ANL-ARC-288

Advanced Fast Reactor – 100 (AFR-100) Report for the Technical Review Panel

Nuclear Engineering Division

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EXECUTIVE SUMMARY

This report is written to provide an overview of the Advanced Fast Reactor-100 in the requested format for a DOE technical review panel. This report was prepared with information that is responsive to the DOE Request for Information, DE-SOL-0003674 Advanced Reactor Concepts, dated February 27, 2012 from DOE's Office of Nuclear Energy, Office of Nuclear Reactor Technologies. The document consists of two main sections. The first section is a summary of the AFR-100 design including the innovations that are incorporated into the design. The second section contains a series of tables that respond to the various questions requested of the reactor design team from the subject DOE RFI.

The AFR-100 incorporates various innovations that when taken together improve the performance and reduce the cost of the AFR-100 plant. In order to develop and incorporate innovations, the following specific design goals have been established for the AFR-100:

- Long-life core (30 years) with no refueling
- Proliferation resistance for export market
- Modular construction and transportability
- Inherent safety and passive systems
- Easy operation and maintenance
- Simple design to reduce cost
- Actinide transmutation
- Hydrogen co-generation and desalination capabilities, if needed.

The various innovations that have been adopted include:

- Compact Small Modular Core – Non-vented and vented options
- Advanced Shielding Materials
- Advanced Structural Materials
- Compact Fuel Handling System
- Advanced Electromagnetic Pumps
- In-vessel Primary Heat Transport System (PHTS) Configuration with a cold pool and core cover
- Intermediate Heat Exchanger (IHX) – replaceable kidney-shaped twisted tube heat exchanger
- Direct Reactor Auxiliary Cooling System using twisted tube heat exchanger technology
- Advanced Instrumentation and Control technology
- Advanced S-CO_2 Brayton Cycle System

The mission of the AFR-100 is to provide electrical power to remote or isolated areas where the cost of extant electricity is high, where the electrical generation needs are modest (that is where a gigawatt electricity power production is not needed), and the infrastructure to support frequent refueling is not installed or required. Some target areas would be island nations, remote areas such as Alaska, and military installation.

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1 OVERALL PLANT DESCRIPTION SUMMARY

This report presents a summary of the overall plant description for the Advanced Fast Reactor-100 for input to the U.S. Department of Energy's Technical Review Panel. The AFR-100 incorporates various innovations that when taken together improve the performance and reduce the cost of the AFR-100 plant. In order to develop and incorporate innovations, the following specific design goals have been established for the AFR-100:

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The key plant design parameters for the AFR-100 are summarized in Table 1.

Table 1 – AFR-100 Overall Plant Design Parameters

Reactor Power	~100 MWe/ 250 MWth
Core Fuel	Uranium binary metallic fuel – low enrichment
Core Life	30 years without refueling
Plant Life	60 years
Reactor Vessel Size	6.19 m ID, 11.07 m height, 25.4cm thickness
Coolant	Sodium
Coolant Temperature, Inlet/Outlet	395ºC/550ºC 155C ΔT
Power Conversion Cycle	Supercritical CO_2 Brayton Cycle
Thermal Efficiency	~41% net

The overall plant site is depicted in Figure 1, and the primary system, the intermediate system and the Brayton cycle power conversion system are depicted in Figure 2 As shown in Figure 3 the primary and intermediate systems are embedded mostly below grade level.

The primary heat transport system is configured as a pool-type arrangement, with the reactor core, primary pumps, intermediate heat exchangers, and direct reactor auxiliary cooling system (DRACS) heat exchangers submerged and contained within the single reactor vessel. Within the reactor vessel, the primary self-cooled electromagnetic pumps (4) take suction from the lower regions of the cold pool and discharge the sodium into a header that distributes the sodium into multiple feeder pipes. The feed pipes distribute sodium evenly into the inlet plenum. The inlet plenum distributes the primary sodium to the inlet of the core assemblies located in the core assembly lower adaptor. The core assemblies are individually orificed for proper flow distribution. The sodium is heated as it flows through the core (going from T_c to T_h) exiting the core assemblies at the outlet plenum. The hot sodium is collected within the outlet plenum. A core cover directs the hot sodium through a piping duct to the inlet of the intermediate heat exchangers (IHX). The primary sodium flows down through the IHX, going from T_h to T_c, giving its sensible heat to the intermediate heat transfer fluid (also called the secondary sodium). The primary heat transport system is shown in Figure 4.

The primary sodium enters the IHX shell about 30 cm below the upper tubesheet to minimize thermal shock of the tubesheet during transient conditions. The sodium flows down through the shell side of the IHX around baffle plates that evenly distribute the sodium in the IHX. After the primary sodium transfers its heat to the intermediate sodium, it exits the IHX about 30 cm above the lower tubesheet and into the lower regions of the cold pool. The IHX is kidney-shaped to contour to the annular gap between the core barrel and the reactor vessel to minimize the overall diameter of the reactor vessel. A guard vessel surrounds the reactor vessel to capture and contain any reactor vessel leakage and to prevent the IHX inlet, DRACS heat exchangers, and core assemblies from being uncovered if the reactor vessel should leak.

The AFR-100 utilizes a supercritical carbon dioxide Brayton cycle power conversion system to transfer the thermal energy of the reactor core to electrical energy on the grid. The secondary sodium, once heated by the primary sodium in the IHX, is pumped through parallel stages of compact heat exchangers that heat supercritical carbon dioxide to high temperatures in the S-CO_2 Brayton cycle balance of plant.

The intermediate sodium exits the IHX and flows to the Na-to-CO_2 heat exchanger located on the nuclear island. The Na-to-CO_2 heat exchanger is part of the Brayton cycle power conversion unit. The intermediate sodium heats up the supercritical CO_2 which then flows into the turbine generator performing work and generating electricity. The CO_2 then goes through a series of recuperator heat exchangers, coolers, and compressors before it re-enters the Na-to-CO_2 heat exchanger.

The CO_2 rejects about 58% of its heat to the forced draft cooling tower which provides the ultimate heat sink for the Brayton cycle power conversion unit. The overall thermodynamic cycle is shown in Figure 5.

Core assemblies are transferred at 30 year intervals from the core to a sodium-filled storage tank located within the reactor building. Fresh fuel is pre-staged in this storage tank. The refueling system consists of an innovative compact fuel handling system (pantograph refueling system) with a single rotatable plug as shown in Figure 6. During normal reactor operations, the refueling system is not located on the site because of the long-lived core. It is only required every 30 years. Thus, it is impossible to remove fuel from the reactor vessel without the refueling machine. In addition, a plug is required to be removed from the reactor vessel to accommodate insertion of the in-vessel refueling machine. Specialized equipment is required to remove the plug and insert the in-vessel refueling machine while maintaining the inert atmosphere of the primary vessel. The reactor vessel is maintained in a sealed configuration until refueling is required.

During a refueling event, various pieces of equipment are transferred to the reactor site. This equipment includes the refueling equipment removal coffin, the in-vessel fuel handling machine, and the fuel unloading machine. This equipment is shipped and installed at the site before refueling is required. In addition, fresh fuel is also transferred to the site in preparation for refueling. Because of the size of the equipment and fuel transfer locks, the equipment can be installed in the reactor building during power production, if needed. The fuel unloading machine is used to transfer the fresh fuel from the transfer position into the fuel storage tank. This pre-positioning of fuel is required to speed up the process of replacing the spent fuel with fresh fuel that is already heated and submerged in sodium. At the end of core life, the reactor is shut down and the spent fuel is allowed to decay until the time when the spent fuel is sufficiently cool to allow transfer of the spent fuel to the fuel storage tank.

