. For shutdown cooling, the FASTER design includes a safety-grade emergency heat removal system, independent from the normal heat removal system and capable of removing residual decay heat by natural circulation. In addition to the normal offsite power supply, FASTER is equipped with a second independent offsite power connection. The two offsite power connections are supplemented by a safety-grade onsite emergency power supply.

The third level of safety provides additional protection of the public health and safety in an extremely unlikely event that is not expected to occur in the life of the plant, or which was not foreseen at the time the plant was designed and constructed. In the FASTER design, the Level 3 protections for cooling assurance and containment of radioactivity are provided by the reactor guard vessel and the reactor containment building. The reactor guard vessel is designed to hold primary coolant in the event of a leak in the primary coolant system. The reactor guard vessel ensures that the reactor core remains covered with sodium and cooled by the emergency heat removal system, even if the primary reactor vessel fails. If primary coolant leaks and oxidizes in the reactor building air atmosphere, or if failures of the cladding and the primary system barriers lead to release of gaseous fission products, the reactor containment building provides a final low-leakage barrier to release of radioactivity to the environment.

The three levels of safety together are the safety design basis for FASTER. For the purposes of subsequent safety design development, qualification, and documentation, it is customary during the conceptual design phase to identify general design criteria (GDC) that collectively serve as the basis for safety assessment of the design. GDCs for advanced reactors, and SFRs in particular, are currently being developed in a joint DOE and NRC initiative. The FASTER design satisfies all of those currently existing draft SFR GDCs.

The proposed FASTER design is capable of accommodating various beyond-design-basis accident initiators without producing high temperatures and conditions that might lead to a severe accident, such as coolant boiling, cladding failures, or fuel melting. The inherent neutronic, hydraulic, and thermal performance characteristics of the FASTER design provide self-protection in beyond-design-basis sequences to limit accident consequences without activation of engineered systems or operator actions. This characteristic has been termed 'inherent passive safety.'

The efficacy of such passive safety was demonstrated through two landmark tests conducted on the Experimental Breeder Reactor-II (EBR-II), namely SHRT-45R, a loss-of-flow without scram test, and BOP-302R, a loss-of-heat-sink without scram test. With the automated safety systems disabled, the two most demanding accident initiating events were deliberately induced with the reactor at full power, first one then the other. Each time the reactor simply coasted to a safe low power state without any damage at all to the fuel or any reactor component. These tests proved conclusively that passive safety design is achievable for metallic-fueled fast reactors with sodium cooling.

Within the overall safety framework for FASTER, passive safety serves to provide additional margins for public protection in the event of very low probability events whose frequency of occurrence is lower than the normal threshold for deterministic assessment. The FASTER passive safety performance characteristic ensures that no abnormal radioactivity releases will occur in the event of beyond-design-basis accidents, and that all of the multiple defense-in-depth barriers (fuel cladding, reactor vessel, containment building) for public protection will remain intact, just as for design basis accidents. The passive safety performance of FASTER eliminates the potential for severe accident consequences in very low frequency, beyond-design-basis sequences. Consequently, for FASTER, beyond-design-basis accidents need to be considered only in the context of probabilistic risk assessments.

Security must now be considered as an integral part of the design. The inherent and passive safety features of FASTER offer a high level of protection against malevolent events, as well as against accidents. Since the inherent and passive features do not rely on operator action, external power or functioning of active components, they remove these potential vulnerabilities. In addition, the location of the reactor vessel, the core, and the primary heat transport system below grade within a strong containment structure provides protection against external threats.

4.2 Source Term

Development of a source term and assessment of offsite consequences that can result from radionuclide release are typically the final components of an integrated safety analysis for a reactor. Historically, source term analyses for LWRs have utilized bounding, deterministic assumptions and are based on LWR fuel forms and accident scenarios, as indicated by the formal guidance from the NRC on acceptable source term methodologies in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," [9] and Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" [10]. However, due to deviations in the accident bases and properties of the coolant (i.e., compatibility of sodium with metallic fuel, and high solubility of radionuclides within sodium), LWR –derived source terms are considered inappropriate for application in an SFR safety analysis. Additionally, improvements in the knowledge state and computational capabilities have led to renewed interest in mechanistic treatment of source term phenomena.

To that end, ongoing work at Argonne on development of a mechanistic source term (MST) for SFRs [11] has resulted in identification of the significant sources and important transport and retention mechanisms for releases from postulated highly unlikely core damage and fuel handling accidents. The majority of transport and retention mechanisms are well understood, however, uncertainty remains in the understanding of radionuclide formation, transport, and release in-pin as a function of burnup, particularly at high burnup. Under the current DOE sponsored work, it is expected that a MST for SFRs will be available during the conceptual design phase of the FASTER reactor or before.

An SFR MST can be defined as follows [11]:

An SFR MST is the result of an analysis of radionuclide release, in terms of quantities, timing, and other characteristics, resulting from the specific event sequences being evaluated. It is developed using best estimate phenomenological models of the transport of radionuclides from the source through all holdup volumes and barriers, taking into account mitigation features, and finally, into the environs.

For a pool-type SFR, the five key barriers to release include the fuel matrix, cladding, primary sodium (core uncover accidents are highly unlikely in pool-type configurations), primary circuit boundary, and containment. During normal operation, the majority of fission gases will migrate to the fission gas plenum, with a portion of these gases and vapors transporting through bond sodium where there is the opportunity for fission products to dissolve in the bond sodium. If cladding breach occurs, radionuclide gases and vapors in the fission gas plenum and some fraction of the bond sodium will be ejected from the pin. However, many radionuclides will be retained within the fuel matrix, as uranium is an excellent solvent. Once released to the primary coolant, it is expected that only gases and vapors with high vapor pressure and low sodium solubility (e.g., noble gases) will transport directly to the cover gas. The remaining radionuclides will condense and dissolve (gas/vapor) or dissolve and deposit (particulate) in sodium and on structure. Gases and vapors transported in bubbles will be directly released to the cover gas, but dissolved vapors, gases, and particulates must vaporize to enter the cover gas where they may condense and mechanically deposit following vaporization. Airborne radionuclides can then transport through leakage through the seals in the reactor head and into containment. Within containment, phenomena similar to those experienced in the cover gas are expected to occur, although the presence of oxygen and water vapor may induce additional phenomena. Cooler containment temperatures, the likely availability of sodium vapor (which escaped the primary boundary via the reactor head leak pathway), and the small containment design basis leak rate are all expected to enable high retention of radionuclides in containment due to the various condensation, dissolution, and holdup mechanisms.

In the case of an SFR, it is likely that the highest releases will result from spent fuel movement and handling accidents, where the two most significant barriers, primary sodium coolant and cover gas, are not available. Also, in contrast to LWR bypass scenarios, an SFR can experience primary circuit boundary bypass via the cover gas cleanup or primary sodium purification systems. Potential bypass scenarios are addressed by housing the cold traps and sinks associated with these systems in sealed rooms with relatively robust containment capabilities.

4.3 Emergency Decay Heat Removal Capability

The decay heat removal system (DHRS) heat rejection rate capacity was determined by analyzing a series of unprotected beyond-design-basis accident scenarios with various heat rejection capacities. It was assumed that heat rejection through the BOP is unavailable and the control rods fail to scram. Unprotected loss of flow and unprotected loss of heat sinks scenarios, two highly unlikely (less than 10^{-6} per reactor-year) beyond-design-basis accidents, were simulated at beginning and end of equilibrium cycle conditions. As temperatures in the core increase to elevated but safe levels, reactivity feedbacks shut down the fission process, leaving only decay heat. In these scenarios, the DHRS is the sole mechanism for rejecting heat generated in the core.

The necessary DHRS heat rejection rate capacity was determined as the amount of decay heat rejection rate necessary to maintain acceptable safety margins during unprotected transients with one of the three DHRS units assumed to be unavailable, either due to maintenance issues or a failure of the passively opened air dampers to activate. Heat rejection rate capacities between 0.25% and 0.50% of the nominal core power per DHRS unit were examined with 0.25% per unit determined to provide sufficient decay heat rejection to maintain acceptable safety margins. With an assumed failure of one unit, the nominal total decay heat rejection rate is 0.50% of nominal power. Because the safety margins with 0.25% per unit are large enough, the additional cost to increase the size of the DHRS units to provide 0.50% per unit, for example, is not justified.

4.4 FASTER Test Reactor Safety Performance

The normal process of performing safety assessments considers a spectrum of DBAs as tests of the various safety systems. These DBAs generally assume single failures. Accidents within the design basis must be accommodated by the design and shown to present risks to the public that are within regulatory standards. Beyond the design basis, there exists a class of accidents of such low probability that they have been termed "hypothetical." These events involve multiple failures of safety grade systems, and usually are considered to have a frequency of less than 10^{-6} per reactor-year. Because of the potentially severe consequences of accidents in this class, they have received significant regulatory scrutiny in prior sodium-cooled fast reactor licensing reviews for the purpose of characterizing thermal and structural safety margins beyond the design basis. These accidents are currently referred to as design extension conditions (DECs, between 10^{-6} and 10^{-8} per reactor-year) or severe accidents (SAs, less than 10^{-8} per reactor-year) depending upon the specific frequency.

Three DECs calculated here, each involving failure of both reactor scram systems, have received attention in past licensing safety assessments. In the unprotected loss-of-flow (ULOF) sequence, it is assumed that power is lost to all primary and secondary coolant pumps and the reactor scram systems fail to activate. In the unprotected transient overpower (UTOP) sequence, it is assumed that one or more inserted control rods are withdrawn, and the reactor scram systems fail to operate. In the unprotected loss-of-heat-sink (LOHS) accident, it is assumed that heat removal through the power conversion system is lost, and the reactor scram systems do not activate. Taken collectively, these three accident initiators encompass all the ways that an operating reactor can be perturbed, i.e. by a change in coolant flow, by a change in reactivity, or by a change in coolant inlet temperature. A preliminary safety analysis was performed for FASTER using the systems analysis code SAS4A/SASSYS-1 to assess the reactor's safety performance during the transients. [12] A series of ULOF, ULOHS, and UTOP transients were simulated at both beginning of cycle (BOC) and end of cycle (EOC) conditions, except for the UTOP, which was only simulated at BOC because the control rods are already withdrawn at end of cycle.

Maintaining coolable geometry within the core is the primary consideration when evaluating events with such low frequencies. For example, fuel melting can be tolerated at the center of the fuel pin as long as molten fuel is not breaching the cladding and entering the coolant channel. However, the favorable features of FASTER, and SFRs in general (i.e., strong inherent reactivity feedbacks, the excellent heat transfer capabilities of sodium, and the large heat sink of multiple sodium regions inside of the pool-type reactor vessel), lead to such large safety margins that the transient scenarios can be evaluated under higher scrutiny. For this analysis, the results of a transient are considered acceptable when reasonably large margins to sodium boiling and fuel melting are maintained.

Best estimate simulations of ULOF, UTOP, and 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 circuit. 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. [13]

Results from the ULOF, ULOHS, and UTOP transient simulations are summarized in the tables below. Adequate safety margins are maintained during each of the analyzed transients. In the UTOP scenario, a single control rod is assumed to be unintentionally withdrawn until it reaches its rod stop, 6 cm above the critical insertion depth, limiting the reactivity insertion to 0.5 $. The UTOP scenario attains the highest fuel temperatures of all of the transients, with a peak fuel temperature of 889°C; a fuel melting margin of 182°C is maintained.

Table 22 - 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 23 - 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

In the ULOHS scenario, a loss of heat rejection through the IHX leads to elevated cold pool and core inlet temperatures. Elevated temperatures in the core induce sufficient negative reactivity to shut down the fission process. As the transient continues, decay heat production gradually decreases until it is equal to the decay heat rejection capacity of the DHRS. The reactor vessel temperature levels off at 562°C, maintaining a margin of 170°C for ensuring structural stability of the reactor vessel.

In an ULOF accident, power is simultaneously lost to all primary and secondary sodium pumps. The ULOF transient is driven by an increasing power-to-flow ratio. After the primary pumps trip, the sodium flow rate begins to coast down with a flow halving time of approximately 10 seconds. Elevated temperatures at the top of the core induce a negative reactivity feedback from radial core expansion, causing the fission power to decrease.

Within the first minute, the rate of the flow decrease slows down as natural circulation is established and the power-to-flow ratio peaks at 1.9 of the nominal value. The peak cladding temperature increases from 568°C to 720°C but it only remains above 700°C for less than one minute. The peak coolant temperature follows a similar trend, increasing from 549°C to 719°C, maintaining a sufficiently large 234°C sodium boiling margin. Because power begins decreasing fairly quickly at the start of the transient, the effect on the peak fuel temperature is even smaller, increasing from 712°C to only 741°C.

After the flow rate starts to level off, power continues to decrease causing temperatures in the core to also decrease. Approximately four minutes into the transient, peak temperatures in the core have dropped below their nominal values and the reactivity feedbacks from Doppler, axial fuel expansion, and radial core expansion are slightly positive. The control rod driveline, however, had been heating up due to the hotter core outlet temperatures, inducing a negative reactivity feedback as the driveline expands thermally and inserts the control rods further into the core. The negative reactivity from control rod driveline expansion more than compensates for the other feedbacks maintaining a sufficiently negative net reactivity to prevent a self-sustaining chain reaction in the core.

In the longer term, decay heat continues to decrease. A small amount of fission power (less than 0.25%) is maintained. Although the cold pool temperature has increased nearly 100°C, the hot pool temperature increase was much smaller, approximately 30°C. With elevated cold pool temperatures, the heat rejection rate capacity of the

DHRS has increased from 0.5% to approximately 0.7%. By 17 hours, the fission and decay heat production in the core are matched by the DHRS decay heat rejection rate capacity and temperatures throughout the primary system gradually begin to decrease. The normalized power-to-flow and peak in-core temperatures for the ULOF transient are shown in the figures below.

The AOOs, DBAs, DECs, and SAs for FASTER include those that have been previously identified for metallic-fueled SFRs such as for PRISM and SAFR together with those particular to specific design features of FASTER that includes DBAs involving the closed loop systems. A category for SAs is included in the main document to indicate the proper characterization of an unprotected station blackout accident.

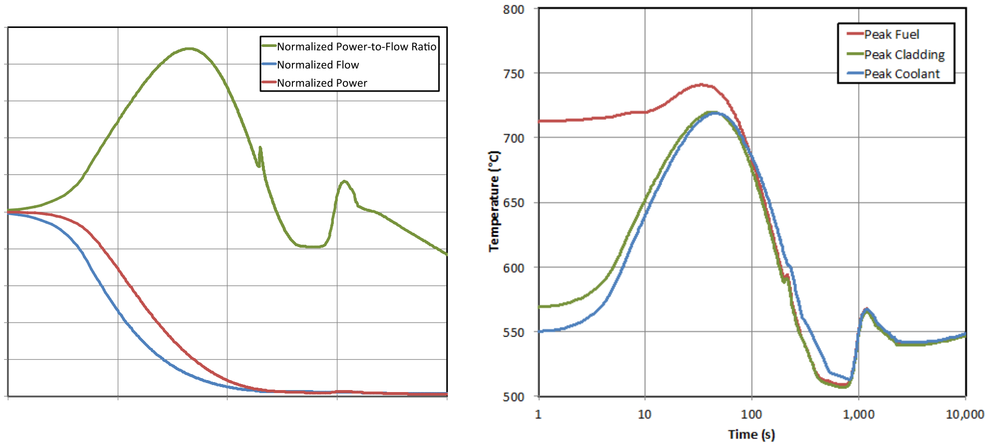

Figure 10 – (Left) Normalized Power, Flow, and Power-to-Flow Ratio, (Right) Peak In-Core Temperatures

5 Technology Readiness of Test Reactor Concept

The FASTER reactor plant, with its sodium coolant, pool plant geometry, and metallic fuel can trace its heritage to the beginning of fast reactor technology with the Experimental Breeder Reactor-II. Base sodium-cooled fast reactor technology has been utilized in the FASTER reactor concept to increase the technology readiness of the system and components.

Table 24 – TRL Evaluation of FASTER Reactor Plant Systems and Components

Component	TRL	Risk Description
Driver Fuel	TRL8	U-19Pu-6Zr - Reestablishment of domestic fabrication capability or foreign supply required. The preferred option would be to reestablish the capability of fabricating this fuel at a collocated facility with the FASTER reactor plant. Assumes that prior irradiation testing data is suitable for licensing.
Reflector Assemblies	TRL8	Reflector assemblies made from metal pins or blocks are conventional sodium fast reactor technology and thus achieve a high TRL level.
Shield Assemblies	TRL8	Shield assemblies made from clad boron carbide pins are conventional sodium fast reactor technology and thus achieve a high TRL level.
Control Rod Assemblies	TRL8	The FASTER control rod assemblies are constructed of boron carbide rods clad in HT-9. This technology is conventional sodium fast reactor technology and thus achieves a high TRL level.
Core Structural Materials – Fast Zone	TRL8	Demonstrated at full scale. Burnup is very low so there will be zero issue using HT-9 as the core structural materials in the fast neutron spectrum zone.
Core Structural Materials – Thermal Zone	TRL5	Literature review will need to be performed for structural materials in thermal zone. Zircaloy is compatible with sodium but is typically not used as core structural materials for fast reactor cores (cladding and ducting). Zircaloy is also an oxygen getter – so time in service will need to be understood and evaluated, but is not expected to be an issue. The zircaloy will also clad beryllium with the beryllium being a moderator.
Coolant Control Technology	TRL9	Coolant control technology using cold traps and nuclide traps is conventional proven sodium coolant control technology and has been demonstrated and operated at full scale. Coolant control technology used in EBR-II, FFTF, and developed for CRBRP can be adopted for use in the FASTER reactor plant.
Cover Gas Technology	TRL9	Argon cover gas is conventional proven technology and this will be used in the FASTER reactor plant systems. Nitrogen may also be used in some locations where needed for space inerting.
Cover Gas Cleanup Technology	TRL8-9	The cover gas cleanup technology will be initially based upon that used at EBR-II and FFTF and proposed for CRBRP with updates to technologies that may be used today for cleaning radioactively contaminated argon cover gas.
Gas Seal Technology	TRL7	Gas seal technology is known, but will need to be updated/evaluated from the 1970-1980's technology used in FFTF and EBR-II to determine whether advancements in elastomers and other gas seal technologies are suitable and perform better than the technologies used in the past.
Primary System Configuration	TRL7	Several components must be adapted from previous fast reactor designs, however, a pool plant geometry has proven to be most cost effective. However, no pool plant geometry has included 3 closed loops and multiple instrumented test assembly locations that stay in the reactor greater than one cycle.
Reactor Vessel and Structures	TRL8	For pool configuration, certain features such as the redan and reactor enclosure must be adapted for the particular geometry
Primary and Secondary Pumps	TRL8	The primary and secondary pumps are based upon mechanical centrifugal pump technology which has been deployed successfully in EBR-II, FFTF, and developed and tested for CRBRP. The exact pump will have characteristics that are specific for the FASTER reactor plant, but is expected to be smaller in capacity to the FFTF reactor plant.
Intermediate Heat exchangers	TRL8	The (4) IHX are tube-and-shell heat exchangers and are conventional sodium-to-sodium heat exchangers with a high TRL level. 9Cr-1Mo heat exchangers have been developed and installed in the Indian reactor. 9Cr-1Mo is a code qualified materials.