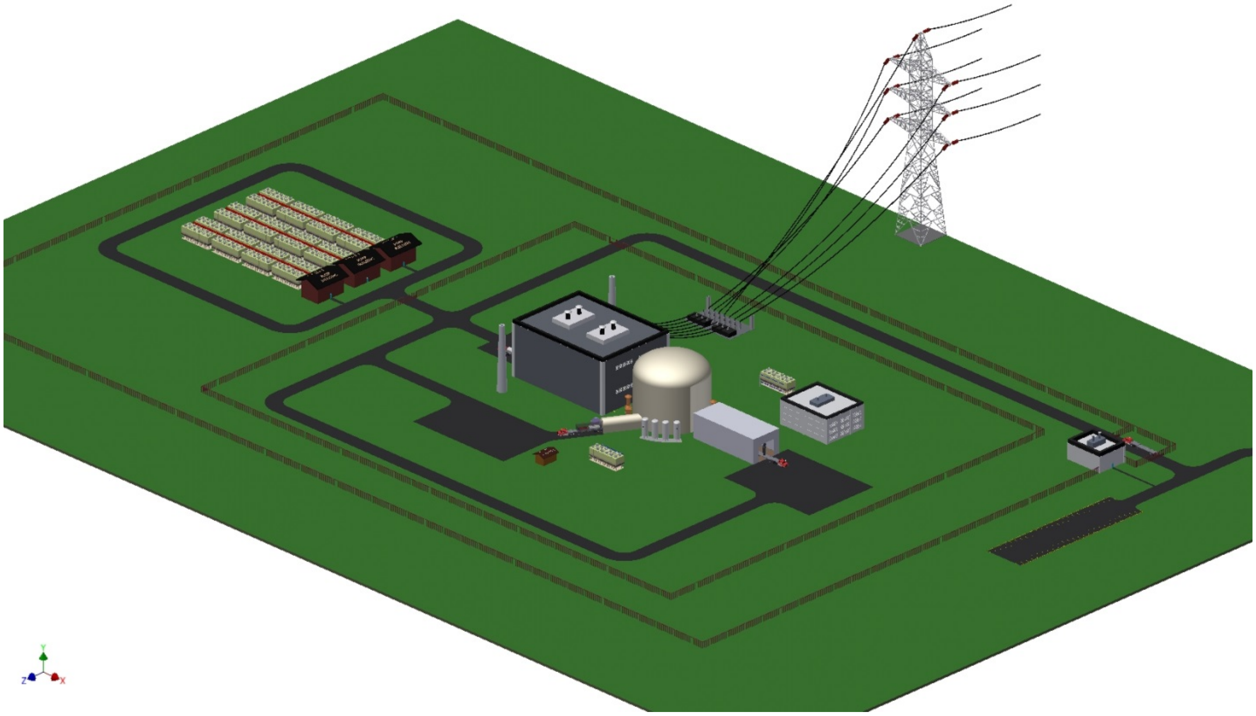

Figure 1 - AFR-100 Site Layout

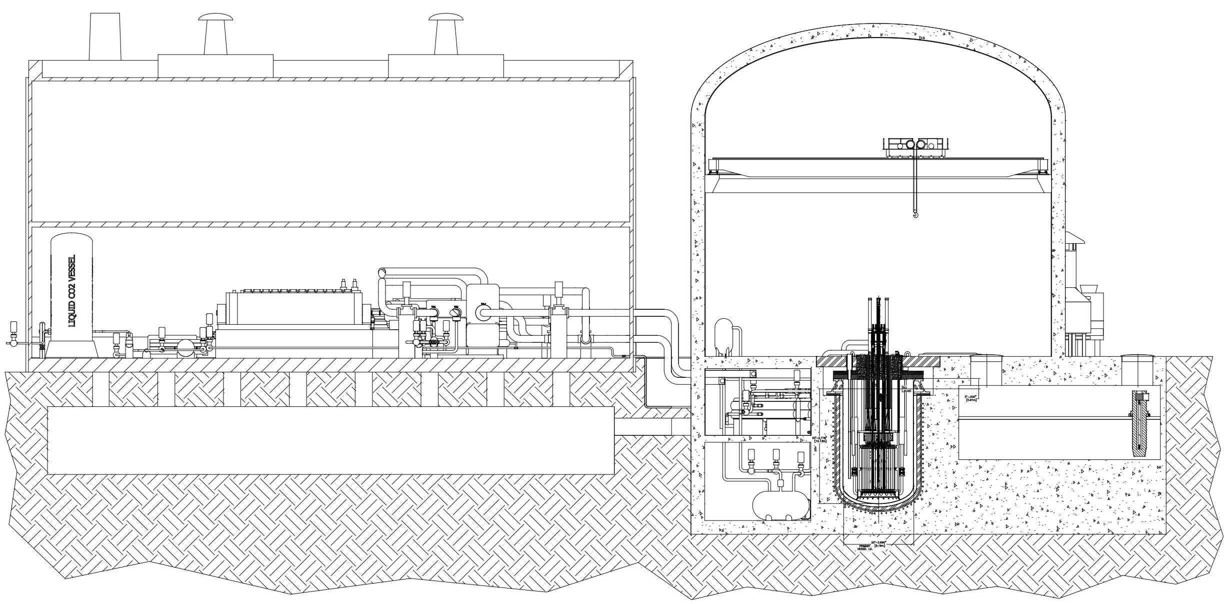

Figure 2 - AFR-100 (Showing Primary, Intermediate, and BOP Systems)

Normal decay heat removal is from the reactor core, through the primary and intermediate heat transport systems and through portions of the balance of plant. When site power is lost, emergency decay heat removal is through the primary heat transport system under natural convection cooling and through multiple (parallel) direct reactor auxiliary cooling systems (DRACS). There are three separate and independent DRACS cooling systems that each provide for 0.25% of 100% full power heat removal. One of the cooling systems can be down for maintenance with two available. This configuration provides for the passive removal of decay heat from the reactor core under all potential accident conditions including various protected and unprotected accidents.

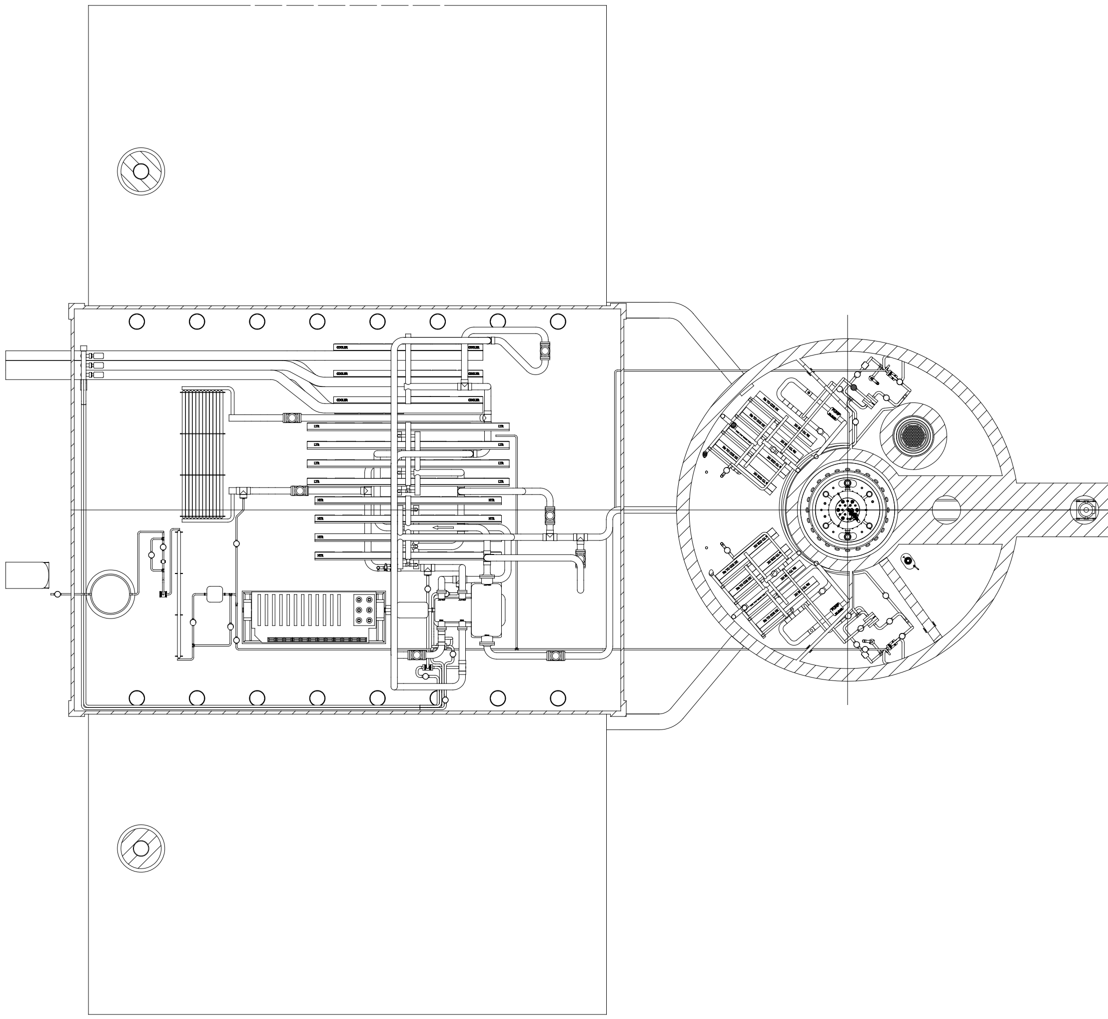

Figure 3 - AFR-100 in Plan View

Safeguard systems are provided to ensure that fuel is not diverted from its intended use in the reactor. Special nuclear materials are always in the form of either fresh fuel assemblies or used fuel assemblies for the AFR-100. The transfer of fuel is monitored when it is undergoing movement from the reactor vessel to the storage vessel to the transfer building. During normal reactor operations, there is no transfer of fuel and the fuel is located in the reactor vessel and possibly in the storage vessel submerged under sodium. There is no possible way that fuel can be diverted from these areas without the use of the in-vessel fuel handling system, the fuel unloading machine, and other sophisticated technical hardware specifically designed for AFR-100 use.

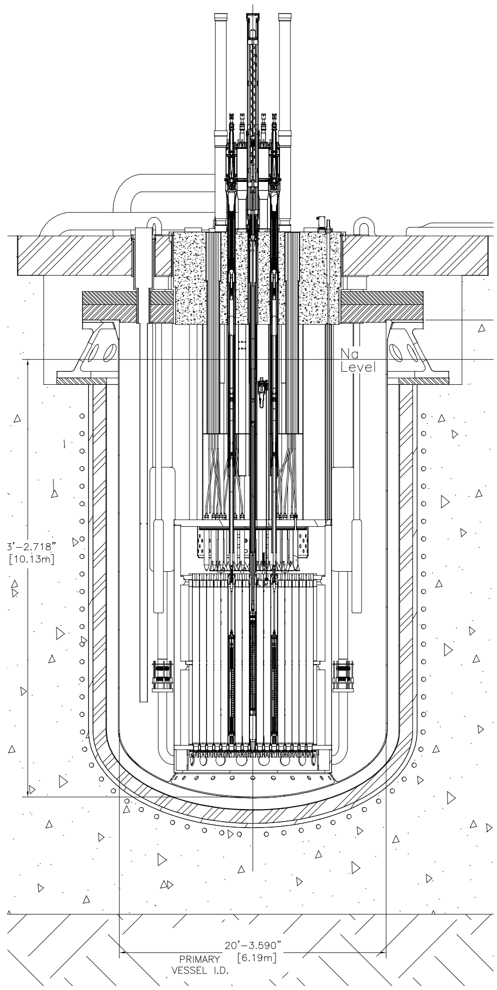

Figure 4 – AFR-100 Elevation View of Primary Heat Transport System

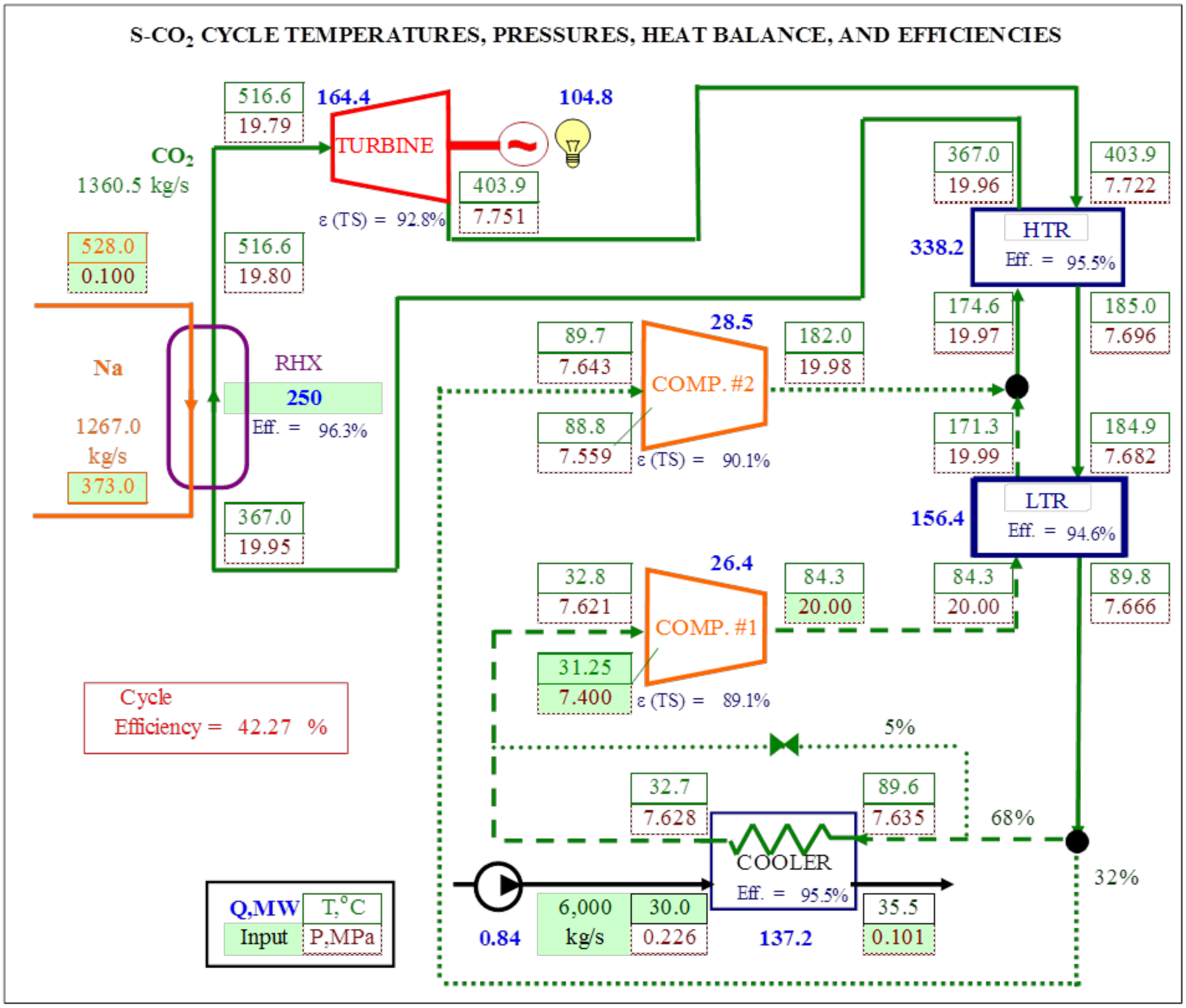

Figure 5 - AFR-100 Thermodynamic Cycle

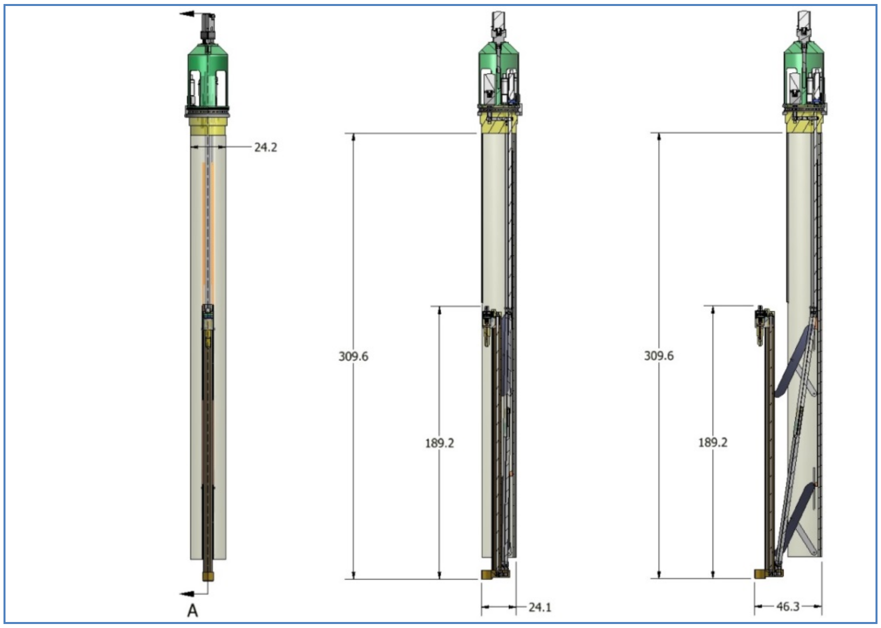

Figure 6 - AFR-100 In-vessel Transfer Machine

The following summarizes some of the innovations adopted for the AFR-100.

The AFR-100 power rating was developed targeting the emerging markets where a clean, secure and stable electricity resource is required, but a large-scale plant cannot be accommodated. The AFR-100 is targeted for small local grids, transportable from pre-licensed factories to remote plant sites for affordable supply, and operated for a long period without refueling. To achieve these strategic goals, several design requirements were proposed for the AFR core development; i.e., a power rating of ~100 MWe that is equivalent to 250 MWth for a target thermal efficiency of at least 40%, a long refueling interval (30 years or more) with small burnup reactivity swing, a core barrel diameter of less than 3.0 m, and an active core height of less than 1.5 m.

Under the US-DOE programs, several options and innovative fast reactor technologies have been investigated or are being developed for improving overall core performance, achieving capital cost reductions, and increasing inherent safety. These features include the following:

- Compact Core – Vented and Non-Vented Core
- Advanced Shielding Materials
- Advanced Structural Materials
- Passive Core Restraint System
- Compact Fuel Handling Systems
- Electromagnetic Pumps
- In-vessel PHTS configuration
- Core Cover with cold pool
- Reactor vessel and core support configurations
- Replaceable kidney-shaped tube-and-shell twisted tube Intermediate Heat Exchanger
- Direct Reactor Auxiliary Cooling System – cold pool – using twisted tube heat exchanger technology
- Advanced I&C
- Containment options
- Advanced S-CO_2 Brayton Cycle System

These innovative fast reactor technologies are not currently available, but it is expected that they will be when the AFR-100 is deployed. Thus, the AFR-100 core concept developed herein incorporated these technologies in order to investigate the effect on plant economics. To this end, the typical core design parameters were determined by extending the design criteria beyond the current irradiation range.