Component	TRL	Risk Description
Direct Reactor Auxiliary Cooling System	TRL8	DRACS emergency decay heat removal systems have been used in EBR-II and other sodium cooled fast reactors.
Direct Reactor Heat Exchangers	TRL8	The (3) DRACS heat exchangers are tube-and-shell heat exchangers and are based on conventional sodium-to-sodium heat exchangers with a high TRL level. These heat exchangers will have to be scaled appropriately for the size requirements for the FASTER reactor plant, but besides this, that is the only issue.
Air dump heat exchangers	TRL8	The (3) DRACS system each have one sodium-to-air heat exchanger which will be based upon the technology developed for FFTF, EBR-II, and CRBRP.
Steam Generator	TRL7	The steam generator adopted for the FASTER reactor plant is the helical coil steam generator (HSCG). There are two HCSGs that each provide 150MWth of superheated steam for the steam turbine. The HCSG is approximately twice the size of the HSCG that was developed by Babcock and Wilcox and tested at ETEC. That HCSG used 2.25Cr-1Mo steel for the tubes. The FASTER HCSG uses 9Cr-1Mo steel which is an acceptable alternative for heat exchanger materials. Superphenix installed a HCSG that was larger than needed for the FASTER reactor plant, but was made from Alloy600.
Sodium Water Reaction Protection System	TRL7	The technology for accommodating the sodium-water reaction resulting from a tube leak in a sodium-heated steam generator was developed in the past liquid metal technology development programs and tested at the Energy Technology Engineering Center.
Sodium Fire Mitigation System	TRL8	The technology for mitigation of sodium fires has been developed in past liquid metal technology programs.
Balance of Plant	TRL8	Superheated steam BOP is a conventional technology used with sodium-cooled fast reactor technologies around the world.
Containment	TRL8	Steel-reinforced concrete containment technology has been demonstrated at full scale both domestically and internationally.
Seismic Restraints	TRL8	Seismic restraints for piping and other components are expected to be conventional reactor technology.
Reactor Instrumentation and Control	TRL7	Demonstrated at full scale. The reactor project expects to adopt digital controls as appropriate.
Primary Control Rod Drive System	TRL7 TRL4 – Rod Stops for UTOP	The primary control rod drive system will be similar to the system for the AFR-100 plant which is an adaptation of the PCRDS from FFTF and CRBR. Interface between the PCRDS and the three instrumentation trees will need to be determined, but will be similar to technology developed for FFTF. Control rod stops will need to be implemented into the PCRDS system.
Secondary Control Rod Drive System	TRL6	The secondary control rod drive system is similar to the system for AFR-100 plant and the SCRDS for CRBR and ALMR. Engineering will have to be performed to ensure that the SCRDS will interface successfully with the three instrumentation trees in the FASTER reactor vessel.
Fuel Handling	TRL5	The starting point for the FASTER in-vessel refueling machines will be the machine developed for the AFR-100 plant. Mechanisms related to the in-vessel refueling machine will need to be evaluated to ensure that the machine will function as designed. If FFTF IVHM system was adopted a higher TRL would be assigned.
Maintenance and Inspection	TRL5	New techniques for in-service inspection, especially under sodium inspection, needs to be demonstrated. Technologies developed for FFTF and EBR-II need to be redeveloped.
FASTER Irradiation Capability		

Component	TRL	Risk Description
Closed Loop Technology	TRL8 for sodium TRL3-6 for other fast reactor coolants TRL3-6 for the thermal reactor coolants.	The starting point for the closed loop technology will be the designs that were created by Westinghouse for the FFTF reactor. There are three closed loops in the FASTER reactor plant – two fast closed loops and one thermal closed loop. The power/heat removal rate capability of the sodium closed loop will be identical to the FFTF reactor closed loop in-reactor assembly and primary module. It is expected that the two fast closed loops will be very similar in technology design to FFTF and thus will have a very high TRL. The thermal spectrum close loop will have a lower TRL mainly to reflect the design of the module that supplies the primary (in-loop) coolant for the thermal closed loop.
Instrumented test assembly technology	TRL7	The starting point for the instrumented test assembly technology will be the technology used in EBR-II and FFTF. The technology will be updated with modern sensors for temperature, flow, and other parameters as appropriate, but the technological foundation will be EBR-II, FFTF, and CRBR technologies and will evolve from there. Test
Non-instrumented test assembly technology	TRL8	Non-instrumented test assembly technology are configured identical to a fuel, reflector, and shield test assemblies. The technology to include test pins in fueled core assemblies is known and thus they are evaluated with a high TRL level.
Licensing		Technical criteria for advanced reactor licensing needs to be clarified and the regulatory structure re-established. It is expected that the selected ATDR reactor will be licensed by the U.S. NRC.
Fast Reactor Manufacturing Infrastructure		Infrastructure for manufacturing fast reactor components will need to be reestablished domestically if this is important to DOE, otherwise, there are international sources of liquid metal technology components such as pumps, valves, heat exchangers, etc.

6 Test Reactor Licensing, Development and Deployment Plans

For the PRISM and SAFR designs, which were also SFRs utilizing metallic fuel with strong inherent reactivity feedbacks and passive safety, Preliminary Safety Information Documents (PSIDs) were prepared by each vendor and submitted to the U.S. NRC, and preapplication interactions between each vendor and NRC staff were conducted. For PRISM, the NRC issued a Preapplication Safety Evaluation Report (PSER), NUREG-1368, concluding that “no obvious impediments to licensing the PRISM design had been identified.” Work on SAFR was discontinued by DOE before the NRC evaluation was completed but DOE requested that NRC document what they had done. The resulting SAFR PSER, NUREG-1369, concluded that the SAFR design had the potential for a level of safety at least equivalent to then current LWR plants. In 2010, General Electric Hitachi provided the NRC with a draft licensing strategy for the PRISM design for informal NRC consideration.

FASTER will be licensed under 10 CFR Part 50 as a testing facility that also produces electricity onto an electrical grid. The Preliminary Safety Analysis Report (PSAR) must include the principal design criteria for the facility. 10 CFR Part 50 Appendix A, “General Design Criteria for Nuclear Power Plants,” establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the NRC and provides guidance to applicants for construction permits in establishing principal design criteria for other types of nuclear power units. Historically, specific SFR principal design criteria were developed for CRBR, PRISM, and SAFR, instead of directly utilizing the General Design Criteria (GDC) from 10 CFR Part 50 Appendix A. A set of draft SFR design criteria has been developed under a joint initiative between DOE and the NRC. The criteria include five new criteria specific to SFRs. A report containing the SFR criteria was prepared by DOE and transmitted to the NRC. The NRC internally reviewed the criteria and is expected to soon issue a report for public comment. It is expected that this will be followed by NRC guidance including SFR principal design criteria. The FASTER design satisfies the current draft SFR principal design criteria and will satisfy the final criteria included in the NRC guidance.

Under the DOE-funded Regulatory Technology Development Plan, work has been launched to develop the SFR safety analysis codes and methods required for use in a licensing framework, and to identify the Quality Assurance (QA) requirements for licensing applications. A report, ANL-ART-37, was recently prepared that identifies the types of safety analysis computer codes that may be required for licensing of metallic-fueled SFRs and assesses the current status of existing relevant safety analysis codes including existing QA practices.

Pre-conceptual, conceptual, and final design will be carried out consistent with the DOE CD-0, CD-1, etc. process. A licensing strategy and schedule will be developed for FASTER. Preapplication meetings will be held with the NRC. As required, a PSID will be prepared. Interactions will be conducted with the NRC to pursue applications for a construction permit and operating license. A Preliminary Safety Analysis Report (PSAR) will be prepared as part of an application for a construction permit. A Final Safety Analysis Report (FSAR) will be prepared as part of an application for an operating license. Testing of the particular metallic fuel type used in FASTER shall be carried out to obtain the data needed to enable the use of this fuel. The FASTER design utilizes structural materials that are currently qualified for use by the appropriate ASME codes, with the exception of the Hastelloy-N alloy envisioned for use in closed loop systems for liquid salt. If a need for closed loop testing with liquid salt continues, then effort will be devoted to developing and submitting a code case for Hastelloy-N. Development of required safety analysis computer codes meeting QA requirements for licensing use shall be completed, and the codes will be used in preparation of the PSAR and FSAR. The FASTER design shall incorporate instrumentation to detect postulated sodium leakages from sodium piping and components consistent with ASME Boiler and Pressure Vessel Code requirements for Nuclear Power Plant components.

The licensing strategy will include a strategy for testing of components for FASTER including fuel assemblies, control rods and control rod drive systems, fuel handling systems, steam generators, intermediate heat exchangers, sodium pumps, as well as instrumentation for use in FASTER. Testing will be carried out, where feasible, in existing testing facilities, as well as new testing facilities that will be identified and assembled. Appropriate QA requirements for test data will be identified and followed.

Following the granting of an operating license, fabrication of the first core and loading of the first core, then criticality, low power testing and ascent to full power shall be carried out. During the ascent to full power, transient testing shall be carried out to determine the actual FASTER reactivity feedback behavior and to verify that it meets the requirements for inherent passive safety and inherent passive shutdown.

7 Economics and Schedule

7.1 Schedule

It is expected that the following notional schedule will be used for the design and construction of FASTER.

- 1 year for conceptual design
- 2 years for preliminary design
- 3-5 years for detailed design, licensing, and long-lead item fabrication
- <5 years for construction and final licensing

Total of 11 to 13 years for the FASTER reactor project from CD-0 with the assumption that there are no constraints on cash flow for the project and that licensing will not be the limiting factor. The schedule that includes the remaining research and development to get the FASTER power plant up and running will extend the schedule to 15 years or less with a concurrent R&D program. This schedule is consistent with a small modular advanced demo plant such as PRISM Mod A which has a power level slightly greater than FASTER. The sodium closed loop technology development will run in parallel with the schedule to construct and startup FASTER and use as its starting base, the sodium closed loop technology developed for FFTF. The non-sodium closed loop technology development will be independent of the main FASTER power plant construction, licensing, and operations and will not impact the schedule to startup FASTER.

7.2 Economics

It is estimated that 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 300MWth of heat to the atmosphere, then the TPC cost (design and construction) would be significantly less than $2.5B because the project would save both the cost for design, equipment, and construction for the steam plant (everything from the steam generators through the steam plant and electrical yard). In addition, the FASTER design team did not take any explicit credit during the cost estimate for prior design work and technology development work that may have been performed that would relate to the FASTER reactor plant. So, as the FASTER test reactor moves forward, more detailed cost estimates will better refine these cost figures.

The annual FASTER reactor plant operating costs in power generation mode 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) [14]. All estimates are in 2016 dollars. The replacement fuel is estimated to cost about $20M/year.

The FASTER reactor will provide 120MW$_e$ to the electrical grid at the location of installation. 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.

The cost and schedule estimates are based upon the best available information from the ALMR program, the FFTF project, and current consumer price index and construction cost index escalation factors averaged to 2016.

Operating and Maintenance Costs - The cost estimate to operate and maintain the FASTER reactor is divided into two estimated costs. One to operate the reactor and provide power and maintain the reactor and one that includes those costs and the costs to run experiments in FASTER. The costs to run experiments in FASTER include the costs related to operating and maintaining any experiments, secondary loops or other aspects of the facility such as hot cell facilities that are not related to the functions of the FASTER reactor which provide the experiment conditions (i.e., flux, temperature, etc.) and produce electrical power. Given the purpose of the FASTER reactor as a test reactor, it will be capable of supporting multiple large experiment programs simultaneously which may have significant costs associated with them. These costs are not included as part of the FASTER reactor operation and maintenance (O&M) costs. Those costs are assumed to be part of the larger annual operating costs.

Estimating the O&M costs is difficult because there are no directly comparable sources of data. The estimates must be extrapolated from a variety of sources, each of which has obvious issues with extrapolation to the FASTER reactor. There are three categories of cost. The smaller of the two are consumables (i.e., materials, supplies, etc.) and mandated fees (i.e., NRC fees, insurance, etc.). The largest cost is personnel.

The cost of consumables and fees is expected to be no more than $15M to $20M per year. These estimates are based on the costs associated with power reactors. The fees are not assumed to scale. The consumables are expected to scale, but far from linearly resulting in only modest reductions relative to a larger commercial-scale reactor.

The cost of personnel is based on the number of personnel required and the average cost of those personnel. These estimates are generally broken down to onsite and offsite staff. The onsite staff consisting of the technicians, maintenance, training, and other personnel involved primarily in the direct operation and maintenance of the plant. The offsite staff, which is not necessarily physically offsite, mostly provides the technical support and is a much smaller contributor to cost than the onsite staff.

The nature of the work done by these two manpower groups results in the average cost of onsite staff generally being significantly lower than the average cost of offsite staff. The specific mix for a given power plant will also impact this. The average salaries are expected to average around $100k per year for onsite staff and around $200k per year for offsite staff. For each, there will be a multiplier on these costs for benefits and other related staffing costs of no more than two times and more likely around 1.6 to 1.8.

The AP1000, a large commercial LWR is estimated to require an onsite staff of between 386 and 502, depending on whether it is located on an existing operational nuclear site or on its own. The commercial-scale S-PRISM is expected to have a total staff of roughly 650 for a twin-unit power plant. There are a lot of data on the variation in O&M costs for LWR stations as a function of size and units at the commercial scale. This information is used to identify where there will be cost savings for a smaller unit (e.g., annualized maintenance) and where there will not (e.g., one plant manager regardless). By combining this information and extrapolating it to a much reduced size, it suggests that a reactor of this size could be operated and maintained by a staff of significantly less than 200. To be conservative, since this is an extrapolation, the upper bound was assumed to be a total staff of 250 split into 210 onsite and 40 offsite personnel. This assumes a standalone site that receives no cost benefit from being on an existing nuclear site which will share a lot of resources, particularly related to security. This results in personnel costs in the $40M (staff of 200) to $60M (staff of 250) per year range.

The decommissioning and decontamination (D&D) costs are typically covered by a sinking fund payment. The annual cost of payments to such a fund will be a small fraction of the annual O&M cost assuming that the facility operates for decades. As a result, the D&D funds are within the overall uncertainty and do not impact the estimate at the current level of design detail.

The total O&M costs are expected to be in the range of $55M to $80M per year and certainly could be significantly lower. There may be significant additional costs associated with experiments conducted at this facility that are not included.

Fuel Cost Estimate - The cost associated with fueling the reactor is even more challenging to estimate than the O&M costs. The fuel cost estimate assumes that a fixed unit cost ($ per kg of heavy metal) is charged to the FASTER reactor that is equivalent to the cost of producing the fuel. This unit or levelized cost as it is typical referred to includes the capital investment, O&M, and cost of materials (e.g., plutonium, cladding, etc.). Like most nuclear facilities, the capital investment will be the largest contributor if a new facility must be built to supply the fuel. The other major contributors to cost are the cost of the fissile material and the O&M of the facility. It is assumed that the plutonium will be provided at no cost. The other materials (e.g., cladding and other assembly hardware) tend to be a relatively small cost contributor, although this needs to ultimately be confirmed since producing unique materials in small quantities can result in far higher unit costs.

The cost of manufacturing plutonium fuels on a commercial scale is estimated to be in the range of $2,500 to $7,000 per kg of heavy metal. This includes all costs except the cost of the heavy metal itself (i.e., cladding, assembly hardware, etc.), which is assumed to be negligible (Pu is free and DU will be at very small fraction at most). The assumption is that the manufacture of the FASTER reactor fuel will not require the construction of a standalone fabrication facility like the MOX plant at Savannah River Site, but will be done by upgrading an existing national laboratory facility to produce the relatively small quantities (~60 assemblies per year). There will be significant investment into this facility, but it was assumed that this would be bounded by the upper cost estimate of $7,000 per kg, which results in a cost estimate of approximately $300k per assembly. This would result in the fuel costs being no more than $20M per year. This equivalent annual cost implies an upfront investment in fuel fabrication capabilities in the $100M to $200M range.

Annual Operational Cost Estimate - When the O&M and fuel costs are added together, this results in an annual cost of no more than $100M per year and likely significantly less than that.

The annual costs of FFTF escalated for inflation would be on the order of $150M per year, but this seems to include all costs associated with the experimental programs that were utilizing FFTF and not just the operational cost of the reactor. Given the labor intense nature of the FFTF experience and the high costs of irradiation experiments, this value seems consistent with the expectation that the annual operational costs of the FASTER reactor will be between less than $150M per year.

8 Works Cited

[1] M. A. Smith, C. Adams, W. S. Yang, and E. E. Lewis, "VARI3D & PERSENT: Perturbation and Sensitivity Analysis," Argonne National Laboratory, ANL/NE-13/8, 2013.

[2] M. A. Smith, E. E. Lewis, and E. R. Shemon, "DIF3D-VARIANT 11.0, A Decade of Updates," Argonne National Laboratory, ANL/NE-14/1, 2014.

[3] D. C. Wade and E. K. Fujita, "Trends Versus Reactor Size of Passive Reactivity Shutdown and Control Performance," *Nuclear Science and Engineering*, vol. 103, pp. 182-195, 1989.

[4] K. L. Basehore and N. E. Todreas, "SUPERENERGY-2: A Multiassembly Steady-State Computer Code for LMFBR Core Thermal-Hydraulic Analysis," Pacific Northwest National Laboratory, PNL-3379, 1980.

[5] GateCycle. (2016) GE Power & Water GateCycle Software. [Online]. https://getotalplant.com/GateCycle/docs/GateCycle/index.html

[6] Hanford Engineering Development Laboratory, "A Summary Description of the Fast Flux Test Facility," HEDL-400, 1980.

[7] D.J. Oakley, "Closed Loop In-Reactor Assembly (CLIRA) A Fast Flux Test Facility Vehicle, HEDL-SA-1605FP," in *American Society of Mechanical Engineers Winter Annual Meeting*, San Francisco, 1978.

[8] Idaho National Laboratory, "FY 2009 Advanced Test Reactor National Scientific User Facility Users' Guide," INL/EXT-08-14709, 2009.

[9] U.S. Nuclear Regulatory Commission, "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, 1995.

[10] U.S. Nuclear Regulatory Commission, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," RG 1.183, 2000.

[11] D. Grabaskas, A. J. Brunett, M. Bucknor, J. Sienicki, and T. Sofu, "Regulatory Technology Development Plan - Sodium Fast Reactor: Mechanistic Source Term Development," ANL-ART-3, Argonne National Laboratory, 2015.

[12] Editor: Fanning, T.H., "The SAS4A/SASSYS-1 Safety Analysis Code System: User's Guide," ANL/NE-12/4, Argonne National Laboratory, 2012.

[13] U.S. Department of Energy, "SAFR - Sodium Advanced Fast Reactor," Preliminary Safety Information Document AI-DOE-13527, 1985.

[14] U.S. General Accounting Office, "Fast Flux Test Facility on Standby, Awaiting DOE Decision on Fugure Missions," GAO/RCED-92-121FS, 1992.

FASTER Reactor Plant

Appendix A: Self-Assessment against Test Reactor Metrics

This section provides an assessment of the proposed test reactor, FASTER, against the goals, criteria and metrics established as part of the ATDR study framework. The scores shown here have been determined by the design team that developed the FASTER point design and are briefly justified. More details have been provided in the main body of the summary point design report. In the following tables, the self-assessed score for FASTER is highlighted in green.

Goal T1. Test Reactor provides irradiation services for a variety of reactor and fuel technology options.

Criterion T1.1. Irradiation Conditions

Metric T1.1.1. Fast Flux conditions (>0.1 MeV)

Metric	>5 x 10^{15} n/cm^2-s fast (>0.1 MeV)	5x10^{14} to 5 x 10^{15} n/cm^2-s fast (>0.1 Mev)	<5 x10^{14} fast (>0.1 MeV)
Score	3	2	1

<u>Justification:</u> The peak fast flux achieved in the core has been determined to be 5.2x10^{15} n/cm^2-s in the fuel assemblies surrounding the test assembly located at the very center of the active core region. The radially averaged peak fast flux in this test assembly is larger than 5.0x10^{15} n/cm^2-s.

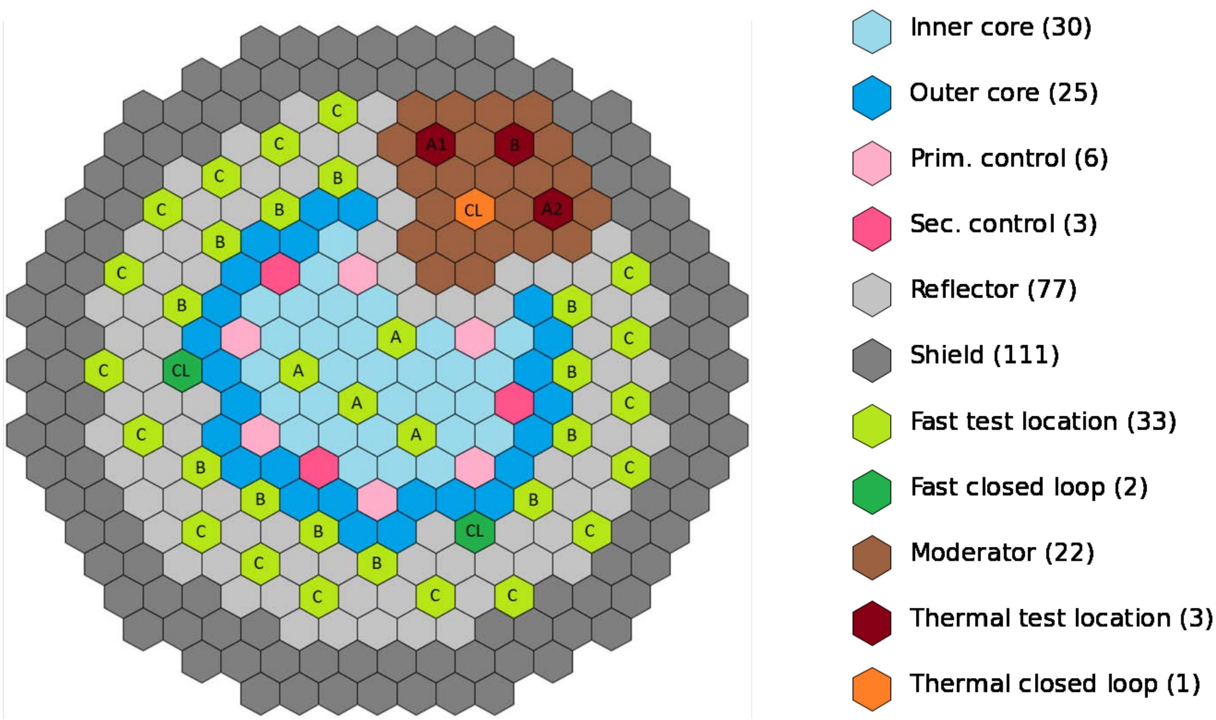

Figure 11 – FASTER Core Design with Test Locations Indicated

Table 25 – Summary of FASTER 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

Metric T1.1.2. Thermal Flux conditions (<0.625 eV)

Metric	>5x10^{14} n/cm^2-s thermal	1 to 5x10^{14} n/cm^2-s thermal	<1x10^{14} thermal
Score	3	2	1

Justification: Thermal flux is achieved through the use of beryllium-based moderator assemblies, making use of the neutrons leaking horizontally from the active core region. The peak thermal flux obtained in the thermal closed loop has been calculated to be 5.8x10^{14} n/cm^2-s, when using an energy threshold of <0.1 eV. With the energy threshold of <0.625 eV, the thermal flux will be significantly larger than the value reported here.

Table 26 – Summary of FASTER Thermal Flux Conditions in the Test Assemblies

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

Metric T1.1.3 Irradiation volumes and length for largest test location

Metric	Volume > 10 liters Length > 2 meter	5 to 10 liters volume 0.5 to 2 meter length	Volume < 5 liters Length < 0.5 m
Score	3	2	1

Justification: The total available length in any given test location is 277 cm (about the length of an assembly). Depending on the test assembly design, which will vary based on the irradiation experiment considered, the usable length inside a test assembly will be slightly less but likely >200 cm. Assuming a 200 cm effective length, this corresponds to a volume of about 24.4 liters. If subtracting the volume “lost” to the inter-assembly gap and duct thickness, the test volume is reduced to 20.8 liters. This is the volume for a single test location.

Metric T1.1.4. Maximum sustainable time at power, to provide a time-at-power for a single irradiation (i.e. cycle length)

Metric	> 90 days	45 to 90 days	< 45 days
Score	3	2	1

Justification: The FASTER core has been designed for a nominal cycle length of 100 effective full power days. Given that the required plutonium weight fraction is about 19.4%, it is possible to go to an even longer cycle length by slightly increasing the plutonium content of the fuel (still keeping it <20%).

Metric T1.1.5. Provisions for testing prototypic and bounding conditions (Temperature, Coolant, Chemistry)

Metric	Prototypic and bounding for different reactor coolants	Prototypic and bounding for base test reactor coolant	Not prototypic or bounding
Score	9	5	1

Justification: The FASTER closed loop in-reactor assemblies can be designed for coolants and temperature and pressure conditions exceeding those of representative reactor types (e.g., SFR, LFR, HTGR, Pebble Bed FHR, AHTR, PWR, Pressurized water-cooled and water-moderated research and test reactors) and provide ample space interior to the double-walled pressure tube for a flow tube to separate downward and upward flows and a core simulation inside of the flowtube. For example, a design temperature of 760 °C and design pressure of 7.82 MPa can be accommodated with a double-walled pressure tube exceeding the 750 °C core outlet temperature and 6.39 MPa pressure of a helium-cooled HTGR. For a pressurized water closed loop, a temperature of 650 °C and design pressure of 17.05 MPa can be accommodated with double-walled pressure tubes exceeding the core outlet temperature and primary system pressure of a PWR. The FASTER design team ranks the metric at a 9 for both prototypic and bounding for different reactor coolants.

Criterion T1.2. Support diverse irradiation testing configurations concurrently (accommodate various sizes and tailor irradiation parameters to wide group of simultaneous users)

Metric T1.2.1. Number of test zones

Metric	> 25 locations	10 to 25 locations	< 10 locations
Score	3	2	1

Justification: In the proposed configuration, the FASTER core offers 33 test locations exposed to a fast flux and three test locations exposed to a thermal flux. Given the nature of SFRs, the test locations in the reflector region are highly adjustable and their number can be significantly increased (or reduced), if needed.

Metric T1.2.2. Number and type of distinct irradiation test loops each with a different cooling system independent of the primary reactor coolant

Metric	3 or more	1 or 2	None
Score	3	2	1

Justification: In the proposed configuration, the FASTER core offers two closed loops exposed to a fast flux and one closed loop exposed to a thermal flux. Each of the three closed loops are placed at a 120° angle from one another. Almost any kind of coolant can be used in a given closed loop (except moderating coolants in the fast flux closed loop). Once the reactor is built, the coolant type can be changed by replacing the replaceable closed loop primary coolant module.

Metric T1.2.3. Ability to insert/retrieve irradiation specimen while staying at power

Metric	At power (e.g. rabbit)	Limited handling capability	Only at shutdown
Score	3	2	1

Justification: The rabbit tubes provide the capacity to insert and retrieve irradiation specimens while at power. The rabbit tubes can be located at the instrumented test assembly locations. In addition, the closed loop test assemblies can have the capability of being configured for insertion and removal of irradiation specimens while the reactor is at power. Closed loop test assemblies used in this configuration may be distinct from a closed loop whose main mission is to provide flowing coolant at prototypic conditions.

Goal T2. Test Reactor will be built and operated reliably and in a sustainable cost-effective manner. (Need to be able to justify initial and long-term expense)

Criterion T2.1. Project Costs and Schedule (including design, licensing, R&D, construction and contingency that reflects technical maturity of the concept)

Metric T2.1.1. Project cost

Metric	< $2.5 B	$2.5 – 4 B	> $4.0 B
Score	3	2	1

Justification: It is estimated that 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 300MWth of heat to the atmosphere, then the TPC cost (design and construction) would be significantly less than $2.5B because the project would save both the cost for design, equipment, and construction for the steam plant (everything from the steam generators through the steam plant and electrical yard). In addition, the FASTER design team did not take any explicit credit during the cost estimate for prior design work and technology development work that may have been performed that would relate to the FASTER reactor plant. So, as the FASTER test reactor moves forward, more detailed cost estimates will better refine these cost figures.

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 will provide 120MW_e to the electrical grid at the location of installation. 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.

The cost and schedule estimates are based upon the best available information from the ALMR program, the FFTF information, and current consumer price index and construction cost index escalation factors averaged to 2016.

Metric T2.1.2. Project Schedule - The time from today to first operation

Metric	< 10 years	10-15 years	>15 years
Score	9	5	1

Justification: FASTER is a small modular reactor (SMR) and shares the benefits that have been identified for SMRs including a shorter construction time relative to economy-of-scale NPPs. It is assumed that the pace of construction will not be limited by availability of funding or unanticipated licensing delays.

The overall schedule is expected to be the following after CD-0:

- 1 year for conceptual design
- 2 years for preliminary design
- 3-5 years for detailed design and procurement of long-lead items and site preparation
- < 5 years for construction and final licensing activities

The pace of construction will not be impacted by design and installation of closed loop technologies or instrumented assemblies which can be performed in parallel with the 11-13 year design and construction schedule. The schedule assumes that cash flow and licensing do not control the schedule. The metric has changed from the original metric. It is expected that from today, early 2016, that the schedule will be approximately 15 years or less due to the addition of R&D time that may or may not be 100% concurrent with the design, anticipated CD-0 in 2018 and first design funds arriving in that time frame to initiate the conceptual design. Furthermore, a highly optimized schedule, taking into account a lot of the design information created for EBR-II and FFTF, could result in operations of the reactor in a timeframe that is significantly less than 15 years from today. This schedule estimate is consistent with a demo reactor plant schedule, such as GE PRISM Mod A.

Criterion T2.2. Operational Costs and Schedule (including contingency that reflects technical maturity of the concept)

Metric T2.2.1. Annual operating costs

Metric	< $100 M/yr	$100-150 M/yr	> $150 M/yr
Score	3	2	1

Justification: The total O&M costs for operating the reactor are expected to be in the range of $55M to $80M per year and certainly could be significantly lower. There will be significant additional costs associated with experiments conducted at this facility that are not included. If the experiment costs are includes in the cost estimate, then it is expected that the total cost (annual reactor operating costs + experiment operations costs) will be between $100 and $150M/year and this is the value that we are assuming.

Criterion T2.3. Reliability of operations

Metric T2.3.1. Availability factor

Metric	>80%	60-80%	<60%
Score	3	2	1

Justification: The FASTER reactor plant has a 100 day refueling cycle and is a small compact core, just 300MWth in size and only 55 core assemblies. Replacing 1/3 of the 55 core assemblies every 100 days will take approximately 6 days for refueling. Replacing non-instrumented test assemblies will add some time to this estimate. Removing instrumented test assemblies and closed loop test assemblies will add additional time onto the reactor downtime, but these test assemblies are not replaced every cycle. In fact, they are expected to remain in the core for four (4) cycles. So, based upon this evaluation, it is expected that the FASTER reactor plant will have an availability of at least 80% that will improve with time.

Goal T3. Capability to accommodate secondary missions (e.g., electricity, isotope production, etc.) of modest value (million dollar) without compromising primary mission of testing fuels and materials for advanced reactor technologies

Criterion T3.1 Identification of Secondary Missions

Metric T3.1.1 Number of secondary missions

Metric	Sale of energy products	Other secondary missions	None
Score	3	2	1

Justification: FASTER is equipped with an energy conversion system which will allow converting the 300 MW_{th} into electricity and put it on the electrical grid. This is expected to represent significant revenue which will surely help recover a large fraction of the annual operating costs. Furthermore, special assemblies initially designed for FFTF could be modified for use in FASTER and would enable some isotope production which will further increase the revenue streams from the FASTER reactor plant.

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FASTER Test Reactor Preconceptual Design Report - Summary

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FASTER Test Reactor

1 Introduction

The FASTER reactor plant is a sodium-cooled fast spectrum test reactor that provides high levels of fast and thermal neutron flux for scientific research and development. 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 FASTER reactor plant has the following main attributes (Table 1):

Table 1 – FASTER Reactor Plant Summary Characteristics

Reactor Power	300MWt / 120MWe/40% efficiency
Coolant	Sodium
Coolant Temperature, inlet/outlet	355°C / 510°C
Coolant Pressure (cover gas 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/peak burnup	34.3 GWd/t
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 reinforced concrete containment
In-vessel Fuel Handling Mechanism	Single Rotatable Plug with pantograph FHM (3)

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 (Table 2 and Table 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.

Due to the required 30 page limit of this condensed report, the report focuses on the main mission of irradiation testing capabilities of the FASTER reactor plant and its performance during transients. The main report will contain further details.

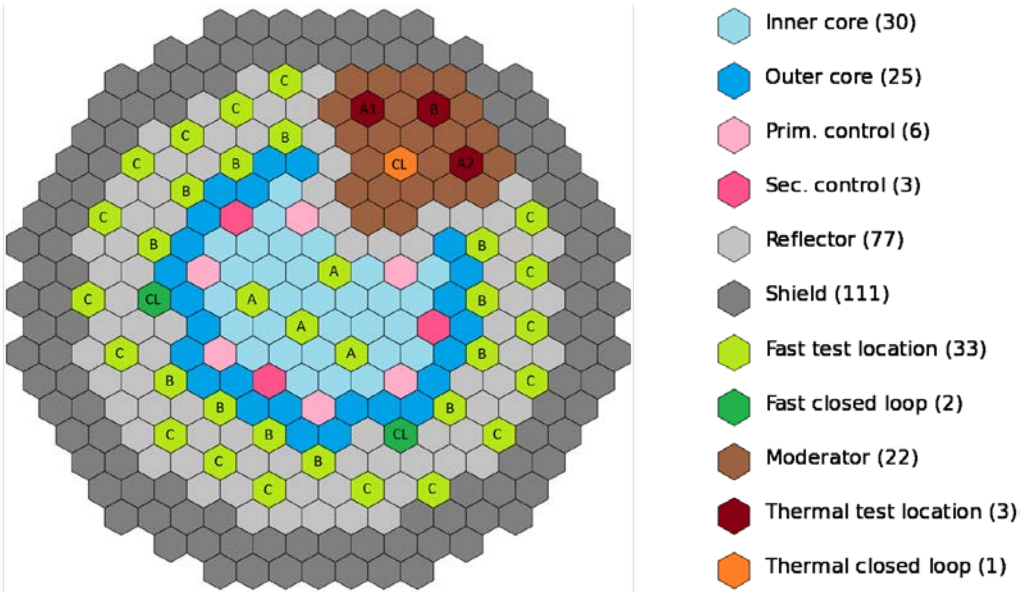

Figure 1 – FASTER Reactor Core Layout

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

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/s)
Closed loop	5.8	3.7
A1	1.9	1.2
A2	1.9	1.2
B	1.7	1.1

2 Test Reactor Objectives and Motivation for Concept 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 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 Test Reactor Point Design Description

Table 4 provides summary characteristics for the FASTER reactor plant.

Table 4 – 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
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PHTS Configuration	Pool plant geometry
Reactor vessel structural materials	Austenitic stainless steel
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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 assembly descriptions and dimensions are provided in Table 5.

Table 5 – Assembly Descriptions and Dimensions

Assembly type	Fuel	Reflector	Test	Control	Shield
Assembly pitch (hot)	11.870	11.870	11.870	11.870	11.870
Smear density	85	100	100	85	81
Pins/assembly	271	91	91	91	61
Pin Outside Diameter (mm)	5.405	11.072	11.072	9.512	13.462
Cladding Thickness (mm)	0.35	1.0	1.0	0.3	0.4
Pin Pitch/Diameter Ratio	1.210	1.023	1.023	1.024	1.012
Clad inner diameter (mm)	4.701	9.059	9.059	8.941	12.833
Outer Flat-to-Flat Distance (cm)	11.568	11.568	11.568	11.568	11.568
Duct Wall Thickness (mm)	3.0	3.0	3.0	3.0	3.0
Duct inside flat-to-flat distance (cm)	10.964	10.964	10.964	10.964	10.964
Duct Bundle Clearance (mm)	0.05	0.05	0.05	0.05	0.05
Inter-assembly Gap (mm)	3.0	3.0	3.0	3.0	3.0
Inlet coolant height (cm)	35.0	35.0	N/A	N/A	35.0
Lower reflector height (cm)	95.0	277.0	N/A	N/A	277.0
Fuel length (cm)	80.0	277.0	N/A	N/A	277.0
Fission gas plenum (flooded) height (cm)	12.0	277.0	N/A	N/A	277.0
Fission gas plenum (empty) height (cm)	40.0	277.0	N/A	N/A	277.0
Upper shield height (cm)	50.0	277.0	N/A	N/A	277.0
Outlet coolant height (cm)	30.0	30.0	N/A	N/A	30.0

For the purpose of the point design activity, the test locations have been modeled as entirely filled with sodium when determining the core neutronics performance characteristics, in order to maximize the neutron leakage probability and therefore the Pu wt% required in the fuel. For the safety analysis, including calculation of reactivity coefficients, the test locations have been filled will reflector assemblies in order to minimize the neutron leakage probability and not to overestimate the favorable effect resulting from sodium voiding.