Compact Small Modular Core (Vented and non-Vented)

U-10Zr binary metallic fuel was used as the primary fuel form for the AFR-100 because of its inherent safety characteristics, favorable performance for long refueling intervals, and finally because it can be rapidly deployed due to its extensive use in fast reactors such as EBR-II. Other fuels were also investigated as part of this study. In order to specify the design parameters, the feasible design domain of the small fast reactor was explored using sensitivity studies. The specific power density was determined so as to maximize the cycle length and the assembly pitch.[1] The active core height and number of driver assemblies was selected to ensure that the core fits within the 3.0 m diameter core barrel and that the active core height is less than 1.5 m.

A unique enrichment zoning strategy was adopted for the AFR-100 core for which the low enriched fuel is surrounded axially and radially by higher enriched fuel. This enrichment zoning strategy allows for a 30-year refueling interval with a minimal burnup reactivity swing and a flat power distribution. The core performance characteristics, kinetics parameters, reactivity feedback coefficients, and the reactivity control requirement were calculated using the ANL suite of fast reactor analysis codes.

Orifice design calculations and steady-state thermal-hydraulic analyses were performed using the SuperEnergy2-ANL code. Thermal margins were evaluated by comparing the peak temperatures to the design limits for parameters such as the fuel melting temperature and the fuel-cladding eutectic temperature. The inherent safety features of the AFR-100 core were assessed using the integral reactivity parameters of the quasi-static reactivity balance analysis. The resultant core design is shown in Figure 1.

The AFR-100 core uses fission-gas vented fuel as one option. This fuel form does not affect the core performance characteristics, but does affect the radioactivity levels in the primary coolant and cover gas space. The volumetric activity of the primary coolant increases slightly because of the reduced primary coolant and the additional activation products from dissolved fission gas. However, additional biological shielding can be incorporated to maintain dose levels below the design constraint. Using vented fuel can increase the effective burnup of the fuel, reduces the overall length of the core assemblies and thus can reduce the overall height of the reactor vessel, in-vessel fuel transfer machine and ex-vessel fuel transfer machine.

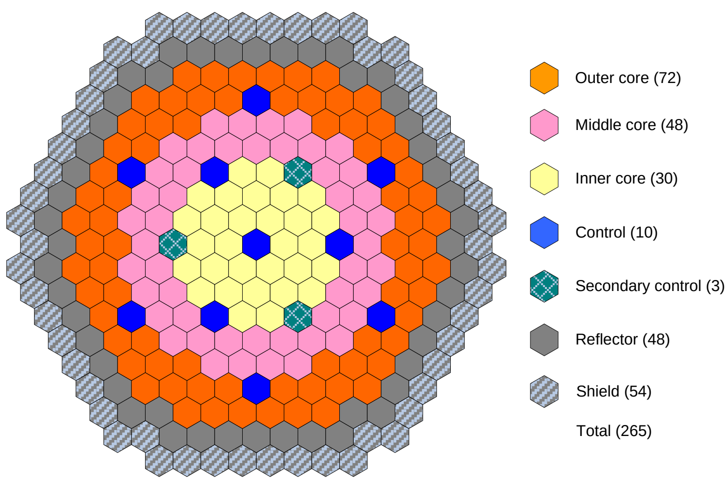

Figure 7 - AFR-100 Radial Core Layout

[1] Assembly pitch is the center-to-center distance between assemblies.

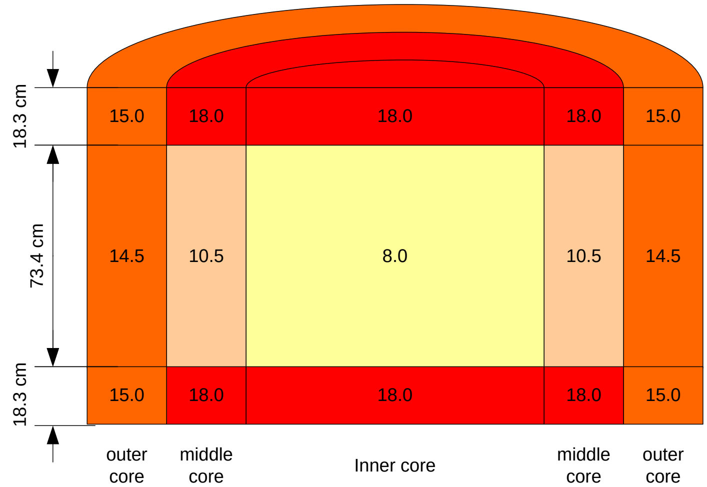

Figure 8 – Enrichment Zoning

Advanced Shielding Materials

To reduce the thickness of the radial reflector, which reduces capital cost by decreasing the core (and thereby reactor vessel) diameter, advanced shielding material was used. The AFR-100 reactor core was designed with one row of shielding made from zirconium hydride (ZrH_2) and boron carbide (B_4C). These advanced shielding materials were selected because extensive irradiation experience has been gained with their use in commercial and research reactors. For the AFR-100, the fractional loading in the radial shielding assembly was selected as 75% for zirconium hydride and 25% for (natural) boron carbide.

With the advanced shielding material, the fast neutron fluence beyond the core barrel was reduced below the reference core concept due to the moderating effect of the hydrogen in the zirconium hydride despite the fact that one reflector row was removed. Furthermore, the thermal neutron flux beyond the core barrel was significantly reduced by neutron absorption in the boron.

Core Restraint System

Bowing of core assemblies can cause significant changes in reactivity during start-up, overpower, and loss-of-flow without scram transients. A performance objective of a sodium-cooled fast reactor is to maximize inherent safety; in particular, to assure net negative reactivity insertion during transients. The core restraint system controls core assembly bowing so as to ensure negative reactivity feedback during transients.

The NUBOW-3D software was originally developed to analyze core bowing over the life of the reactor due to thermal/irradiation creep as well as swelling using duct temperature and neutron fluxes calculated with core physics codes as input. In addition, NUBOW-3D calculates the net reactivity change due to duct bowing displacements using reactivity displacement worths calculated from the physics codes. These features allow NUBOW-3D to be used as an analysis and design tool for core restraint systems.

In order to support the development of the AFR-100 vented and non-vented core design concepts, NUBOW-3D was used to analyze the preliminary core restraint system. As part of the input, NUBOW used duct temperatures, flux, and displacement reactivity worths obtained from the core physics codes SE-II, DIF-3D, and VARI-3D. The vented and non-vented core designs are identical with the exception that the non-vented core has a gas plenum above the fuel that increases the overall core height. As a result, the core physics analyses were only carried out for the vented core design. The temperature and flux profiles from the vented core analysis were thus applied to the non-vented case since the difference in the core design begins above the active core region. Details are provided in the main body of the report.

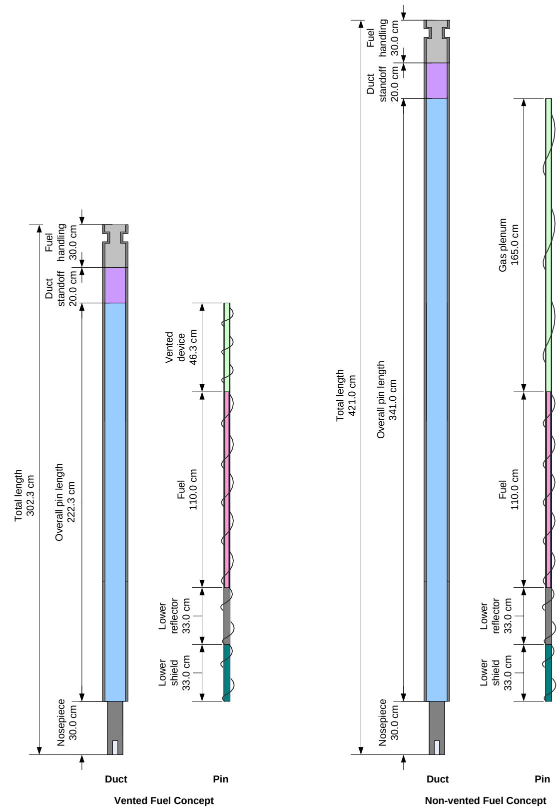

Figure 9 - Schematic of Duct and Pins with and without Venting

In terms of design approach for the AFR-100, two common methods were considered for the restraint system; namely, the *free flowering* and *limited free bow* concepts. The limited free bow system has been shown to provide better inherent safety characteristics and as such has been selected for the AFR-100.

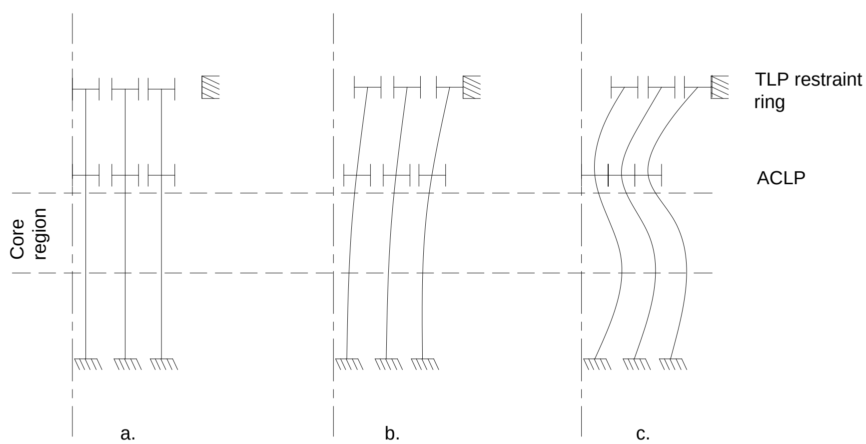

Figure 10 – Pictorial of Limited Free Bow

Compact Fuel Handling Systems

A conceptual pantograph-type fuel handling machine (FHM) has been developed for the AFR-100. This fuel handling mechanism would be removed during power operations and used at other AFR-100 sites for the removal and insertion of fuel. The design was developed to reduce the height of the mechanism to suit a smaller sized primary vessel. Figure 2 shows the overall assembly of the pantograph mechanism concept in the fully extended and retracted positions. Although preliminary, enough detail has been incorporated to simulate and evaluate overall operational characteristics.

The main design goal for the pantograph FHM was to reduce the size of the FHM assembly both inside and outside the reactor vessel in order to reduce overall costs of the primary reactor system as well as the containment building.

The pantograph is controlled by a computerized drive system which provides several distinct and isolated motions to manipulate the fuel, control rod, reflector and shielded assemblies inside of the reactor vessel. The drive motors utilize planetary reduction gear sets to achieve the required torque to move the simulated core assembly while under maximum design loads. The calculated maximum forces required to operate the mechanism have been used to determine the size of various components based on expected frictional resisting forces. Actuation of any single isolated motion does not affect other component position or orientation. The motions controlled are:

1. Raising or lowering of the entire mechanism to apply a required hold down force to the core assemblies surrounding the core assembly being inserted or removed from the core (performed only when the pantograph arm is above the position of interest).

2. Horizontal positioning of the pantograph arm above the core assembly of interest via extension or retraction of the pantograph arm linkages (performed only while the entire mechanism is lifted above the core).

3. Rotation of the pantograph inside the vessel to reach various positions above the core by rotation of the mechanism on the bearings between the housing and stepped plug (performed only while the entire mechanism is lifted above the core and the arm is retracted).

4. Raising or lowering the core assembly by rotation of a screw mechanism in the gripper head which uses a series of shafts and gears inside the pantograph arm (performed when the FHM assembly is in the lowered or “hold down” position).

5. Rotation of the core assembly inside the pantograph arm by rotating the gripper using a series of shafts and gears (this is performed only while the core assembly is fully lifted).

6. Opening or closing of the gripper to engage or release the core assembly being serviced through a series of shafts and gears (this is performed only with the core assembly fully lowered).

The pantograph FHM is supported from a shielded plug (Figure 13) which fits into a port in the reactor vessel rotatable cover. A stepped shielded circular plug located inside this housing rotates on a set of bearings that support the entire weight of the mechanism from the shielded plug. Bi-directional servo gear motors which drive the mechanism motions are placed above this shielded plug for ease of maintenance, manual control access and increased reliability. Manual connections for operating the pantograph (in case of computer or electrical failure) are located on the top of the stepped circular plug. Mechanical and electronic visual indicators show the positions and orientations of all mechanism components during transfer operations.

In-vessel Primary Heat Transport System

The in-vessel primary heat transport system consists of compact loop submerged within a pool of cold liquid sodium. This technology was used at EBR-II and has been adopted for use within the AFR-100. This PHTS configuration reduces the heat load, thermal stresses, and long-term creep effects on the reactor vessel and reactor vessel enclosure. In addition, this configuration increases the amount of thermal inertia that is seen during severe accidents thus supporting the natural safety aspect of the reactor plant.

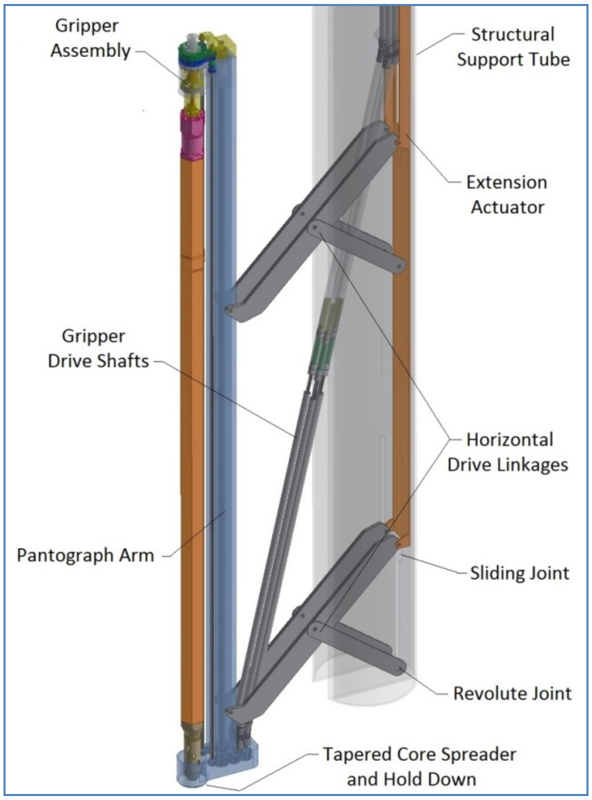

Figure 11 - Pantograph In-vessel Fuel Handling Mechanism (Extended)

Self-cooled Electromagnetic Pumps

Self-cooled EM pumps are proposed for the AFR-100. These pumps must deliver 370kg/s of sodium flow through the core. These pumps are called annular linear induction pumps (ALIPs) and require advancements in insulation materials to ensure their effectiveness at the temperatures and irradiation conditions typical of the AFR-100. Table 2 provides the design values for the primary EM pumps. The secondary EM pumps will be twice the capacity of the primary EM pumps. Also, by adopting advanced magnetic core materials with higher magnetic saturation limit and/or better thermal conductivity, the stator can be designed with improved heat conduction through it. Alternatively, development of an advanced high temperature electrical insulator material that can work at temperatures much higher than the current limit of 1023 K would also be a solution.

ANL has also proposed an alternative ALIP configuration to eliminate power feedthrough penetrations. In this new design, the electrically active components are all placed in the "core" region, see Figure 12. In addition to elimination of the penetrations through the working fluid, this design allows the "outer core" to be a part of permanently installed piping in the reactor vessel and allows the inner stator to be removable without disconnecting the piping. For the conventional design, when pump removal is necessary, the whole pump needs to be withdrawn by disconnecting the piping below the pump.

Intermediate Heat Exchanger

The intermediate heat exchanger is made from advanced ferritic steels (modified (9Cr-1Mo) reducing the overall size of the heat exchanger. To make a compact IHX, two design options were identified to reduce the IHX size and/or space occupied by the by the IHXs in the AFR-100 pool plant configuration. These are: i) high thermal conductance materials and ii) kidney-shaped planar construction. Finally, the units are assumed to be made from ferritic steel (e.g. 9Cr-1Mo) that have thermal conductivities that are ~ 40 % higher in comparison to austenitic steels at 500 ºC.

Table 2 – EM Pump Design Information

	Unit	
Pump type		**Annular Linear Induction**
Design mass flow rate	kg/s	**320**
Design pressure head	Pa	**7.58E+05**
Temperature	K	**668.15**
Drive voltage	Volts	**560**
Drive frequency	Hz	**31**
Length	m	**2**
Diameter	m	**0.648**
Center core hole diameter	m	**0.139**
Magnetic core OD	m	**0.231**
Core OD with containment	m	**0.233**
Annulus thickness	m	**0.041**
Duct ID	m	**0.315**
Duct wall thickness	m	**4.191E-03**
Duct OD	m	**0.32385**
Stator ID	m	**0.32385**
Stator tooth width	m	**0.0333**
Stator slot width	m	**0.0333**
Stator slot height	m	**0.0954**
Stator OD	m	**0.548**
Number of coils	-	**30**
Number of turns	-	**14.121**
Conductor width	m	**0.0266**
Conductor height	m	**0.0064**
Thickness of housing wall	m	**0.050**
Total current per coil	A	**1047.7**
Total power	W	**638627**
Pumping power	W	**282934**
Efficiency	%	**44.3**
Estimated mass	kg	**2554**

The conventional shell-and-tube type heat exchanger design has been deployed in many fast reactor applications, but not in an annular configuration as investigated as part of this work. This concept consists of a number of round tubes attached to a tubesheet inside a cylindrical vessel, with tube size, tube length, and total number of tubes varying depending on the system thermal requirements. The tube bundle normally contains a number of baffles to accomplish the dual objectives of providing a support structure for the tubes, and to direct the shell-side flow across the tubes rather than along the tubes. The resulting cross-flow on the shell side yields a relatively high pressure drop because energy is used to reverse the flow rather than to enhance heat transfer.