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.

3.2 FASTER Reactor Performance Characteristics

The core performance characteristics at equilibrium are provided in Table 6. It should be noted that most of those characteristics will vary slightly based on the type and amount of materials loaded in the test locations, in particular those located in the active core region. In practice, a new core analysis will be required for every new core configuration, including each time a new test assembly is inserted. However, the results presented here are expected to be conservative and not to vary significantly. The reactor power level of 300 MW_{th} is the minimum that assures that the neutron flux goals can be met.

The required Pu wt% is 19.4%, which is slightly lower than the 20% limit used. The average fuel discharge burnup is about 34 GWd/t while the peak discharge burnup is less than 50 GWd/t. This is consistent with the assumptions made for the fuel smear density and plenum length, which were based on a low fuel burnup. It is important to note that the average core power density is about 560 W/cm^3 over the active fuel region. When normalized only over the fuel volume, this corresponds to an average power density of 1580 W/cm^3 and a peak value of almost 3000 W/cm^3. This is larger than what is typically observed in SFRs, and proper cooling of the core will need to be ensured. The thermal flux provided in this table is calculated for the moderated region.

Table 6 – Core Performance Characteristics

Characteristic	FASTER
Nominal power, MW_{th}	300
Required enrichment	19.41%
BOC/EOC k_{eff}	1.02672/1.00080
Average discharge burnup, GWd/t	34.3
Total HM mass, kg	2621
HM charge per cycle, kg	874
Average power density, W/cm^3	558.8
Peak power density, W/cm^3	917.0
Specific power, W/g	114.5
Average linear power, kW/m	25.2
Axial/radial leakage	11.36%/30.17%
Total flux, 10^{15} n/cm^2-s	3.74
Peak total flux, 10^{15} n/cm^2-s	6.78
Fast flux (>0.1 MeV), 10^{15} n/cm^2-s	2.70
Peak fast flux (>0.1 MeV), 10^{15} n/cm^2-s	5.19
Peak thermal flux (<0.1 eV), 10^{14} n/cm^2-s	6.19
Peak thermal flux (<0.625eV), 10^{14} n/cm^2-s	>12.0

The core power density distribution is shown in Figure 2, and since almost no energy is deposited in the moderated region it does not show up in this figure. It is also observed that the peak value is reached at the core center and not at the interface between the fast and moderated region, indicating that no thermal neutrons are returning from the moderated region.

The fast flux and thermal flux distributions are shown together in Figure 3. Different scales are used in the two regions. The thermal flux is only significant on the right-end side of the figure, corresponding to the moderated region of the core, while the fast flux is only largest in the active core region where neutrons are produced. Although not shown in Figure 3 because of the threshold value used, the fast flux remains significant in the entire reflector region. It is reduced by a factor of ~50 between its peak value at the core center and the values observed in the test assemblies located the farthest away from the core center (i.e., in the reflector).

The only region in which a significant thermal flux is observed is the moderated region. All the neutrons present in this region are neutrons leaking from the active core region and being moderated. Although this region is not fully optimized, a peak thermal flux of at least 6.0×10^{14} n/cm^2-s is achievable solely by using leaking neutrons. In this study, the energy threshold used for thermal neutrons is taken to be 0.1 eV. However, the metrics used to evaluate the test reactor performance later stated that the threshold energy for thermal neutrons should be taken as 0.625 eV. This means that the thermal fluxes claimed in this document are conservative and would be two to three times larger when using the energy threshold stated in the ATDR study.

3.3 Reactivity Coefficients and Quasi-Static Balance

The reactivity worth of the primary and secondary control rod assemblies (PCRA and SCRA) has been determined for the FASTER core in order to ensure that the core can be safely shut down from any operating condition. All nine control rods present in the core are composed of a 90 cm tall region containing a 47.58% by volume of B_4C, as well as 29.4% sodium and 23.1% HT-9. The B_4C is assumed to have an 85% smear density (i.e., the effective B_4C vol% is 39.8%). Natural boron is used in both the primary and secondary control systems. During nominal operations, the secondary control rods are fully withdrawn and only the primary control rods are used to compensate for the burnup reactivity swing.

At the beginning of the equilibrium cycle (BOEC) the PCRAs need to be inserted by about 34 cm to make up for the initial excess reactivity, and by the end of the equilibrium cycle (EOEC) PCRAs are fully withdrawn from the core. Although the requirement is that any of the control systems must be able to safely shut down the core with the highest worth control rod being stuck in the operating condition, the total worth of each system has also been evaluated when no rod is stuck. The results are shown in Table 7.

Table 7 – Reactivity Worth of the Primary and Secondary Control Systems

Worth [$]	BOEC	EOEC
Primary	21.1	22.0
Primary –rod stuck	17.3	17.9
Secondary	10.6	11.2
Secondary –rod stuck	5.7	6.1

The methodology used to determine the shutdown margins for each reactivity control system is not detailed here, but it is the methodology typically used for fast reactor analysis. In particular, it accounts for the reactivity change between the "resting state of the core" and its operating condition, for a single rod reactivity fault (ejection), considering uncertainties and a number of other factors. The shutdown margins, with the highest worth control rod being stuck, are provided in Table 8. Given that the core has the most excess reactivity at BOEC, this is when the shutdown margins are the smallest. The margins obtained are very satisfactory and could allow for reducing the B_4C volume fraction used, if needed.

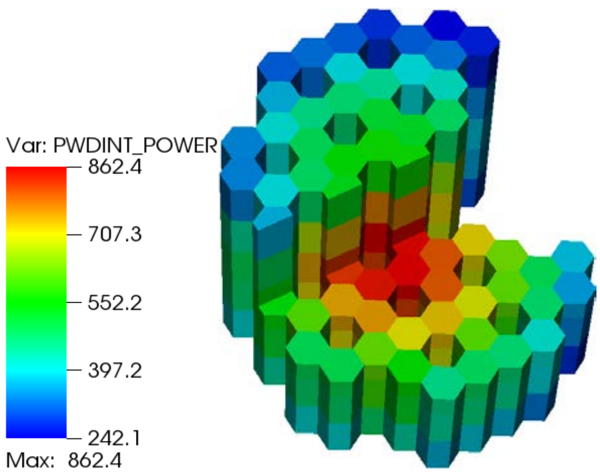

Figure 2 – Power Density Distribution (in W/cm^3)

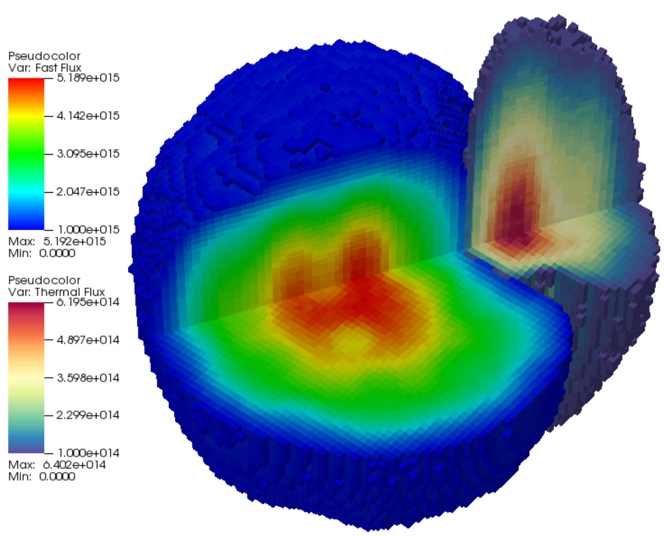

Figure 3 – Fast and thermal fluxes distribution (in n/cm^2-s)

Table 8 – Reactivity Shutdown Margins of the Primary and Secondary Control Systems

Shutdown margin [$]	BOEC	EOEC
Primary	3.21	15.14
Secondary	3.59	5.04

The reactivity coefficients required for the safety analysis have been determined at BOEC and EOEC using the PERSENT [1] code coupled with DIF3D [2]. Although not detailed here, the assumptions used for determining these coefficients are commonly used for fast reactor analysis. The various reactivity coefficients calculated are summarized in Table 9. It is important to note that the sodium void worth is negative during the entire cycle despite using Pu-based fuel. This is due to the very large neutron leakage probability and high quality Pu being used. Another noteworthy reactivity coefficient is the radial expansion which is observed to be very negative due the active fuel region having a relatively small height-to-diameter ratio and very large neutron leakage probability. Other reactivity coefficients are typical of a SFR fueled with metal fuel.

Table 9 – FASTER Reactivity Coefficients

		BOEC	EOEC
β_{eff}		0.00332	0.00333
Prompt lifetime	μs	0.349	0.373
Generation time	μs	0.348	0.373
Radial expansion	cent/°C	-0.445	-0.451
Axial expansion	cent/°C	-0.126	-0.118
Sodium void worth	$	-2.242	-2.420
Sodium density	cent/°C	-0.063	-0.067
Doppler	cent/°C	-0.109	-0.119
Sodium voided Doppler	cent/°C	-0.089	-0.097
Fuel density	cent/°C	-0.820	-0.800
Structure density	cent/°C	0.009	0.010

The quasi-static reactivity balance [3] has been performed for the FASTER core, using the reactivity coefficients previously discussed. The results are provided in Table 10. The single rod reactivity fault, $\Delta\rho$, is about 1.13$ at BOEC and nearly 0.0$ at EOEC. Although the results indicate that all three required conditions are met and that the core is expected to be inherently safe, it might still be necessary to use control rod stops in order to reduce the single rod reactivity fault and limit the maximum temperatures reached during unprotected transients.

Table 10 – Quasi-static Reactivity Balance Coefficients and Conditions

Coefficients		
A - Power/flow reactivity decrement (cents)	-33.9	-34.2
B - Power/flow coefficient (cents)	-141.4	-106.4
C - Inlet temperature coefficient (cents/° C)	-1.4	-0.9
Quasi-static reactivity conditions		
A/B <= 1	0.24	0.32
1 <= CDT_c/B <= 2	1.51	1.34
$\Delta\rho$/|B| <= 1	0.80	0.03

3.4 Orificing Strategy and Temperature Distribution

In order to achieve the desired flux levels, the FASTER core has been designed with a very high power density and therefore adequate coolability of the core needs to be ensured. The multi-assembly steady state thermal hydraulic code SE2-ANL [4] is used to determine orifice groups and calculate the required coolant flow rates and temperature distributions. The inlet coolant temperature is 355°C and the bulk outlet coolant temperature is 510°C, corresponding to an average coolant temperature rise of 155°C. These temperatures are typical for SFRs with metallic fuel.

In order to limit the peak coolant and peak mid-wall cladding temperatures to acceptable values it was found that using four orifice groups for the fuel assemblies is sufficient. The temperature distribution has also been determined when using a larger number of groups, but this resulted in a reduction of the peak temperatures of only a few degrees. The proposed orifice groups are represented in Figure 4.

Seven orifice groups are used for the non-fuel assemblies, but the flow rates in those assemblies are significantly smaller than in the fuel assemblies, meaning it is possible to use more or fewer groups without significantly affecting the bulk outlet coolant temperature. Additionally, test assemblies and closed loop locations are currently modeled with reflector assemblies and will likely require specific orifice sizes depending on the desired test conditions.

The coolant flow rate, velocity, number of assemblies, and other characteristics are provided in Table 11 for each orifice group shown in Figure 4. The temperatures are provided in °C. The coolant velocity required in the orifice group containing the highest power assemblies is about 10 m/s. This is only the coolant velocity required inside the active fuel region, and is different from the velocity of the coolant exiting the core and going to the heat exchanger. The peak 2σ mid-wall temperature for the non-fuel assemblies is larger than for the fuel assemblies, but this is not a concern since no fuel-cladding interaction will occur.

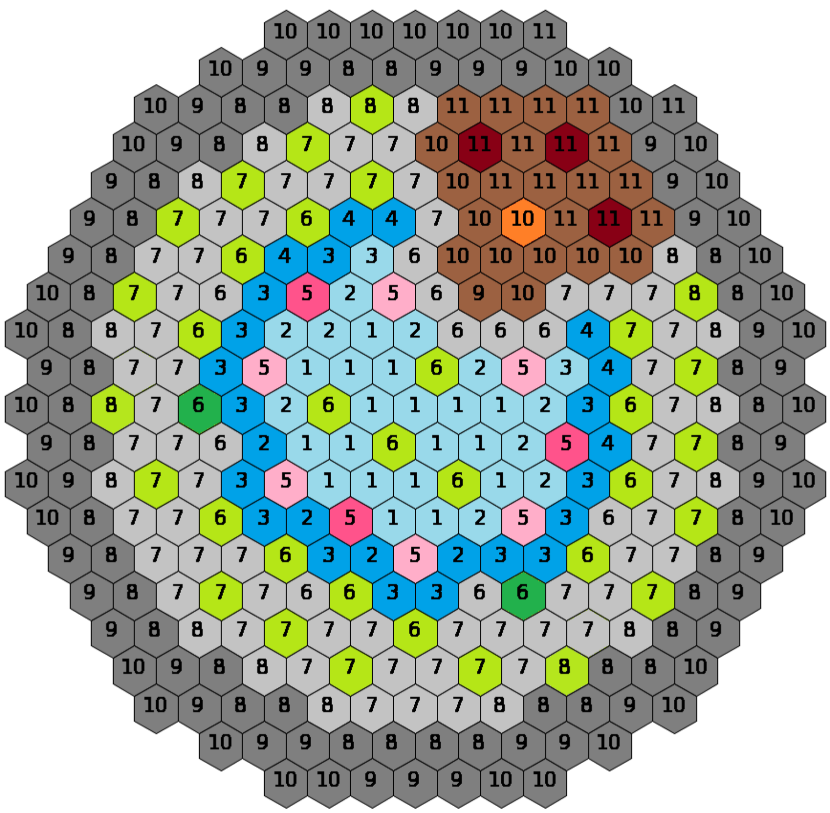

Figure 4 – Orifice Groups

Table 11 – Coolant Flow Characteristics for each Orifice Group

	Group	Assembly #	Flow rate /assembly, kg/s	Total flow rate, kg/s	Velocity, m/s	Average outlet coolant Temp., °C	Peak outlet coolant Temp., °C	Peak 2σ mid-wall Temp., °C
Fuel	1	18	34.53	621.5	10.3	510	549	587
	2	14	28.07	392.9	8.4	510	548	587
	3	17	21.43	364.3	6.4	510	549	587
	4	6	16.77	100.6	5.0	513	550	587
Non-fuel	5	9	0.55	4.95	<1	506	744	n.a
	6	26	0.55	14.3	<1	514	779	1039
	7	67	0.22	14.9	<1	499	667	909
	8	56	0.068	3.82	<1	495	589	786
	9	37	0.018	0.68	<1	480	553	739
	10	46	0.006	0.28	<1	473	525	620
	11	17	0.001	0.02	<1	489	533	636

3.5 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 5). The primary pumps are mechanical centrifugal pumps with the characteristics shown in Table 12.

Table 12 – Primary Pump Design Characteristics

Flow rate, m^3/s	758.3
Pump head, kPa	704
Power, kW	950
Efficiency, %	90
Pump length, m	9.18
Pump diameter (case), m	0.889

The IHXs are conventional 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 design characteristics of the IHX are provided in Table 13.

Table 13 – IHX Design Parameters

Heat transfer capacity, MW	75
Thermal Design Margin, (for average case)	±%25
Heat transfer area, m^2	231
Primary sodium temperature, inlet, °C/outlet, °C	355 / 510
Primary sodium mass flowrate, kg/s	379.15
Secondary sodium temperature, inlet, °C/outlet, °C	279 / 499
Secondary side sodium mass flowrate, kg/s	265
Tube outer diameter, cm	1.59
Tube wall thickness, mm	0.889
Tube pitch, cm	2.5
Effective tube length, m	3.85
Number of tubes	1,200
Tube sheet thickness, mm	100
Downcomer piping-OD, cm	32.385
Downcomer piping-thickness, mm	9.525

Downcomer piping-length, m	TBD
Outlet piping –OD, cm	45.72
Outlet piping –thickness, mm	9.525
Outlet piping –length, m	TBD
Shell (primary) side pressure drop, kPa	15.96
Tube (secondary) side pressure drop, kPa	30.76
Shell height, m	4.96
Shell outside diameter, m, main body/maximum	1.038/1.145
Shell thickness, mm	19
Tube material	9 Cr-1Mo

The intermediate heat transport system (IHTS) (Figure 6) 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 components are connected via the IHTS piping.

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 traps located in the primary and intermediate coolant systems. These cold traps will remove oxygen, hydrogen, and other impurities via conventional crystallization techniques. 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.

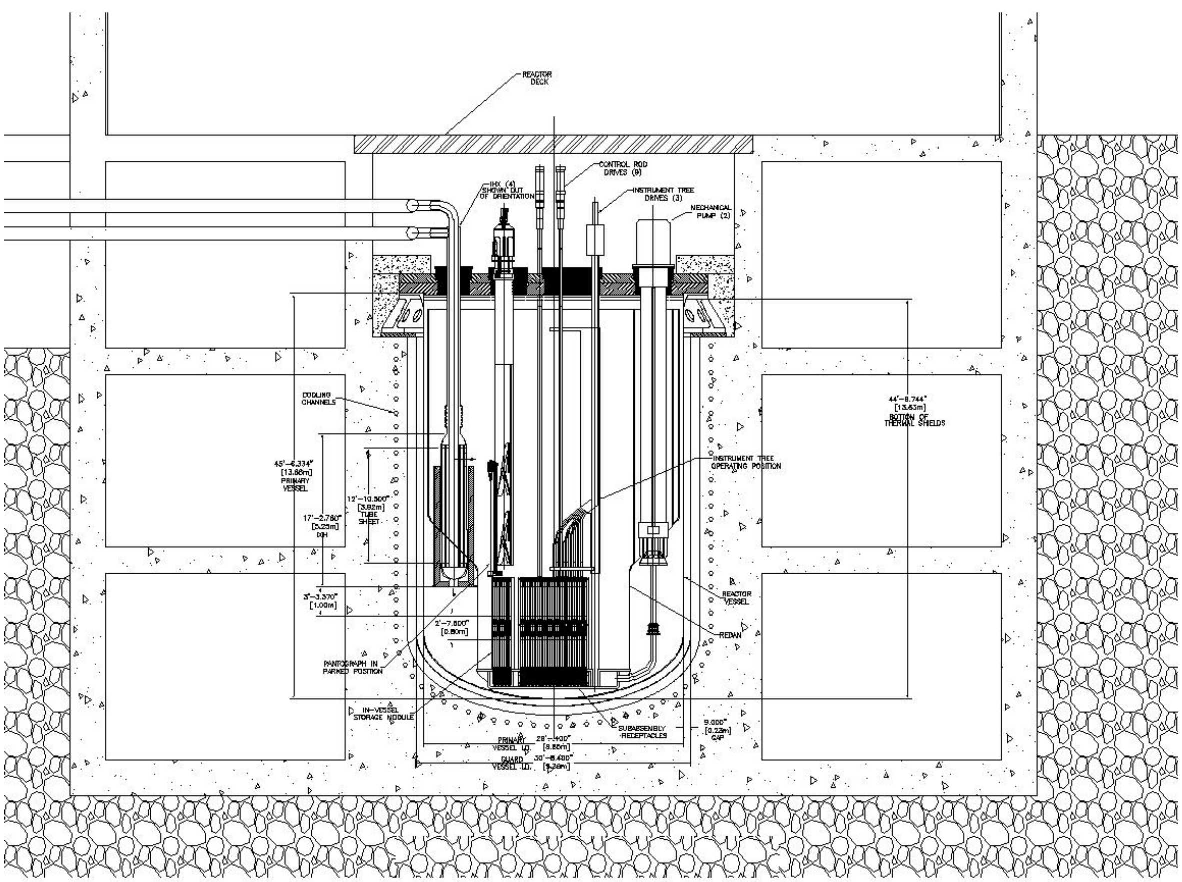

Figure 5 – Elevation View of PHTS

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. The containment and those parts of the nuclear island (NI) containing sodium piping and components incorporate design features to mitigate the effects of postulated sodium leakages and sodium fires such that temperature and pressure loadings following sodium leakage remain small or negligible. These features include sodium leak detectors, detecting a sodium leak when it is small before it can grow, shutting down the pump and draining the sodium from a failed sodium loop into the loop sodium dump tank to limit the amount of sodium released upon detection of a sodium leak or sodium burning, automatic isolation of ventilation lines serving a compartment upon detection of aerosols in the outlet duct, compartmentalization to reduce the volume of an air-filled partially-sealed compartment housing sodium piping or components thereby reducing the amount of available oxygen such that a sodium fire will rapidly consume the available oxygen and burn itself out, use of sodium catch pan fire suppression decks to significantly reduce the sodium burning rate from a sodium pool and protect the underlying concrete, confining sodium released from a failed pipe inside the gap between the pipe and surrounding thermal insulation and draining it through drain pipes to eliminate or significantly reduce the potential for formation of sodium jets or sprays, and the use of steel liners on compartment inner surfaces to further protect concrete. These mitigation approaches were previously developed and tested as part of the CRBR, PRISM, and SAFR design activities.

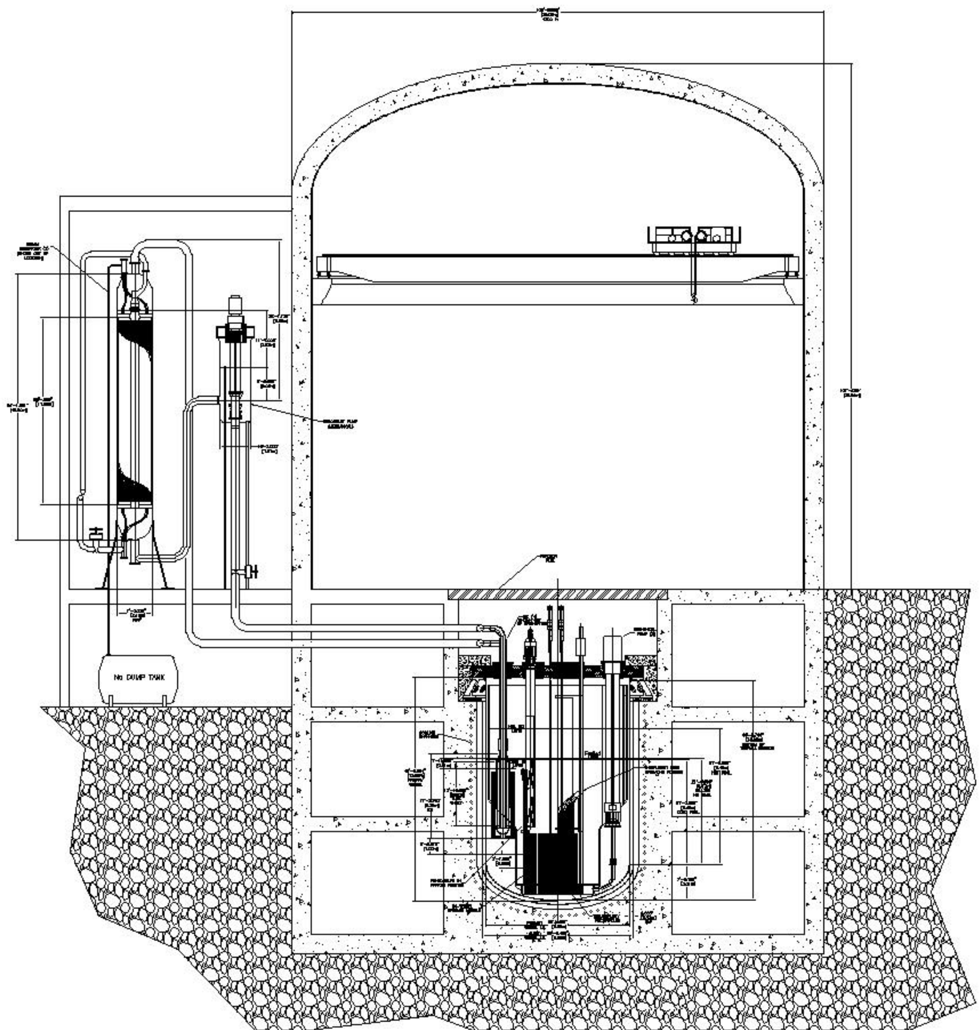

Figure 6 – FASTER NSSS – Elevation View

Emergency decay heat removal is provided through three 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.

Table 14 – DRACS HX - each

Heat transfer capacity, KW	750
Heat transfer area, m^2	7.64
Primary sodium temperature, inlet, C°/outlet, C°	510/355
Primary sodium mass flowrate, kg/s	3.793
Secondary NaK temperature, inlet, C°/outlet, C°	328/484
Secondary NaK mass flowrate, kg/s	5.47
Tube outer diameter, cm	2.22
Tube wall thickness, mm	0.9
Tube pitch, cm	3.79
Effective tube length, m	2.489
Number of tubes	44
Shell OD, cm	32.26
Shell wall thickness, mm	9.525
Material	9Cr-1Mo

The balance of plant consists of a conventional superheated steam cycle attached to the (2) once-through sodium heated steam generators. Conditions are calculated with the GateCycle software [5] (Figure 7).

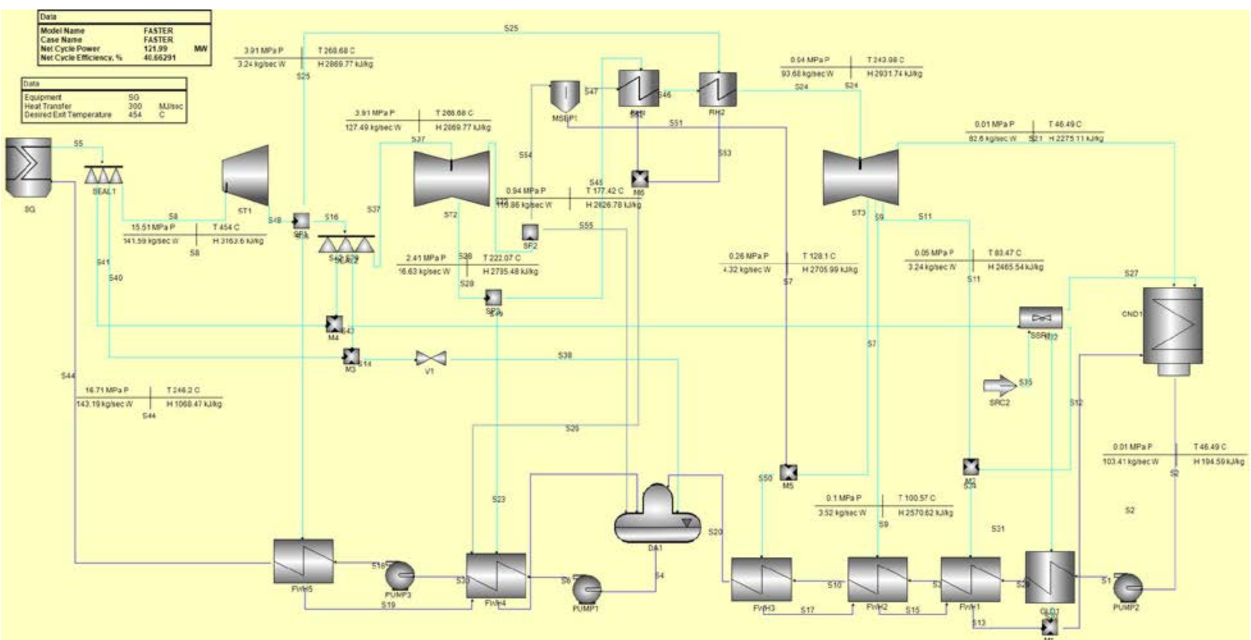

Figure 7 – FASTER Thermodynamic Cycle and Balance of Plant

3.6 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.

3.7 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 8 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 15. 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.

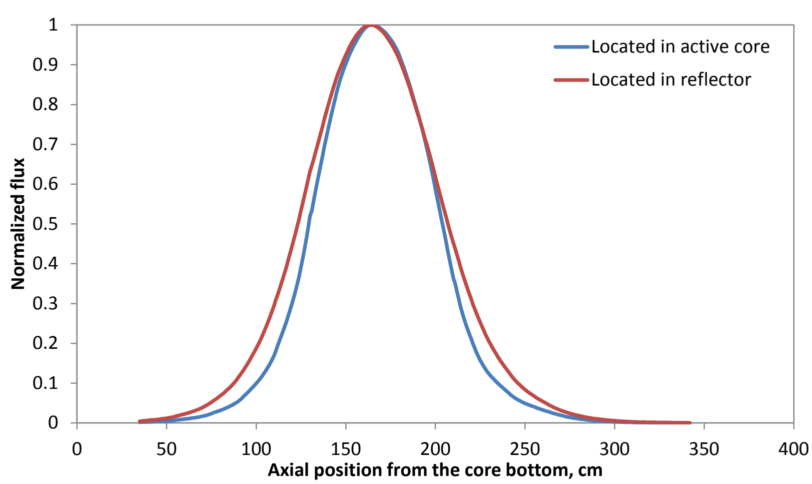

Figure 8 – Normalized axial fast flux distribution in test locations

Table 15 – 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 16 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 9. The rough aspect of the curve is due to the uncertainties of the calculations performed with MCNP.

Table 16 – 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

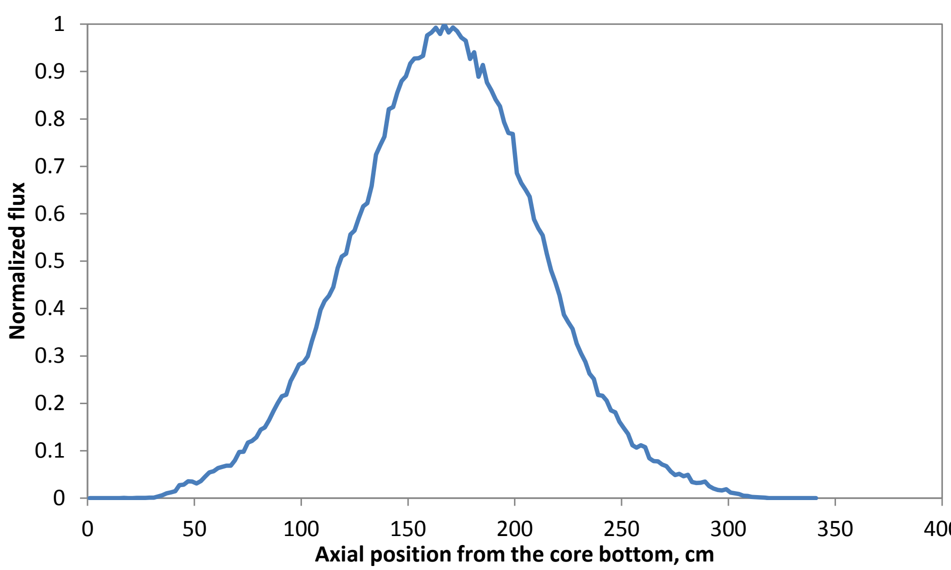

Figure 9 – Normalized axial thermal flux distribution in the test locations

3.8 Closed Loop Systems

The three closed loop systems (CLS) are an important capability of FASTER and part of the study 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 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 fuel, 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 [6], [7] 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. Pressurized water CLs are also an integral part of the Advanced Test Reactor (ATR) design and are utilized for irradiation and testing [8].

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 [6], [7]. Heat removal rate and coolant flowrate requirements for different coolants for different example reactor designs are shown in Table 17 for test sections simulating a small portion of each reactor core inside of the flow tube. 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 17 - 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, $2LiF\text{-}BeF_2$	Liquid Salt, FLiBe, $2LiF\text{-}BeF_2$	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 Velocity, m/s	5.52	5.52	2.0	1.94	2.0	20.2	4.85	15.5

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 17. The required pressure tube dimensions for a design temperature of 649 °C (1200 F) are shown in Table 18. 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 insure 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 17.

Table 18 – 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 (1300 F), ample space is still available with the low pressure and pressurized helium coolants (Table 19). There still remains space when the design temperature is further increased to 760 °C (1400 F) as shown in Table 20.

The test sections in the FFTF closed loop in-reactor assemblies were designed for a sodium outlet temperature of 760 °C (1400 F) while the double-walled pressure tube and other closed loop hardware was designed for 649 °C (1200 F). 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 19 – Required Pressure Tube Dimensions for 704 °C (1300 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)	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 20 - Required Pressure Tube Dimensions for 760 °C (1400 F) 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 21. 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 21 - 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 to Preclude Leakage	Double-Wall Flow Tube with Monitored Gap to Preclude Leakage	Double-Wall Flow Tube with Monitored Gap to Preclude Leakage	Double-Wall Flow Tube with Monitored Gap to Preclude Leakage and for Thermal Insulation	Double-Wall Flow Tube with Monitored Gap 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 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 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 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.

Details of the closed loop modules are provided in the Table 21 and in the main report.

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) [7]. 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 incorporated a tungsten funnel to collect fuel, a TZM molybdenum alloy cup with six inwardly projecting fins to contain molten fuel, and a tungsten washer followed by a borated graphite shield block beneath the TZM cup. For each individual CLS primary coolant type and reactor type core simulation, an assessment will to be carried out of the accident phenomena and an approach to contain the test section materials as well as mitigate the release of radionuclides from the in-reactor assembly will be engineered. This type of analysis could not be carried out during the time frame of this study.

3.9 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 main capabilities of the instrumented assemblies envisioned for FASTER are hereunder described. Considering, as a minimum requirement, the same capabilities offered by the FFTF design, instrumented assemblies loaded with fuel elements allow for monitoring of parameters such as sodium temperature within the fuel bundle, cladding temperature, duct temperature, fuel centerline temperature, and fission gas pressure within the pin. Transducers can also be used to measure differential pressure within the fuel bundle. In FFTF, up to fifty-one leads for pressure, temperature, and electrical connections were used. Such a number represents a lower bound for a FASTER instrumented assembly. In addition, in FFTF, open test assemblies could be removed from the core after first severing the instrumentation leads, and then reinserted into the core for a post-irradiation open test after examination in a hot cell.

Additionally, instrumented assemblies loaded with encapsulated structural experiments allow for the evaluation of irradiation behavior of structural materials. These assemblies will provide detailed temperature control and measurement during irradiation as well as density and dimensional measurements on test specimens made ex-core during reactor shutdown. In FFTF, the open test assemblies loaded with material specimens could contain up to forty-eight canisters, thirty of which with independent temperature control, and a total of 2500 cm^3 of in-core irradiation volume per assembly. In addition, up to eighty-two leads for pressure, temperature and electrical connections were available. Again, this is to be considered a lower bound for FASTER instrumented assemblies. Instrumented assemblies loaded with structural material specimens are also designed to be removed from the core at the end of an operating cycle, the test specimens then examined, and the experiment then reinserted into the core for the start of the next cycle.

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 insert 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.

3.9.1 Identification of prototypic or scalable aspects of fuel performance testing.

The instrumented assembly will provide a one-of-a-kind capability for validation of advanced simulation tools. Fuel pin wire wraps induce complex three-dimensional mixing that historically has not been accurately represented in design calculations. Designers understood the limitations of their models and correspondingly introduced uncertainty factors to account for lack of fidelity. In turn this leads to a derating of the core. Ongoing research and development of multi-physics simulations for improving fast reactor economics is specifically targeting improved prediction of subassembly temperatures by better modeling the effects of wire-wrapped induced mixing. The usefulness of these models for future design work and recovery of thermal margin is critically dependent on validating these models against prototypic experiment data.

The instrumented assembly offers a unique opportunity to generate prototypic experiment data for validation of these advanced computational design tools. With the multi-lead capability provided by the instrumented assembly, it is possible to provide a spatially rich set of temperature measurements within the fuel assembly. The needed measurements can be acquired using two different measurement capabilities. The first uses thermocouples embedded in wire wraps to sense local coolant temperature. The second approach replaces the wire wrap with a stainless steel capillary of the same outside diameter through which is threaded an optical fiber. This so-called *Bragg fiber* is capable of generating temperature data to the sub-centimeter level along its length yielding here-to-fore unrealized spatial resolution of coolant temperature within a fueled assembly. The development of these fibers for higher temperature applications is ongoing. It is expected that soon there will be a fiber capable of operating at SFR temperatures for the time needed at full power to acquire the needed steady-state temperature data.

3.10 Plant Security and Safeguards

The plant is separated into nuclear and nonnuclear areas, referred to as the nuclear island (NI) protected area and the balance-of-plant (BOP) controlled area, respectively. The NI contains all safety-grade components and systems, while all non-safety-grade components and systems are located in the BOP area or the owner-controlled area. This results in a zoned approach to security with fewer facilities and personnel within a small NI-protected area.

In addition, the plant has inherent and unique operating characteristics and safety features that reduce the scope of the security system. These include:

- Physical separation of the NI from the BOP with no safety related components and systems in the intermediate heat transport system (IHTS) or BOP controlled area. This is a feature of the FASTER SFR design that does not need the IHTS or BOP for emergency heat removal
- Inherent, passive reactivity feedbacks
- Multiple passive decay heat removal systems where only 2 out of 3 are required
- Long grace period before corrective action is needed.

The FASTER security system is designed to protect special nuclear material (SNM) and vital equipment in the most efficient way so that cost, operational impact, and security force size are all minimized. The design integrates the often competing objectives of assuring nuclear plant safety and physical security while not adversely affecting the safety of plant personnel. Thus, the design reflects the interface between the safety and security requirements.