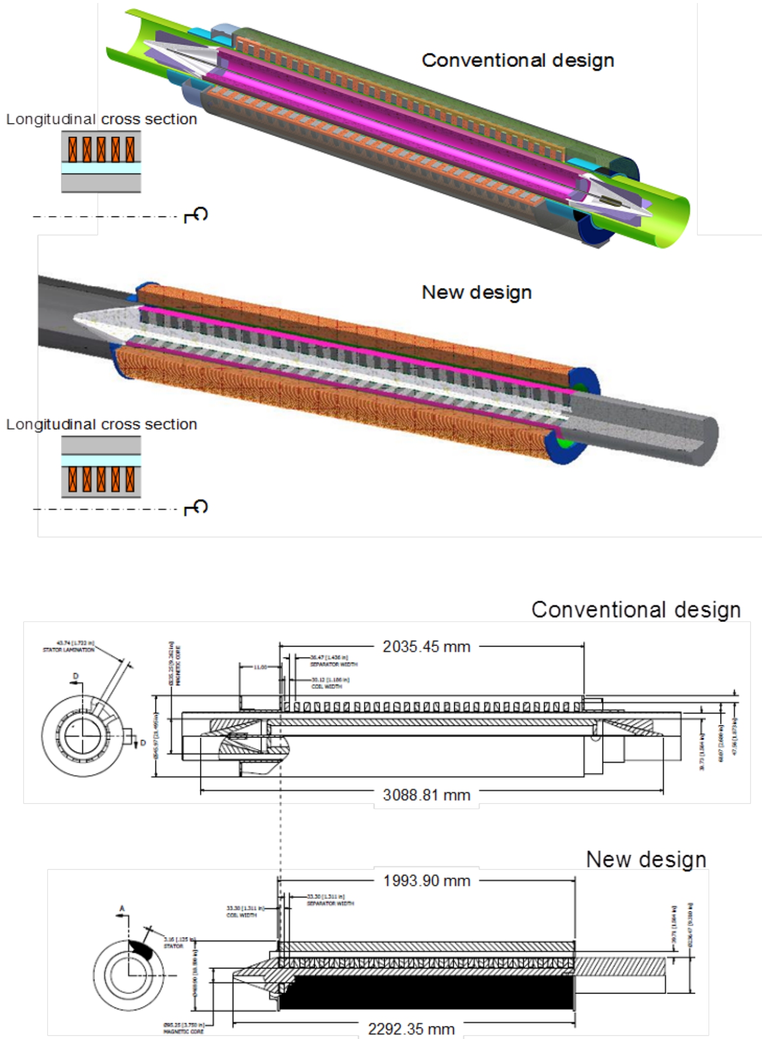

Figure 12 – Self-cooled Electromagnetic Pump

The Twisted Tube heat exchanger originated in Eastern Europe and became commercially available in Sweden in the mid 1980's. This concept was developed to overcome inherent limitations with conventional shell-and-tube designs described above. Applications of this technology have principally been in areas involving single-phase and condensing flows in the pulp and paper industries, as well as the chemical process industries. The twisted tube concept consists of a tube bundle assembled from uniquely formed tubes in an arrangement that does not use baffles; see Figure 14. The tubes are formed into an oval cross section with a superimposed helical spiral in the axial direction of the bundle. The forming process ensures that tube wall thickness remains constant and the material yield point is not exceeded thereby retaining mechanical integrity. The tube ends are round to allow conventional tube-to-tubesheet joints. A wide range of tube materials have been used including carbon and stainless steels, Cr-Mo alloys, duplex and super duplex alloys as well as exotic metals such as titanium, zirconium and tantalum. Tube sizes may vary from 1.3 cm to 2.5 cm. As illustrated in Figure 15, the tubes are assembled into a bundle on a triangular pitch one row at a time with each tube oriented to align the twists at every plane along the bundle length. This alignment results in tubes contacting adjacent tubes at many points along the length of the bundle, and this minimizes flow-induced vibration. Bundles can be constructed with more than 5000 tubes in tubesheets up to 1.8 m in diameter; tube lengths can range up to 24 m. The shell-side flow path is complex and predominantly axial in nature. Typically, the shell side flow area is approximately equal to the tube side flow area. The twisted tube design imparts a swirl flow on both the shell and tube sides of the unit that enhances the heat transfer coefficients relative to those achieved in a traditional parallel flow heat exchanger designs. To the best of our knowledge, the twisted tube concept has not been previously employed in fast reactor applications. This is the current final design innovation that is incorporated into the advanced IHX design for the AFR-100.

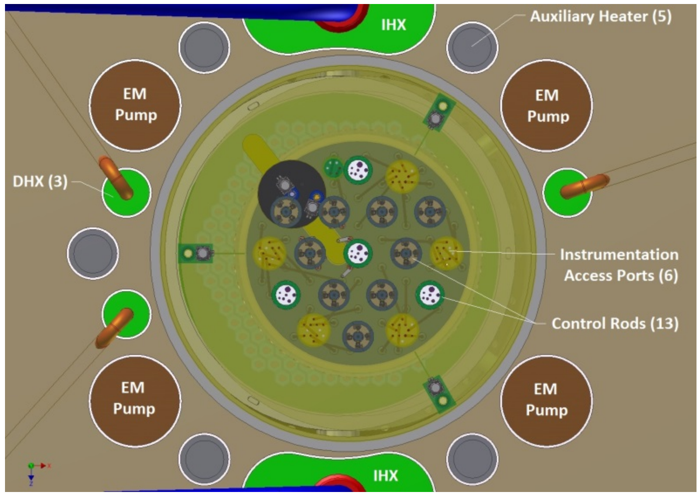

Figure 13 – Plan View of AFR-100 Reactor Vessel

Figure 14 – Illustration of a Fully Assembled Twisted Tube Heat Exchanger Bundle

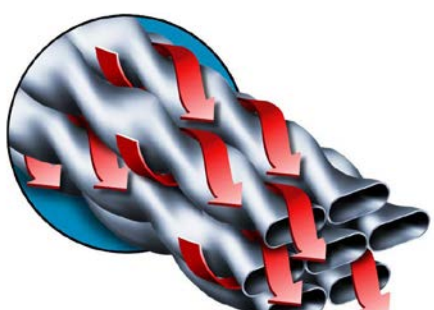

Figure 15 - Illustration of Twisted Tube Bundle Arrangement Showing Flow Behavior

Instrumentation and Control

Various advances in I&C and automation will be required to support the AFR-100. These advancements include passively-cooled high temperature nuclear detectors, advanced near autonomous reactor plant instrumentation and control, high temperature sensors for pressure and level, high temperature sensors for monitoring the health of the reactor vessel and other permanent structures in the reactor plant, automated refueling of the reactor (as discussed above), and advanced reactivity control.

Advanced Supercritical Carbon Dioxide (S-CO_2) Brayton Cycle Technology

The S-CO_2 Brayton cycle technology has been adopted for the AFR-100 with the steam plant as a backup option. This advanced cycle consists of a compact sodium-to-S-CO_2 heat exchanger bank, high pressure piping, a compact turbine powering a generator, multiple banks of compact S-CO_2–to-S-CO_2 heat exchangers for high temperature and low temperature recuperation, an S-CO_2–to-water heat exchanger (cooler), and two compressors as shown in Figure 2 and Figure 3. The overall cycle efficiency is 42.3% and the net cycle efficient is over 41%. The cycle operates at a maximum working pressure of approximately 20MPa (2,900 psig) and 520.2°C. The sodium-to-S-CO_2 heat exchangers are compact diffusion-bonded modular units for which failures are not expected, or if they occur, are expected to result in the formation of microcracks and a small flow of CO_2 into the sodium. A postulated conservative large break in a sodium-to-S-CO_2 heat exchanger from the high pressure S-CO_2 to the low pressure IHTS sodium will result in a pressure pulse into the IHTS that will need to be accommodated by the plant protection system. The PPS will need to take action to manage the pressure pulse and isolate the faulted heat exchanger from the sodium system. After a large enough break, the IHTS sodium will need to be cleaned up in the IHTS sodium purification system. The S-CO2 Brayton cycle technology is small and compact. The technology has the potential for significantly reducing the cost of the BOP systems and components compared with a conventional superheated steam cycle.

AFR-100 Summary

Table 3 provides a summary for the summary characteristics of the AFR-100. Figures Figure 16 and Figure 17 provide three-dimensional views of the AFR-100 reactor building. Table 4 provides a tabular description of the AFR-100 plant.