The security system is designed to:

- Allow authorized personnel and material access to the facility
- Keep out all unauthorized personnel and material
- Detect and verify all unauthorized activities
- Delay unauthorized activities
- Deter potential adversary actions
- Prevent theft of special nuclear material
- Prevent sabotage of vital equipment.

The physical security system for the NI-protected area has the following primary components:

- Multiple perimeter barriers and isolation zones
- Redundant intrusion detection systems
- Surveillance and assessment equipment
- Hardened guardhouse facilities
- A computer-based card-key access control system
- Positive personnel identification systems
- Central and secondary alarm stations
- Redundant and uninterruptible power systems
- Hardened, access-controlled boundaries for vital areas and equipment
- On-site guard force backed by off-site response forces.

These systems provide detection, assessment and delay of, and response against the threats of radiological sabotage and theft. Detection is accomplished using sensors located at the NI-protected area boundary. Assessment of sensor alarms is remotely accomplished using a closed-circuit television (CCTV) system. A delay to intrusion is provided by the perimeter barriers to permit CCTV assessment. A more substantial delay to allow adequate time for an effective response is achieved at the exterior building envelope. Delay times and specific features of the barrier system depend on tradeoffs involving the response force, such as the number of guards and the time it takes for guards to engage the intruders.

The objectives of a safeguards system are to provide:

- A physical protection system that prevents theft of SNM and that prevents sabotage of vital equipment whose failure, destruction, or misuse could result in a radiological hazard to the public
- A materials control and accounting system to maintain records on the amount and location of SNM in the plant and to monitor the SNM to verify that it has not been stolen or diverted.

The safeguards system for FASTER is relatively straightforward, The fuel will come into the facility via transfer casks and will exit the facility using the same transfer casks. Spent fuel will be stored on site as part of the FASTER reactor plant operations. Fuel specimens for examination will be sent to hot cell facilities located on site, but are not part of this study.

3.11 Decommissioning and Waste Generation Aspects

At the end of the FASTER reactor plant life, the fuel will be off-loaded from the core and the dummy 316SS core assemblies will be installed in the core. The cold traps and nuclide traps will be cut/removed from the system and handled separately. The primary and secondary sodium will be removed and processed on site or shipped off site for treatment and disposal. The residual sodium will be deactivated using moist CO_2 and moist N_2 and ultimately flushed with water. Once the sodium in a system has been deactivated, normal decommissioning techniques will be used for decommissioning the plant.

4 FASTER Test Reactor Safety

4.1 FASTER Reactor Safety Basis

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 an 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. This approach is usually identified as “defense in depth.”

The basic principle of “defense in depth” is to provide multiple levels of protection against release of radioactive material. One part of defense-in-depth is physical barriers, like the multiple barriers to release of radioactivity provided by the fuel cladding, the primary coolant system boundary, and the reactor containment building. Generally, active or passive safety systems are provided to protect the physical barriers. These include the reactor shutdown systems and the reactor cooling systems. Inherent characteristics of the design, such as negative reactivity feedback and long flow coast-down, may provide an additional level of protection. Emergency planning provides an additional layer of defense-in-depth, should the other barriers be threatened. However, in all instances, the “defense in depth” strategy depends on the independence of the protective measures, so that no single event can breach more than one protective level.

The FASTER safety design approach implements the “defense in depth” strategy by adopting the traditional three levels of safety. In addition, the FASTER design features have been selected to provide significant safety margin enhancements by inherent passive safety responses to upset conditions and equipment failures.

At the first level, FASTER is designed to operate with a high level of reliability, so that accident initiators are prevented from occurring. The first level of safety is ensured in part by selection of fuel, cladding, coolant, and structural materials that are stable and compatible, and provide large margins between normal operating conditions and limiting failure conditions. An arrangement of components was adopted that allows monitoring, inspection, and testing for performance changes or degradation. Finally, the FASTER design provides for repair and replacement of components as necessary to ensure that safety margins are not degraded.

The selection of liquid sodium coolant and metallic fuel with a pool-type primary system arrangement provides a highly reliable reactor system with large operational safety margins. The coolant thermophysical properties provide superior heat removal and transport characteristics at low operating pressure with a large temperature margin to boiling. The metallic fuel operates at a relatively low temperature, below the coolant boiling point, due to its high thermal conductivity. The pool-type primary system confines all significantly radioactive materials within a single vessel, allows for easy removal and replacement of components as well as shutdown heat removal by natural circulation.

At the second level of safety, FASTER is designed to provide protection in the event of equipment failure or operating error. This level of protection is provided by engineered safety systems for reactor shutdown, reactor heat removal, and emergency power. Each of these safety-grade backup systems functions in the event of failure in the corresponding operating system, and are subjected to continuous monitoring and periodic testing and inspection.

The FASTER design provides an independently powered and instrumented secondary reactor shutdown system that operates automatically to reduce reactor power rapidly in the event that the primary shutdown system fails. For shutdown cooling, the FASTER design includes a safety-grade emergency heat removal system, independent from the normal heat removal system and capable of removing residual decay heat by natural circulation. In addition to the normal offsite power supply, FASTER is equipped with a second independent offsite power connection. The two offsite power connections are supplemented by a safety-grade onsite emergency power supply.

The third level of safety provides additional protection of the public health and safety in an extremely unlikely event that is not expected to occur in the life of the plant, or which was not foreseen at the time the plant was designed and constructed. In the FASTER design, the Level 3 protections for cooling assurance and containment of radioactivity are provided by the reactor guard vessel and the reactor containment building. The reactor guard vessel is designed to hold primary coolant in the event of a leak in the primary coolant system. The reactor guard vessel ensures that the reactor core remains covered with sodium and cooled by the emergency heat removal system, even if the primary reactor vessel fails. If primary coolant leaks and oxidizes in the reactor building air atmosphere, or if failures of the cladding and the primary system barriers lead to release of gaseous fission products, the reactor containment building provides a final low-leakage barrier to release of radioactivity to the environment.

The three levels of safety together are the safety design basis for FASTER. For the purposes of subsequent safety design development, qualification, and documentation, it is customary during the conceptual design phase to identify general design criteria (GDC) that collectively serve as the basis for safety assessment of the design. GDCs for advanced reactors, and SFRs in particular, are currently being developed in a joint DOE and NRC initiative. The FASTER design satisfies all of those currently existing draft SFR GDCs.

The proposed FASTER design is capable of accommodating various beyond-design-basis accident initiators without producing high temperatures and conditions that might lead to a severe accident, such as coolant boiling, cladding failures, or fuel melting. The inherent neutronic, hydraulic, and thermal performance characteristics of the FASTER design provide self-protection in beyond-design-basis sequences to limit accident consequences without activation of engineered systems or operator actions. This characteristic has been termed ‘inherent passive safety.’

The efficacy of such passive safety was demonstrated through two landmark tests conducted on the Experimental Breeder Reactor-II (EBR-II), namely SHRT-45R, a loss-of-flow without scram test, and BOP-302R, a loss-of-heat-sink without scram test. With the automated safety systems disabled, the two most demanding accident initiating events were deliberately induced with the reactor at full power, first one then the other. Each time the reactor simply coasted to a safe low power state without any damage at all to the fuel or any reactor component. These tests proved conclusively that passive safety design is achievable for metallic-fueled fast reactors with sodium cooling.

Within the overall safety framework for FASTER, passive safety serves to provide additional margins for public protection in the event of very low probability events whose frequency of occurrence is lower than the normal threshold for deterministic assessment. The FASTER passive safety performance characteristic ensures that no abnormal radioactivity releases will occur in the event of beyond-design-basis accidents, and that all of the multiple defense-in-depth barriers (fuel cladding, reactor vessel, containment building) for public protection will remain intact, just as for design basis accidents. The passive safety performance of FASTER eliminates the potential for severe accident consequences in very low frequency, beyond-design-basis sequences. Consequently, for FASTER, beyond-design-basis accidents need to be considered only in the context of probabilistic risk assessments.

Security must now be considered as an integral part of the design. The inherent and passive safety features of FASTER offer a high level of protection against malevolent events, as well as against accidents. Since the inherent and passive features do not rely on operator action, external power or functioning of active components, they remove these potential vulnerabilities. In addition, the location of the reactor vessel, the core, and the primary heat transport system below grade within a strong containment structure provides protection against external threats.

4.2 Source Term

Development of a source term and assessment of offsite consequences that can result from radionuclide release are typically the final components of an integrated safety analysis for a reactor. Historically, source term analyses for LWRs have utilized bounding, deterministic assumptions and are based on LWR fuel forms and accident scenarios, as indicated by the formal guidance from the NRC on acceptable source term methodologies in NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” [9] and Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors” [10]. However, due to deviations in the accident bases and properties of the coolant (i.e., compatibility of sodium with metallic fuel, and high solubility of radionuclides within sodium), LWR –derived source terms are considered inappropriate for application in an SFR safety analysis. Additionally, improvements in the knowledge state and computational capabilities have led to renewed interest in mechanistic treatment of source term phenomena.

To that end, ongoing work at Argonne on development of a mechanistic source term (MST) for SFRs [11] has resulted in identification of the significant sources and important transport and retention mechanisms for releases from postulated highly unlikely core damage and fuel handling accidents. The majority of transport and retention mechanisms are well understood, however, uncertainty remains in the understanding of radionuclide formation, transport, and release in-pin as a function of burnup, particularly at high burnup. Under the current DOE sponsored work, it is expected that a MST for SFRs will be available during the conceptual design phase of the FASTER reactor or before.

An SFR MST can be defined as follows [11]:

An SFR MST is the result of an analysis of radionuclide release, in terms of quantities, timing, and other characteristics, resulting from the specific event sequences being evaluated. It is developed using best estimate phenomenological models of the transport of radionuclides from the source through all holdup volumes and barriers, taking into account mitigation features, and finally, into the environs.

For a pool-type SFR, the five key barriers to release include the fuel matrix, cladding, primary sodium (core uncover accidents are highly unlikely in pool-type configurations), primary circuit boundary, and containment. During normal operation, the majority of fission gases will migrate to the fission gas plenum, with a portion of these gases and vapors transporting through bond sodium where there is the opportunity for fission products to dissolve in the bond sodium. If cladding breach occurs, radionuclide gases and vapors in the fission gas plenum and some fraction of the bond sodium will be ejected from the pin. However, many radionuclides will be retained within the fuel matrix, as uranium is an excellent solvent. Once released to the primary coolant, it is expected that only gases and vapors with high vapor pressure and low sodium solubility (e.g., noble gases) will transport directly to the cover gas. The remaining radionuclides will condense and dissolve (gas/vapor) or dissolve and deposit (particulate) in sodium and on structure. Gases and vapors transported in bubbles will be directly released to the cover gas, but dissolved vapors, gases, and particulates must vaporize to enter the cover gas where they may condense and mechanically deposit following vaporization. Airborne radionuclides can then transport through leakage through the seals in the reactor head and into containment. Within containment, phenomena similar to those experienced in the cover gas are expected to occur, although the presence of oxygen and water vapor may induce additional phenomena. Cooler containment temperatures, the likely availability of sodium vapor (which escaped the primary boundary via the reactor head leak pathway), and the small containment design basis leak rate are all expected to enable high retention of radionuclides in containment due to the various condensation, dissolution, and holdup mechanisms.

In the case of an SFR, it is likely that the highest releases will result from spent fuel movement and handling accidents, where the two most significant barriers, primary sodium coolant and cover gas, are not available. Also, in contrast to LWR bypass scenarios, an SFR can experience primary circuit boundary bypass via the cover gas cleanup or primary sodium purification systems. Potential bypass scenarios are addressed by housing the cold traps and sinks associated with these systems in sealed rooms with relatively robust containment capabilities.

4.3 Emergency Decay Heat Removal Capability

The decay heat removal system (DHRS) heat rejection rate capacity was determined by analyzing a series of unprotected beyond-design-basis accident scenarios with various heat rejection capacities. It was assumed that heat rejection through the BOP is unavailable and the control rods fail to scram. Unprotected loss of flow and unprotected loss of heat sinks scenarios, two highly unlikely (less than 10^{-6} per reactor-year) beyond-design-basis accidents, were simulated at beginning and end of equilibrium cycle conditions. As temperatures in the core increase to elevated but safe levels, reactivity feedbacks shut down the fission process, leaving only decay heat. In these scenarios, the DHRS is the sole mechanism for rejecting heat generated in the core.

The necessary DHRS heat rejection rate capacity was determined as the amount of decay heat rejection rate necessary to maintain acceptable safety margins during unprotected transients with one of the three DHRS units assumed to be unavailable, either due to maintenance issues or a failure of the passively opened air dampers to activate. Heat rejection rate capacities between 0.25% and 0.50% of the nominal core power per DHRS unit were examined with 0.25% per unit determined to provide sufficient decay heat rejection to maintain acceptable safety margins. With an assumed failure of one unit, the nominal total decay heat rejection rate is 0.50% of nominal power. Because the safety margins with 0.25% per unit are large enough, the additional cost to increase the size of the DHRS units to provide 0.50% per unit, for example, is not justified.

4.4 FASTER Test Reactor Safety Performance

The normal process of performing safety assessments considers a spectrum of DBAs as tests of the various safety systems. These DBAs generally assume single failures. Accidents within the design basis must be accommodated by the design and shown to present risks to the public that are within regulatory standards. Beyond the design basis, there exists a class of accidents of such low probability that they have been termed “hypothetical.” These events involve multiple failures of safety grade systems, and usually are considered to have a frequency of less than 10^{-6} per reactor-year. Because of the potentially severe consequences of accidents in this class, they have received significant regulatory scrutiny in prior sodium-cooled fast reactor licensing reviews for the purpose of characterizing thermal and structural safety margins beyond the design basis. These accidents are currently referred to as design extension conditions (DECs, between 10^{-6} and 10^{-8} per reactor-year) or severe accidents (SAs, less than 10^{-8} per reactor-year) depending upon the specific frequency.

Three DECs calculated here, each involving failure of both reactor scram systems, have received attention in past licensing safety assessments. In the unprotected loss-of-flow (ULOF) sequence, it is assumed that power is lost to all primary and secondary coolant pumps and the reactor scram systems fail to activate. In the unprotected transient overpower (UTOP) sequence, it is assumed that one or more inserted control rods are withdrawn, and the reactor scram systems fail to operate. In the unprotected loss-of-heat-sink (LOHS) accident, it is assumed that heat removal through the power conversion system is lost, and the reactor scram systems do not activate. Taken collectively, these three accident initiators encompass all the ways that an operating reactor can be perturbed, i.e. by a change in coolant flow, by a change in reactivity, or by a change in coolant inlet temperature. A preliminary safety analysis was performed for FASTER using the systems analysis code SAS4A/SASSYS-1 to assess the reactor’s safety performance during the transients. [12] A series of ULOF, ULOHS, and UTOP transients were simulated at both beginning of cycle (BOC) and end of cycle (EOC) conditions, except for the UTOP, which was only simulated at BOC because the control rods are already withdrawn at end of cycle.

Maintaining coolable geometry within the core is the primary consideration when evaluating events with such low frequencies. For example, fuel melting can be tolerated at the center of the fuel pin as long as molten fuel is not breaching the cladding and entering the coolant channel. However, the favorable features of FASTER, and SFRs in general (i.e., strong inherent reactivity feedbacks, the excellent heat transfer capabilities of sodium, and the large heat sink of multiple sodium regions inside of the pool-type reactor vessel), lead to such large safety margins that the transient scenarios can be evaluated under higher scrutiny. For this analysis, the results of a transient are considered acceptable when reasonably large margins to sodium boiling and fuel melting are maintained.

Best estimate simulations of ULOF, UTOP, and 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 circuit. 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. [13]

Results from the ULOF, ULOHS, and UTOP transient simulations are summarized in the tables below. Adequate safety margins are maintained during each of the analyzed transients. In the UTOP scenario, a single control rod is assumed to be unintentionally withdrawn until it reaches its rod stop, 6 cm above the critical insertion depth, limiting the reactivity insertion to 0.5 $. The UTOP scenario attains the highest fuel temperatures of all of the transients, with a peak fuel temperature of 889°C; a fuel melting margin of 182°C is maintained.

Table 22 - 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 23 - 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

In the ULOHS scenario, a loss of heat rejection through the IHX leads to elevated cold pool and core inlet temperatures. Elevated temperatures in the core induce sufficient negative reactivity to shut down the fission process. As the transient continues, decay heat production gradually decreases until it is equal to the decay heat rejection capacity of the DHRS. The reactor vessel temperature levels off at 562°C, maintaining a margin of 170°C for ensuring structural stability of the reactor vessel.

In an ULOF accident, power is simultaneously lost to all primary and secondary sodium pumps. The ULOF transient is driven by an increasing power-to-flow ratio. After the primary pumps trip, the sodium flow rate begins to coast down with a flow halving time of approximately 10 seconds. Elevated temperatures at the top of the core induce a negative reactivity feedback from radial core expansion, causing the fission power to decrease.

Within the first minute, the rate of the flow decrease slows down as natural circulation is established and the power-to-flow ratio peaks at 1.9 of the nominal value. The peak cladding temperature increases from 568°C to 720°C but it only remains above 700°C for less than one minute. The peak coolant temperature follows a similar trend, increasing from 549°C to 719°C, maintaining a sufficiently large 234°C sodium boiling margin. Because power begins decreasing fairly quickly at the start of the transient, the effect on the peak fuel temperature is even smaller, increasing from 712°C to only 741°C.

After the flow rate starts to level off, power continues to decrease causing temperatures in the core to also decrease. Approximately four minutes into the transient, peak temperatures in the core have dropped below their nominal values and the reactivity feedbacks from Doppler, axial fuel expansion, and radial core expansion are slightly positive. The control rod driveline, however, had been heating up due to the hotter core outlet temperatures, inducing a negative reactivity feedback as the driveline expands thermally and inserts the control rods further into the core. The negative reactivity from control rod driveline expansion more than compensates for the other feedbacks maintaining a sufficiently negative net reactivity to prevent a self-sustaining chain reaction in the core.

In the longer term, decay heat continues to decrease. A small amount of fission power (less than 0.25%) is maintained. Although the cold pool temperature has increased nearly 100°C, the hot pool temperature increase was much smaller, approximately 30°C. With elevated cold pool temperatures, the heat rejection rate capacity of the

DHRS has increased from 0.5% to approximately 0.7%. By 17 hours, the fission and decay heat production in the core are matched by the DHRS decay heat rejection rate capacity and temperatures throughout the primary system gradually begin to decrease. The normalized power-to-flow and peak in-core temperatures for the ULOF transient are shown in the figures below.

The AOOs, DBAs, DECs, and SAs for FASTER include those that have been previously identified for metallic-fueled SFRs such as for PRISM and SAFR together with those particular to specific design features of FASTER that includes DBAs involving the closed loop systems. A category for SAs is included in the main document to indicate the proper characterization of an unprotected station blackout accident.

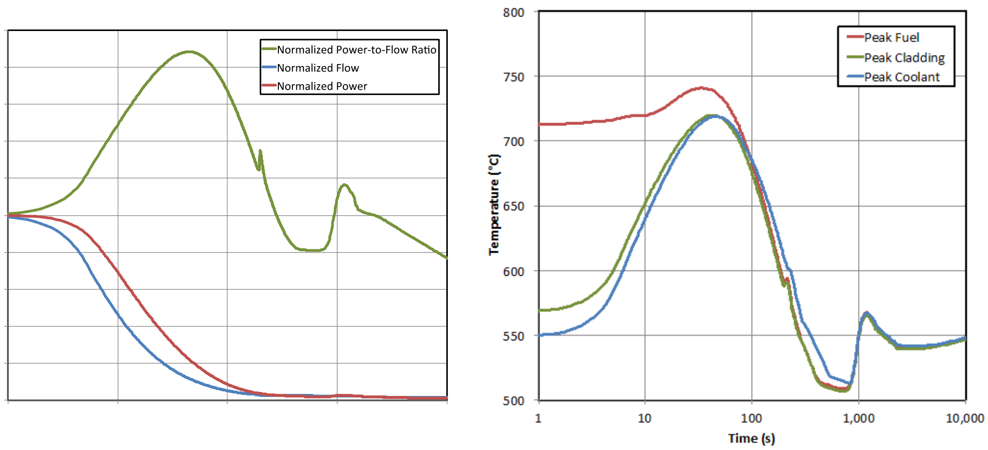

Figure 10 – (Left) Normalized Power, Flow, and Power-to-Flow Ratio, (Right) Peak In-Core Temperatures

5 Technology Readiness of Test Reactor Concept

The FASTER reactor plant, with its sodium coolant, pool plant geometry, and metallic fuel can trace its heritage to the beginning of fast reactor technology with the Experimental Breeder Reactor-II. Base sodium-cooled fast reactor technology has been utilized in the FASTER reactor concept to increase the technology readiness of the system and components.

Table 24 – TRL Evaluation of FASTER Reactor Plant Systems and Components

Component	TRL	Risk Description
Driver Fuel	TRL8	U-19Pu-6Zr - Reestablishment of domestic fabrication capability or foreign supply required. The preferred option would be to reestablish the capability of fabricating this fuel at a collocated facility with the FASTER reactor plant. Assumes that prior irradiation testing data is suitable for licensing.
Reflector Assemblies	TRL8	Reflector assemblies made from metal pins or blocks are conventional sodium fast reactor technology and thus achieve a high TRL level.
Shield Assemblies	TRL8	Shield assemblies made from clad boron carbide pins are conventional sodium fast reactor technology and thus achieve a high TRL level.
Control Rod Assemblies	TRL8	The FASTER control rod assemblies are constructed of boron carbide rods clad in HT-9. This technology is conventional sodium fast reactor technology and thus achieves a high TRL level.
Core Structural Materials – Fast Zone	TRL8	Demonstrated at full scale. Burnup is very low so there will be zero issue using HT-9 as the core structural materials in the fast neutron spectrum zone.
Core Structural Materials – Thermal Zone	TRL5	Literature review will need to be performed for structural materials in thermal zone. Zircaloy is compatible with sodium but is typically not used as core structural materials for fast reactor cores (cladding and ducting). Zircaloy is also an oxygen getter – so time in service will need to be understood and evaluated, but is not expected to be an issue. The zircaloy will also clad beryllium with the beryllium being a moderator.
Coolant Control Technology	TRL9	Coolant control technology using cold traps and nuclide traps is conventional proven sodium coolant control technology and has been demonstrated and operated at full scale. Coolant control technology used in EBR-II, FFTF, and developed for CRBRP can be adopted for use in the FASTER reactor plant.
Cover Gas Technology	TRL9	Argon cover gas is conventional proven technology and this will be used in the FASTER reactor plant systems. Nitrogen may also be used in some locations where needed for space inerting.
Cover Gas Cleanup Technology	TRL8-9	The cover gas cleanup technology will be initially based upon that used at EBR-II and FFTF and proposed for CRBRP with updates to technologies that may be used today for cleaning radioactively contaminated argon cover gas.
Gas Seal Technology	TRL7	Gas seal technology is known, but will need to be updated/evaluated from the 1970-1980's technology used in FFTF and EBR-II to determine whether advancements in elastomers and other gas seal technologies are suitable and perform better than the technologies used in the past.
Primary System Configuration	TRL7	Several components must be adapted from previous fast reactor designs, however, a pool plant geometry has proven to be most cost effective. However, no pool plant geometry has included 3 closed loops and multiple instrumented test assembly locations that stay in the reactor greater than one cycle.
Reactor Vessel and Structures	TRL8	For pool configuration, certain features such as the redan and reactor enclosure must be adapted for the particular geometry
Primary and Secondary Pumps	TRL8	The primary and secondary pumps are based upon mechanical centrifugal pump technology which has been deployed successfully in EBR-II, FFTF, and developed and tested for CRBRP. The exact pump will have characteristics that are specific for the FASTER reactor plant, but is expected to be smaller in capacity to the FFTF reactor plant.
Intermediate Heat exchangers	TRL8	The (4) IHX are tube-and-shell heat exchangers and are conventional sodium-to-sodium heat exchangers with a high TRL level. 9Cr-1Mo heat exchangers have been developed and installed in the Indian reactor. 9Cr-1Mo is a code qualified materials.

Component	TRL	Risk Description
Direct Reactor Auxiliary Cooling System	TRL8	DRACS emergency decay heat removal systems have been used in EBR-II and other sodium cooled fast reactors.
Direct Reactor Heat Exchangers	TRL8	The (3) DRACS heat exchangers are tube-and-shell heat exchangers and are based on conventional sodium-to-sodium heat exchangers with a high TRL level. These heat exchangers will have to be scaled appropriately for the size requirements for the FASTER reactor plant, but besides this, that is the only issue.
Air dump heat exchangers	TRL8	The (3) DRACS system each have one sodium-to-air heat exchanger which will be based upon the technology developed for FFTF, EBR-II, and CRBRP.
Steam Generator	TRL7	The steam generator adopted for the FASTER reactor plant is the helical coil steam generator (HSCG). There are two HCSGs that each provide 150MWth of superheated steam for the steam turbine. The HCSG is approximately twice the size of the HSCG that was developed by Babcock and Wilcox and tested at ETEC. That HCSG used 2.25Cr-1Mo steel for the tubes. The FASTER HCSG uses 9Cr-1Mo steel which is an acceptable alternative for heat exchanger materials. Superphenix installed a HCSG that was larger than needed for the FASTER reactor plant, but was made from Alloy600.
Sodium Water Reaction Protection System	TRL7	The technology for accommodating the sodium-water reaction resulting from a tube leak in a sodium-heated steam generator was developed in the past liquid metal technology development programs and tested at the Energy Technology Engineering Center.
Sodium Fire Mitigation System	TRL8	The technology for mitigation of sodium fires has been developed in past liquid metal technology programs.
Balance of Plant	TRL8	Superheated steam BOP is a conventional technology used with sodium-cooled fast reactor technologies around the world.
Containment	TRL8	Steel-reinforced concrete containment technology has been demonstrated at full scale both domestically and internationally.
Seismic Restraints	TRL8	Seismic restraints for piping and other components are expected to be conventional reactor technology.
Reactor Instrumentation and Control	TRL7	Demonstrated at full scale. The reactor project expects to adopt digital controls as appropriate.
Primary Control Rod Drive System	TRL7 TRL4 – Rod Stops for UTOP	The primary control rod drive system will be similar to the system for the AFR-100 plant which is an adaptation of the PCRDS from FFTF and CRBR. Interface between the PCRDS and the three instrumentation trees will need to be determined, but will be similar to technology developed for FFTF. Control rod stops will need to be implemented into the PCRDS system.
Secondary Control Rod Drive System	TRL6	The secondary control rod drive system is similar to the system for AFR-100 plant and the SCRDS for CRBR and ALMR. Engineering will have to be performed to ensure that the SCRDS will interface successfully with the three instrumentation trees in the FASTER reactor vessel.
Fuel Handling	TRL5	The starting point for the FASTER in-vessel refueling machines will be the machine developed for the AFR-100 plant. Mechanisms related to the in-vessel refueling machine will need to be evaluated to ensure that the machine will function as designed. If FFTF IVHM system was adopted a higher TRL would be assigned.
Maintenance and Inspection	TRL5	New techniques for in-service inspection, especially under sodium inspection, needs to be demonstrated. Technologies developed for FFTF and EBR-II need to be redeveloped.
FASTER Irradiation Capability		

Component	TRL	Risk Description
Closed Loop Technology	TRL8 for sodium TRL3-6 for other fast reactor coolants TRL3-6 for the thermal reactor coolants.	The starting point for the closed loop technology will be the designs that were created by Westinghouse for the FFTF reactor. There are three closed loops in the FASTER reactor plant – two fast closed loops and one thermal closed loop. The power/heat removal rate capability of the sodium closed loop will be identical to the FFTF reactor closed loop in-reactor assembly and primary module. It is expected that the two fast closed loops will be very similar in technology design to FFTF and thus will have a very high TRL. The thermal spectrum close loop will have a lower TRL mainly to reflect the design of the module that supplies the primary (in-loop) coolant for the thermal closed loop.
Instrumented test assembly technology	TRL7	The starting point for the instrumented test assembly technology will be the technology used in EBR-II and FFTF. The technology will be updated with modern sensors for temperature, flow, and other parameters as appropriate, but the technological foundation will be EBR-II, FFTF, and CRBR technologies and will evolve from there. Test
Non-instrumented test assembly technology	TRL8	Non-instrumented test assembly technology are configured identical to a fuel, reflector, and shield test assemblies. The technology to include test pins in fueled core assemblies is known and thus they are evaluated with a high TRL level.
Licensing		Technical criteria for advanced reactor licensing needs to be clarified and the regulatory structure re-established. It is expected that the selected ATDR reactor will be licensed by the U.S. NRC.
Fast Reactor Manufacturing Infrastructure		Infrastructure for manufacturing fast reactor components will need to be reestablished domestically if this is important to DOE, otherwise, there are international sources of liquid metal technology components such as pumps, valves, heat exchangers, etc.

6 Test Reactor Licensing, Development and Deployment Plans

For the PRISM and SAFR designs, which were also SFRs utilizing metallic fuel with strong inherent reactivity feedbacks and passive safety, Preliminary Safety Information Documents (PSIDs) were prepared by each vendor and submitted to the U.S. NRC, and preapplication interactions between each vendor and NRC staff were conducted. For PRISM, the NRC issued a Preapplication Safety Evaluation Report (PSER), NUREG-1368, concluding that “no obvious impediments to licensing the PRISM design had been identified.” Work on SAFR was discontinued by DOE before the NRC evaluation was completed but DOE requested that NRC document what they had done. The resulting SAFR PSER, NUREG-1369, concluded that the SAFR design had the potential for a level of safety at least equivalent to then current LWR plants. In 2010, General Electric Hitachi provided the NRC with a draft licensing strategy for the PRISM design for informal NRC consideration.

FASTER will be licensed under 10 CFR Part 50 as a testing facility that also produces electricity onto an electrical grid. The Preliminary Safety Analysis Report (PSAR) must include the principal design criteria for the facility. 10 CFR Part 50 Appendix A, “General Design Criteria for Nuclear Power Plants,” establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the NRC and provides guidance to applicants for construction permits in establishing principal design criteria for other types of nuclear power units. Historically, specific SFR principal design criteria were developed for CRBR, PRISM, and SAFR, instead of directly utilizing the General Design Criteria (GDC) from 10 CFR Part 50 Appendix A. A set of draft SFR design criteria has been developed under a joint initiative between DOE and the NRC. The criteria include five new criteria specific to SFRs. A report containing the SFR criteria was prepared by DOE and transmitted to the NRC. The NRC internally reviewed the criteria and is expected to soon issue a report for public comment. It is expected that this will be followed by NRC guidance including SFR principal design criteria. The FASTER design satisfies the current draft SFR principal design criteria and will satisfy the final criteria included in the NRC guidance.

Under the DOE-funded Regulatory Technology Development Plan, work has been launched to develop the SFR safety analysis codes and methods required for use in a licensing framework, and to identify the Quality Assurance (QA) requirements for licensing applications. A report, ANL-ART-37, was recently prepared that identifies the types of safety analysis computer codes that may be required for licensing of metallic-fueled SFRs and assesses the current status of existing relevant safety analysis codes including existing QA practices.

Pre-conceptual, conceptual, and final design will be carried out consistent with the DOE CD-0, CD-1, etc. process. A licensing strategy and schedule will be developed for FASTER. Preapplication meetings will be held with the NRC. As required, a PSID will be prepared. Interactions will be conducted with the NRC to pursue applications for a construction permit and operating license. A Preliminary Safety Analysis Report (PSAR) will be prepared as part of an application for a construction permit. A Final Safety Analysis Report (FSAR) will be prepared as part of an application for an operating license. Testing of the particular metallic fuel type used in FASTER shall be carried out to obtain the data needed to enable the use of this fuel. The FASTER design utilizes structural materials that are currently qualified for use by the appropriate ASME codes, with the exception of the Hastelloy-N alloy envisioned for use in closed loop systems for liquid salt. If a need for closed loop testing with liquid salt continues, then effort will be devoted to developing and submitting a code case for Hastelloy-N. Development of required safety analysis computer codes meeting QA requirements for licensing use shall be completed, and the codes will be used in preparation of the PSAR and FSAR. The FASTER design shall incorporate instrumentation to detect postulated sodium leakages from sodium piping and components consistent with ASME Boiler and Pressure Vessel Code requirements for Nuclear Power Plant components.

The licensing strategy will include a strategy for testing of components for FASTER including fuel assemblies, control rods and control rod drive systems, fuel handling systems, steam generators, intermediate heat exchangers, sodium pumps, as well as instrumentation for use in FASTER. Testing will be carried out, where feasible, in existing testing facilities, as well as new testing facilities that will be identified and assembled. Appropriate QA requirements for test data will be identified and followed.

Following the granting of an operating license, fabrication of the first core and loading of the first core, then criticality, low power testing and ascent to full power shall be carried out. During the ascent to full power, transient testing shall be carried out to determine the actual FASTER reactivity feedback behavior and to verify that it meets the requirements for inherent passive safety and inherent passive shutdown.

7 Economics and Schedule

7.1 Schedule

It is expected that the following notional schedule will be used for the design and construction of FASTER.

- 1 year for conceptual design
- 2 years for preliminary design
- 3-5 years for detailed design, licensing, and long-lead item fabrication
- <5 years for construction and final licensing

Total of 11 to 13 years for the FASTER reactor project from CD-0 with the assumption that there are no constraints on cash flow for the project and that licensing will not be the limiting factor. The schedule that includes the remaining research and development to get the FASTER power plant up and running will extend the schedule to 15 years or less with a concurrent R&D program. This schedule is consistent with a small modular advanced demo plant such as PRISM Mod A which has a power level slightly greater than FASTER. The sodium closed loop technology development will run in parallel with the schedule to construct and startup FASTER and use as its starting base, the sodium closed loop technology developed for FFTF. The non-sodium closed loop technology development will be independent of the main FASTER power plant construction, licensing, and operations and will not impact the schedule to startup FASTER.

7.2 Economics

It is estimated that 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 300MWth of heat to the atmosphere, then the TPC cost (design and construction) would be significantly less than $2.5B because the project would save both the cost for design, equipment, and construction for the steam plant (everything from the steam generators through the steam plant and electrical yard). In addition, the FASTER design team did not take any explicit credit during the cost estimate for prior design work and technology development work that may have been performed that would relate to the FASTER reactor plant. So, as the FASTER test reactor moves forward, more detailed cost estimates will better refine these cost figures.

The annual FASTER reactor plant operating costs in power generation mode are estimated to be less than $100M. The FASTER reactor plant annual operating costs <u>including irradiation operations</u> are expected to be less than $150M (using FFTF as the high end basis) [14]. All estimates are in 2016 dollars. The replacement fuel is estimated to cost about $20M/year.

The FASTER reactor will provide 120MW_e to the electrical grid at the location of installation. 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.

The cost and schedule estimates are based upon the best available information from the ALMR program, the FFTF project, and current consumer price index and construction cost index escalation factors averaged to 2016.

Operating and Maintenance Costs - The cost estimate to operate and maintain the FASTER reactor is divided into two estimated costs. One to operate the reactor and provide power and maintain the reactor <u>and</u> one that includes those costs and the costs to run experiments in FASTER. The costs to run experiments in FASTER include the costs related to operating and maintaining any experiments, secondary loops or other aspects of the facility such as hot cell facilities that are not related to the functions of the FASTER reactor which provide the experiment conditions (i.e., flux, temperature, etc.) and produce electrical power. Given the purpose of the FASTER reactor as a test reactor, it will be capable of supporting multiple large experiment programs simultaneously which may have significant costs associated with them. These costs are not included as part of the FASTER reactor operation and maintenance (O&M) costs. Those costs are assumed to be part of the larger annual operating costs.

Estimating the O&M costs is difficult because there are no directly comparable sources of data. The estimates must be extrapolated from a variety of sources, each of which has obvious issues with extrapolation to the FASTER reactor. There are three categories of cost. The smaller of the two are consumables (i.e., materials, supplies, etc.) and mandated fees (i.e., NRC fees, insurance, etc.). The largest cost is personnel.

The cost of consumables and fees is expected to be no more than $15M to $20M per year. These estimates are based on the costs associated with power reactors. The fees are not assumed to scale. The consumables are expected to scale, but far from linearly resulting in only modest reductions relative to a larger commercial-scale reactor.

The cost of personnel is based on the number of personnel required and the average cost of those personnel. These estimates are generally broken down to onsite and offsite staff. The onsite staff consisting of the technicians, maintenance, training, and other personnel involved primarily in the direct operation and maintenance of the plant. The offsite staff, which is not necessarily physically offsite, mostly provides the technical support and is a much smaller contributor to cost than the onsite staff.

The nature of the work done by these two manpower groups results in the average cost of onsite staff generally being significantly lower than the average cost of offsite staff. The specific mix for a given power plant will also impact this. The average salaries are expected to average around $100k per year for onsite staff and around $200k per year for offsite staff. For each, there will be a multiplier on these costs for benefits and other related staffing costs of no more than two times and more likely around 1.6 to 1.8.