Table 3 - Summary of AFR-100 Fast Reactor Concept

	AFR-100 concept 3TT
Core	U-Zr metal fueled core, 30 year life
Core Venting	No
Primary Pumps	Electromagnetic
PHTS Configuration	Core Cover – Cold Pool
Core Support	Top Supported with redundant support rods
Intermediate Heat Exchanger Configuration	Kidney-shaped Removable IHX – Straight twisted tubes
Reactor Vessel Support	Conical Ring – Top Support
Emergency Decay Heat Removal	Direct Reactor Auxiliary Heat Exchangers in cold pool – twisted tube HXs
Primary Purification System	Cold trap technology
Power Conversion System	S-CO_2 Brayton Cycle
Containment	Steel Containment with separate aircraft impact shield
Fuel Handling Mechanism	Single Rotatable Plug with pantograph FHM

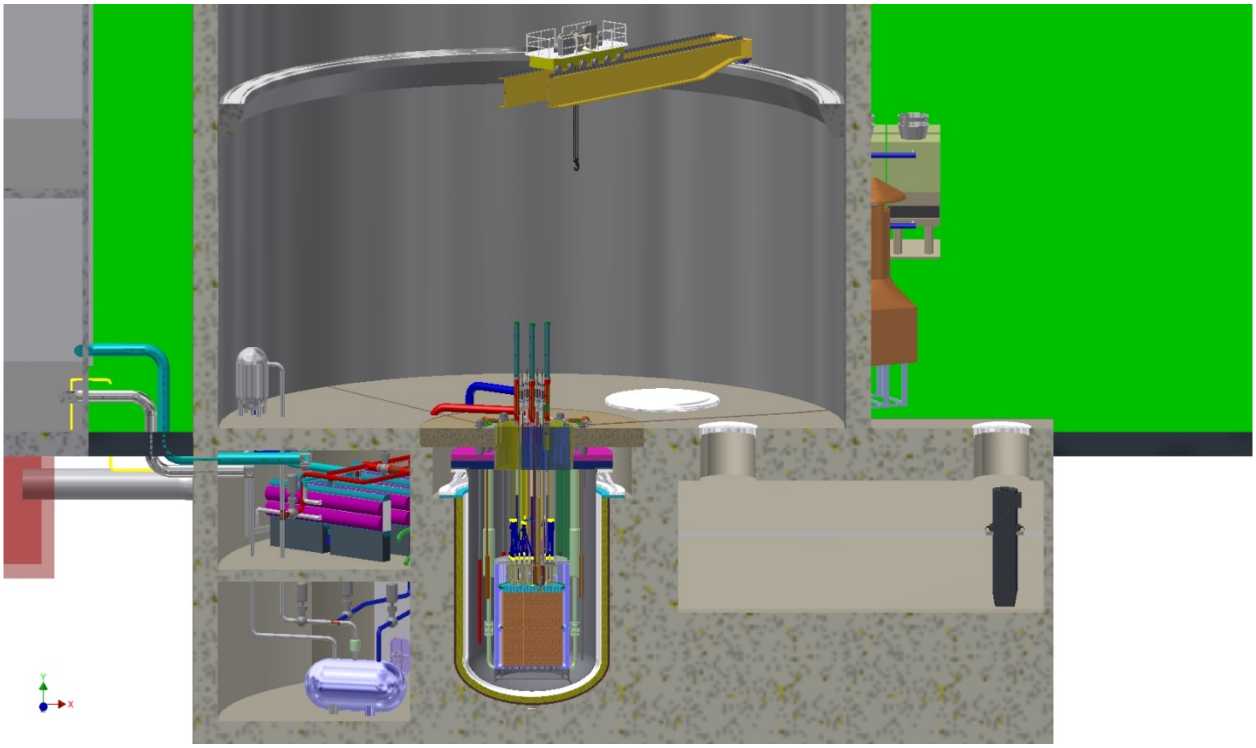

Figure 16 – AFR-100 Reactor Building Elevation View

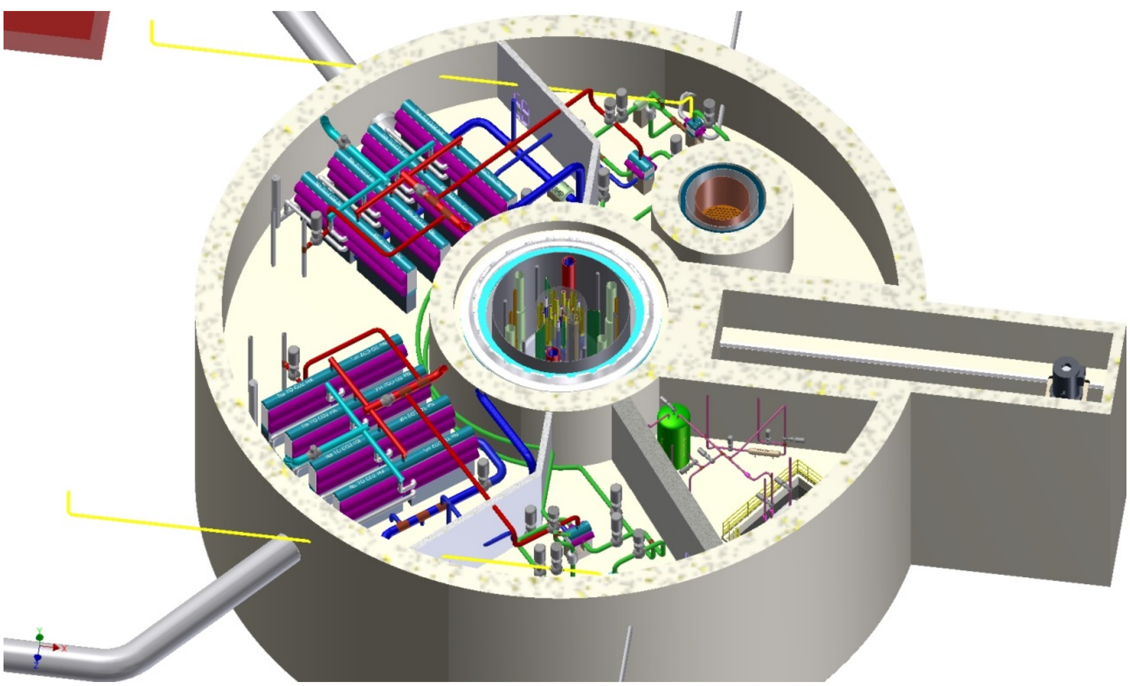

Figure 17 – AFR-100 Reactor Building Plan View

Table 4 - Tabular Description of Advanced Fast Reactor – 100

1. Plant Configuration	
Primary System	Pool (Pumps and IHX in reactor vessel)
Intermediate Loop	Yes, sodium
# of Intermediate Loops	2
# of Turbines	1
Reactor Building Characteristics	Below-grade primary/intermediate Seismic isolation for reactor
2. Energy Conversion and BOP	
Power Conversion Cycle	Supercritical CO_2 Brayton
Thermal Efficiency	42.3% (Cycle)/ 41% net
Turbine Building Characteristics	Above-grade adjacent to reactor No seismic isolation
3. Construction Techniques	
General Approach	Factory fabrication of basic modules Reactor module consists of vessel, enclosure and fixed internals
Transportability	Targeted for domestic and remote sites where a long-lived reactor plant is important and where an SMR is supported by the domestic infrastructure. Shippable modules are assembled in temporary maintenance building and/or reactor building
4. Key Plant Parameters	
Power (MWe/MWt)	~100/250
Primary Coolant	Sodium
Intermediate Heat Transfer Medium	Sodium
Plant lifetime	60 years or greater
Reactor Vessel, diameter (m) height (m)	6.19 (ID) 11.07
5. Core Performance and Safety	
Cycle Length (years)	30
Capacity Factor (%)	90
Fuel Form	U-Zr metallic alloy
Active Core Height, cm	110
Average Power Density (W/cc)	64.3
Fuel Pin Overall Length, cm	222
Initial Heavy Metal Loading, tons	23.9
Specific power density, MW/t	10.5
Average enrichment, %	13.5 U-235 mass fraction per total heavy metal
Burnup Reactivity Swing, %Δk	1.1
Overall fissile breeding ratio	0.80
Average discharge burup, MWd/kg	101
Peak discharge burnup, MWd/kg	172
Peak Fast neutron fluence, 10^{23}n/cm^2	5.97
Linear heat rate (average/peak), kW/m	15.2/28.2
Kinetics Parameters and Reactivity Feedback Coefficient	
Effective delayed neutron fraction (β_{eff}) BOL/MOL/EOL	0.0071 / 0.0056 / 0.0047
Prompt lifetime BOL/MOL/EOL, μs	0.25 / 0.25 / 0.24
Radial Expansion Coefficient, ¢/°C BOL/MOL/EOL	-0.13 / -0.17 / -0.13
Axial expansion coefficient, ¢/°C BOL/MOL/EOL	-0.05 / -0.06 / -0.06