The AP1000, a large commercial LWR is estimated to require an onsite staff of between 386 and 502, depending on whether it is located on an existing operational nuclear site or on its own. The commercial-scale S-PRISM is expected to have a total staff of roughly 650 for a twin-unit power plant. There are a lot of data on the variation in O&M costs for LWR stations as a function of size and units at the commercial scale. This information is used to identify where there will be cost savings for a smaller unit (e.g., annualized maintenance) and where there will not (e.g., one plant manager regardless). By combining this information and extrapolating it to a much reduced size, it suggests that a reactor of this size could be operated and maintained by a staff of significantly less than 200. To be conservative, since this is an extrapolation, the upper bound was assumed to be a total staff of 250 split into 210 onsite and 40 offsite personnel. This assumes a standalone site that receives no cost benefit from being on an existing nuclear site which will share a lot of resources, particularly related to security. This results in personnel costs in the $40M (staff of 200) to $60M (staff of 250) per year range.

The decommissioning and decontamination (D&D) costs are typically covered by a sinking fund payment. The annual cost of payments to such a fund will be a small fraction of the annual O&M cost assuming that the facility operates for decades. As a result, the D&D funds are within the overall uncertainty and do not impact the estimate at the current level of design detail.

The total O&M costs are expected to be in the range of $55M to $80M per year and certainly could be significantly lower. There may be significant additional costs associated with experiments conducted at this facility that are not included.

Fuel Cost Estimate - The cost associated with fueling the reactor is even more challenging to estimate than the O&M costs. The fuel cost estimate assumes that a fixed unit cost ($ per kg of heavy metal) is charged to the FASTER reactor that is equivalent to the cost of producing the fuel. This unit or levelized cost as it is typical referred to includes the capital investment, O&M, and cost of materials (e.g., plutonium, cladding, etc.). Like most nuclear facilities, the capital investment will be the largest contributor if a new facility must be built to supply the fuel. The other major contributors to cost are the cost of the fissile material and the O&M of the facility. It is assumed that the plutonium will be provided at no cost. The other materials (e.g., cladding and other assembly hardware) tend to be a relatively small cost contributor, although this needs to ultimately be confirmed since producing unique materials in small quantities can result in far higher unit costs.

The cost of manufacturing plutonium fuels on a commercial scale is estimated to be in the range of $2,500 to $7,000 per kg of heavy metal. This includes all costs except the cost of the heavy metal itself (i.e., cladding, assembly hardware, etc.), which is assumed to be negligible (Pu is free and DU will be at very small fraction at most). The assumption is that the manufacture of the FASTER reactor fuel will not require the construction of a standalone fabrication facility like the MOX plant at Savannah River Site, but will be done by upgrading an existing national laboratory facility to produce the relatively small quantities (~60 assemblies per year). There will be significant investment into this facility, but it was assumed that this would be bounded by the upper cost estimate of $7,000 per kg, which results in a cost estimate of approximately $300k per assembly. This would result in the fuel costs being no more than $20M per year. This equivalent annual cost implies an upfront investment in fuel fabrication capabilities in the $100M to $200M range.

Annual Operational Cost Estimate - When the O&M and fuel costs are added together, this results in an annual cost of no more than $100M per year and likely significantly less than that.

The annual costs of FFTF escalated for inflation would be on the order of $150M per year, but this seems to include all costs associated with the experimental programs that were utilizing FFTF and not just the operational cost of the reactor. Given the labor intense nature of the FFTF experience and the high costs of irradiation experiments, this value seems consistent with the expectation that the annual operational costs of the FASTER reactor will be between less than $150M per year.

8 Works Cited

[1] M. A. Smith, C. Adams, W. S. Yang, and E. E. Lewis, "VARI3D & PERSENT: Perturbation and Sensitivity Analysis," Argonne National Laboratory, ANL/NE-13/8, 2013.

[2] M. A. Smith, E. E. Lewis, and E. R. Shemon, "DIF3D-VARIANT 11.0, A Decade of Updates," Argonne National Laboratory, ANL/NE-14/1, 2014.

[3] D. C. Wade and E. K. Fujita, "Trends Versus Reactor Size of Passive Reactivity Shutdown and Control Performance," *Nuclear Science and Engineering*, vol. 103, pp. 182-195, 1989.

[4] K. L. Basehore and N. E. Todreas, "SUPERENERGY-2: A Multiassembly Steady-State Computer Code for LMFBR Core Thermal-Hydraulic Analysis," Pacific Northwest National Laboratory, PNL-3379, 1980.

[5] GateCycle. (2016) GE Power & Water GateCycle Software. [Online]. https://getotalplant.com/GateCycle/docs/GateCycle/index.html

[6] Hanford Engineering Development Laboratory, "A Summary Description of the Fast Flux Test Facility," HEDL-400, 1980.

[7] D.J. Oakley, "Closed Loop In-Reactor Assembly (CLIRA) A Fast Flux Test Facility Vehicle, HEDL-SA-1605FP," in *American Society of Mechanical Engineers Winter Annual Meeting*, San Francisco, 1978.

[8] Idaho National Laboratory, "FY 2009 Advanced Test Reactor National Scientific User Facility Users' Guide," INL/EXT-08-14709, 2009.

[9] U.S. Nuclear Regulatory Commission, "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, 1995.

[10] U.S. Nuclear Regulatory Commission, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," RG 1.183, 2000.

[11] D. Grabaskas, A. J. Brunett, M. Bucknor, J. Sienicki, and T. Sofu, "Regulatory Technology Development Plan - Sodium Fast Reactor: Mechanistic Source Term Development," ANL-ART-3, Argonne National Laboratory, 2015.

[12] Editor: Fanning, T.H., "The SAS4A/SASSYS-1 Safety Analysis Code System: User's Guide," ANL/NE-12/4, Argonne National Laboratory, 2012.

[13] U.S. Department of Energy, "SAFR - Sodium Advanced Fast Reactor," Preliminary Safety Information Document AI-DOE-13527, 1985.

[14] U.S. General Accounting Office, "Fast Flux Test Facility on Standby, Awaiting DOE Decision on Fugure Missions," GAO/RCED-92-121FS, 1992.

FASTER Reactor Plant

Appendix A: Self-Assessment against Test Reactor Metrics

This section provides an assessment of the proposed test reactor, FASTER, against the goals, criteria and metrics established as part of the ATDR study framework. The scores shown here have been determined by the design team that developed the FASTER point design and are briefly justified. More details have been provided in the main body of the summary point design report. In the following tables, the self-assessed score for FASTER is highlighted in green.

Goal T1. Test Reactor provides irradiation services for a variety of reactor and fuel technology options.

Criterion T1.1. Irradiation Conditions

Metric T1.1.1. Fast Flux conditions (>0.1 MeV)

Metric	>5 x 10^{15} n/cm^2-s fast (>0.1 MeV)	5x10^{14} to 5 x 10^{15} n/cm^2-s fast (>0.1 Mev)	<5 x10^{14} fast (>0.1 MeV)
Score	3	2	1

<u>Justification:</u> The peak fast flux achieved in the core has been determined to be 5.2x10^{15} n/cm^2-s in the fuel assemblies surrounding the test assembly located at the very center of the active core region. The radially averaged peak fast flux in this test assembly is larger than 5.0x10^{15} n/cm^2-s.

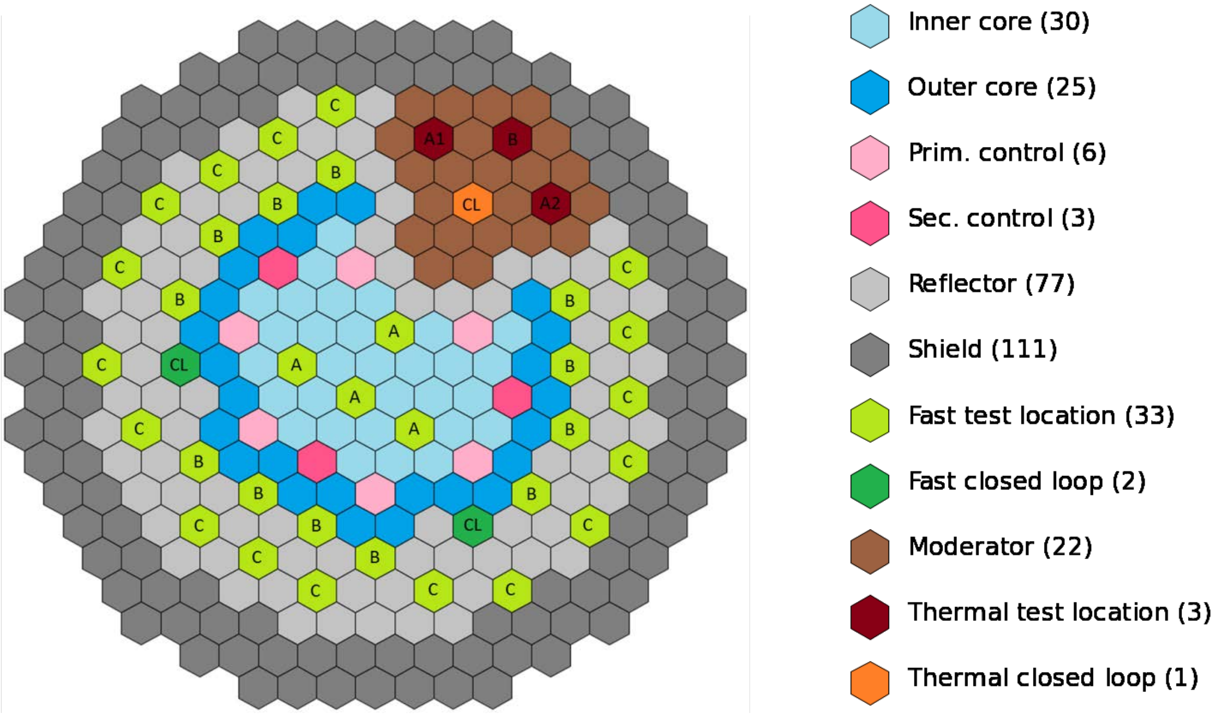

Figure 11 – FASTER Core Design with Test Locations Indicated

Table 25 – Summary of FASTER 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

Metric T1.1.2. Thermal Flux conditions (<0.625 eV)

Metric	>5x10^{14} n/cm^2-s thermal	1 to 5x10^{14} n/cm^2-s thermal	<1x10^{14} thermal
Score	3	2	1

Justification: Thermal flux is achieved through the use of beryllium-based moderator assemblies, making use of the neutrons leaking horizontally from the active core region. The peak thermal flux obtained in the thermal closed loop has been calculated to be 5.8x10^{14} n/cm^2-s, when using an energy threshold of <0.1 eV. With the energy threshold of <0.625 eV, the thermal flux will be significantly larger than the value reported here.

Table 26 – Summary of FASTER Thermal Flux Conditions in the Test Assemblies

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

Metric T1.1.3 Irradiation volumes and length for largest test location

Metric	Volume > 10 liters Length > 2 meter	5 to 10 liters volume 0.5 to 2 meter length	Volume < 5 liters Length < 0.5 m
Score	3	2	1

Justification: The total available length in any given test location is 277 cm (about the length of an assembly). Depending on the test assembly design, which will vary based on the irradiation experiment considered, the usable length inside a test assembly will be slightly less but likely >200 cm. Assuming a 200 cm effective length, this corresponds to a volume of about 24.4 liters. If subtracting the volume "lost" to the inter-assembly gap and duct thickness, the test volume is reduced to 20.8 liters. This is the volume for a single test location.

Metric T1.1.4. Maximum sustainable time at power, to provide a time-at-power for a single irradiation (i.e. cycle length)

Metric	> 90 days	45 to 90 days	< 45 days
Score	3	2	1

Justification: The FASTER core has been designed for a nominal cycle length of 100 effective full power days. Given that the required plutonium weight fraction is about 19.4%, it is possible to go to an even longer cycle length by slightly increasing the plutonium content of the fuel (still keeping it <20%).

Metric T1.1.5. Provisions for testing prototypic and bounding conditions (Temperature, Coolant, Chemistry)

Metric	Prototypic and bounding for different reactor coolants	Prototypic and bounding for base test reactor coolant	Not prototypic or bounding
Score	9	5	1

Justification: The FASTER closed loop in-reactor assemblies can be designed for coolants and temperature and pressure conditions exceeding those of representative reactor types (e.g., SFR, LFR, HTGR, Pebble Bed FHR, AHTR, PWR, Pressurized water-cooled and water-moderated research and test reactors) and provide ample space interior to the double-walled pressure tube for a flow tube to separate downward and upward flows and a core simulation inside of the flowtube. For example, a design temperature of 760 °C and design pressure of 7.82 MPa can be accommodated with a double-walled pressure tube exceeding the 750 °C core outlet temperature and 6.39 MPa pressure of a helium-cooled HTGR. For a pressurized water closed loop, a temperature of 650 °C and design pressure of 17.05 MPa can be accommodated with double-walled pressure tubes exceeding the core outlet temperature and primary system pressure of a PWR. The FASTER design team ranks the metric at a 9 for both prototypic and bounding for different reactor coolants.

Criterion T1.2. Support diverse irradiation testing configurations concurrently (accommodate various sizes and tailor irradiation parameters to wide group of simultaneous users)

Metric T1.2.1. Number of test zones

Metric	> 25 locations	10 to 25 locations	< 10 locations
Score	3	2	1

Justification: In the proposed configuration, the FASTER core offers 33 test locations exposed to a fast flux and three test locations exposed to a thermal flux. Given the nature of SFRs, the test locations in the reflector region are highly adjustable and their number can be significantly increased (or reduced), if needed.

Metric T1.2.2. Number and type of distinct irradiation test loops each with a different cooling system independent of the primary reactor coolant

Metric	3 or more	1 or 2	None
Score	3	2	1

Justification: In the proposed configuration, the FASTER core offers two closed loops exposed to a fast flux and one closed loop exposed to a thermal flux. Each of the three closed loops are placed at a 120° angle from one another. Almost any kind of coolant can be used in a given closed loop (except moderating coolants in the fast flux closed loop). Once the reactor is built, the coolant type can be changed by replacing the replaceable closed loop primary coolant module.

Metric T1.2.3. Ability to insert/retrieve irradiation specimen while staying at power

Metric	At power (e.g. rabbit)	Limited handling capability	Only at shutdown
Score	3	2	1

Justification: The rabbit tubes provide the capacity to insert and retrieve irradiation specimens while at power. The rabbit tubes can be located at the instrumented test assembly locations. In addition, the closed loop test assemblies can have the capability of being configured for insertion and removal of irradiation specimens while the reactor is at power. Closed loop test assemblies used in this configuration may be distinct from a closed loop whose main mission is to provide flowing coolant at prototypic conditions.

Goal T2. Test Reactor will be built and operated reliably and in a sustainable cost-effective manner. (Need to be able to justify initial and long-term expense)

Criterion T2.1. Project Costs and Schedule (including design, licensing, R&D, construction and contingency that reflects technical maturity of the concept)

Metric T2.1.1. Project cost

Metric	< $2.5 B	$2.5 – 4 B	> $4.0 B
Score	3	2	1

Justification: It is estimated that 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 300MWth of heat to the atmosphere, then the TPC cost (design and construction) would be significantly less than $2.5B because the project would save both the cost for design, equipment, and construction for the steam plant (everything from the steam generators through the steam plant and electrical yard). In addition, the FASTER design team did not take any explicit credit during the cost estimate for prior design work and technology development work that may have been performed that would relate to the FASTER reactor plant. So, as the FASTER test reactor moves forward, more detailed cost estimates will better refine these cost figures.

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 will provide 120MW_e to the electrical grid at the location of installation. 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.

The cost and schedule estimates are based upon the best available information from the ALMR program, the FFTF information, and current consumer price index and construction cost index escalation factors averaged to 2016.

Metric T2.1.2. Project Schedule - The time from today to first operation

Metric	< 10 years	10-15 years	>15 years
Score	9	5	1

Justification: FASTER is a small modular reactor (SMR) and shares the benefits that have been identified for SMRs including a shorter construction time relative to economy-of-scale NPPs. It is assumed that the pace of construction will not be limited by availability of funding or unanticipated licensing delays.

The overall schedule is expected to be the following after CD-0:

- 1 year for conceptual design
- 2 years for preliminary design
- 3-5 years for detailed design and procurement of long-lead items and site preparation
- < 5 years for construction and final licensing activities

The pace of construction will not be impacted by design and installation of closed loop technologies or instrumented assemblies which can be performed in parallel with the 11-13 year design and construction schedule. The schedule assumes that cash flow and licensing do not control the schedule. The metric has changed from the original metric. It is expected that from today, early 2016, that the schedule will be approximately 15 years or less due to the addition of R&D time that may or may not be 100% concurrent with the design, anticipated CD-0 in 2018 and first design funds arriving in that time frame to initiate the conceptual design. Furthermore, a highly optimized schedule, taking into account a lot of the design information created for EBR-II and FFTF, could result in operations of the reactor in a timeframe that is significantly less than 15 years from today. This schedule estimate is consistent with a demo reactor plant schedule, such as GE PRISM Mod A.

Criterion T2.2. Operational Costs and Schedule (including contingency that reflects technical maturity of the concept)

Metric T2.2.1. Annual operating costs

Metric	< $100 M/yr	$100-150 M/yr	> $150 M/yr
Score	3	2	1

Justification: The total O&M costs for operating the reactor are expected to be in the range of $55M to $80M per year and certainly could be significantly lower. There will be significant additional costs associated with experiments conducted at this facility that are not included. If the experiment costs are includes in the cost estimate, then it is expected that the total cost (annual reactor operating costs + experiment operations costs) will be between $100 and $150M/year and this is the value that we are assuming.

Criterion T2.3. Reliability of operations

Metric T2.3.1. Availability factor

Metric	>80%	60-80%	<60%
Score	3	2	1

Justification: The FASTER reactor plant has a 100 day refueling cycle and is a small compact core, just 300MWth in size and only 55 core assemblies. Replacing 1/3 of the 55 core assemblies every 100 days will take approximately 6 days for refueling. Replacing non-instrumented test assemblies will add some time to this estimate. Removing instrumented test assemblies and closed loop test assemblies will add additional time onto the reactor downtime, but these test assemblies are not replaced every cycle. In fact, they are expected to remain in the core for four (4) cycles. So, based upon this evaluation, it is expected that the FASTER reactor plant will have an availability of at least 80% that will improve with time.

Goal T3. Capability to accommodate secondary missions (e.g., electricity, isotope production, etc.) of modest value (million dollar) without compromising primary mission of testing fuels and materials for advanced reactor technologies

Criterion T3.1 Identification of Secondary Missions

Metric T3.1.1 Number of secondary missions

Metric	Sale of energy products	Other secondary missions	None
Score	3	2	1

Justification: FASTER is equipped with an energy conversion system which will allow converting the 300 MW_{th} into electricity and put it on the electrical grid. This is expected to represent significant revenue which will surely help recover a large fraction of the annual operating costs. Furthermore, special assemblies initially designed for FFTF could be modified for use in FASTER and would enable some isotope production which will further increase the revenue streams from the FASTER reactor plant.

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