Sodium void worth, $ BOL/MOL/EOL	0.43 / 1.84 / 3.02
Sodium density coefficient, ¢/°C BOL/MOL/EOL	~0.00 / 0.05 / 0.08
Doppler coefficient, ¢/°C, BOL/MOL/EOL	-0.05 / -0.05 / -0.05
Sodium voided Doppler coefficient, ¢/°C, BOL/MOL/EOL	-0.05 / -0.05 / -0.05
Fuel density coefficient, ¢/°C, BOL,MOL,EOL	-0.23 / -0.13 / --0.36
Structure density coefficient, ¢/°C, BOL/MOL/EOL	0.01 / 0.02 / 0.03
Control Rod driveline expansion coefficient, ¢/cm, BOL/MOL/EOL	-10.7 / -22.2 / -12.8
Integral Reactivity Parameters for Quasi-Static Reactivity Balance	
A, power coefficient, ¢, BOL/MOL/EOL	-6.8 / -4.1 / -4.3
B, power/flow coefficient, ¢, BOL/MOL/EOL	-30.9 / -37.4 / -25.5
C, inlet temperature coefficient, ¢/°C, BOL/MOL/EOL	-0.25 / -0.31 / -0.21
$\Delta\rho_{TOP}$, transient over power initiator, ¢, BOL/MOL/EOL	5 / 33 / 9
Sufficient conditions to have favorable inherent safety features	
A/B < 1, BOL/MOL/EOL	0.22 / 0.11 / 0.17
$1 < C\Delta T_c/B < 2$, BOL/MOL/EOL	1.26 / 1.30 / 1.23
$\Delta\rho_{TOP}/\lvert B\rvert < 1$, BOL/MOL/EOL	0.16 / 0.88 / 0.35
6. Coolant and Thermal Performance	
Primary Coolant, T-in (^{o}C) T-out (^{o}C)	395 550
Primary Coolant, Pressure (MPa)	0.1 – 0.3
Primary Coolant, Margin to Boil (^{o}C)	>340^{o}C
Primary Coolant Purification	Cold trap technology
Primary Coolant Corrosion Control	oxygen content < 10 ppm to minimize corrosion
Primary Pumping Power Coolant Flow Rate	4x360kW 4x0.383m^3/s
Intermediate Coolant, T-in (^{o}C) T-out (^{o}C)	373 528
Intermediate Coolant, Pressure (MPa)	0.1 – 0.3
Secondary Coolant, T-in (^{o}C) T-out (^{o}C)	364.4 515.4
Secondary Coolant, Pressure (MPa)	20
7. Fuel Properties	
Fuel Type	U-10%Zr metal alloy – LEU
Thermal Conductivity (W/cm^{o}K)	23.7 (as fabricated at 305°C)
Melting Temperature (^{o}C)	1200°C but decreases with burnup
Bond Material	Sodium
Cladding Material	HT-9 ferritic steel
Pin Diameter (cm)	1.75
Active Core Length (cm)	110
Fission Gas Plenum	165 cm above fuel in pin
8. Refueling System	
Refueling Interval, years	30
Fraction of Core Replaced (%)	100
Refueling Down Time	~6 months

In-Vessel Fuel Handling Machine	Transportable (does not reside at site)
	In-Vessel pantograph design located in single rotating plug
Fuel Unloading Machine	Transportable (does not reside at site) Straight Pull refueling machine with radiation shielding, inert gas, and active cooling
Interbuilding Transfer Cask	Transportable (does not reside at site) with radiation shielding and inert gas
9. Safety Systems	
Emergency Heat Removal	Natural circulation of primary sodium in reactor vessel DRACS loops transport heat to atmosphere heat sink Backed up by guard vessel
Reactor Shutdown Systems	Two independent, diverse design Control rod groups
Inherent Safety Potential	Avoid core damage in double fault beyond-design-basis events
10. Containment System	
Primary Containment Boundary	Reactor vessel, head, and heat exchanger internals Steel liner of reactor containment building
Secondary Confinement	Guard vessel Intermediate System
Containment Design Basis	External Events, Sodium release and sodium pool fire Details TBD
11. Decay Heat Management	
Normal Decay Heat Removal Path	From the primary heat transport loop, through the intermediate heat transport loop, through the balance of plant
Backup Heat Removal System Design	Three direct reactor auxiliary cooling systems (DRACS) comprised of a) natural circulation NaK loop into sodium pool and b)natural draft heat exchanger
Backup Decay Heat Removal Capacity	0.5% of full power total available with one DRACS system down for maintenance Each DRACS is capable of removing 0.625MWt of decay heat.
Ultimate Heat Sink	Outside air through stacks

2 Advanced Fast Reactor-100 - Information Tables for TRP Evaluation

Information Requested	Concept Description
Category I - Safety	
1) Describe design features that address defense-in-depth, accident prevention, accident mitigation, and emergency planning	Defense-in-depth: The defense-in-depth principle provides multiple levels of protection against the release of radioactivity. This involves the traditional physical barriers to the release of radioactivity provided by the fuel matrix, cladding, coolant, primary coolant system boundary, guard vessel, and the reactor containment building. Active or passive safety systems are provided to protect the physical barriers including the reactor shutdown systems and reactor cooling systems. Inherent features of the AFR-100 design provide additional protection of the barriers. One such feature is the strong reactivity feedbacks of the fast neutron spectrum metallic-fueled core with sodium coolant that results in passive shutdown behavior. The AFR-100 design incorporates an intermediate sodium circuit, located between the radioactive primary sodium and the energy conversion system working fluid, such that the high pressure supercritical CO_2 (S-CO_2) working fluid is excluded from the interior of the containment. Thus, the containment does not need to withstand the pressurization loadings that could result from a S-CO_2 line break. The containment needs to withstand the effects of thermal loadings including those resulting from sodium fires. However, the pressurization loadings that can result from burning of sodium are limited due to the limited mass of oxygen inside the containment such that only a small fraction of the primary or intermediate sodium inventories can burn before the oxygen is consumed and the fires go out. Accident Prevention: The AFR-100 is designed with a high level of reliability such that specific traditional accident initiators are eliminated or accident initiators are unlikely

2 Advanced Fast Reactor-100 - Information Tables for TRP Evaluation

Information Requested	Concept Description
	to occur. The liquid sodium coolant, metallic fuel, and pool-type vessel primary system configuration provide a highly reliable reactor with large operational safety margins. The pool-type reactor vessel and surrounding guard vessel without external primary system piping eliminate loss-of-coolant initiators due to primary sodium pipe ruptures. The in-vessel configuration eliminates or reduces the likelihood of flow blockage accidents. Liquid sodium is a low pressure liquid metal coolant with thermophysical and heat transport properties that provide superior heat removal and heat transport together with a large margin to the sodium boiling temperature. The metallic fuel with its high thermal conductivity operates at temperatures below the sodium boiling temperature. The fuel, cladding, coolant, and structural materials are stable and compatible. Purity of the sodium coolant is maintained through the traditional reliable cold trapping approach. The reactor vessel, internal components, and the in-vessel configuration are designed to enable In-Service Inspection (ISI) and monitoring for detection of degradation or performance changes. Accident Mitigation: Accident Mitigation involves: Provision of protection in the event of equipment failures or operating errors; and Provision of additional protection of the public health and safety in an extremely unlikely event that is not expected to occur during the lifetime of the plant or was not foreseen at the time that the plant was designed and constructed. Due to the inherent behavior and passive safety design features of the AFR-100, it is expected that anticipated operational occurrences will not escalate into accidents. Although specific traditional accidents such as loss-of-coolant or flow blockage are practically eliminated,

2 Advanced Fast Reactor-100 - Information Tables for TRP Evaluation

Information Requested	Concept Description
	other traditional accidents such as reactivity insertion due to withdrawal of one or more control rods, loss-of-normal heat sink, or station blackout remain. Due to the passive safety design of the AFR-100, the core, intermediate heat exchangers (IHXs), and DRACS emergency decay heat removal heat exchangers remain covered by the liquid sodium coolant and natural circulation heat transport removes the core power which is removed from the reactor system by the independent, NaK, natural circulation decay heat removal loops each having a DRACS sodium-to-NaK heat exchanger immersed in the sodium cold pool and a natural draught NaK-to-air heat exchanger outside of the containment. Two out of the three emergency decay heat removal systems are sufficient to cool the core following shutdown providing redundancy. The fuel and sodium coolant temperatures remain below the sodium boiling temperature. There is no need for reliance upon active safety systems or operator actions to provide for cooling of the core or heat removal from the reactor system. Traditionally, escalation into a more serious event requires the occurrence of additional failures following the onset of the accident initiator. An example of such an additional failure has been the assumption of failure to scram the reactor by the primary and secondary shutdown systems. For the AFR-100, even when both of the two independent, redundant, and highly reliable shutdown systems fail to operate or operators cannot take action to insert control or shutdown rods, the inherent strong reactivity feedback of the fast neutron spectrum metallic-fueled core causes the power level to autonomously decrease to match the heat removal from the reactor system.

2 Advanced Fast Reactor-100 - Information Tables for TRP Evaluation

Information Requested	Concept Description
	If the reactor vessel were to fail, the steel guard vessel would retain the primary sodium such that the core, IHXs, and DRACS heat exchangers remain covered by single-phase liquid sodium primary coolant. The reactor vessel and upper closure head that is sealed to the vessel will contain radioactivity inside of the vessel. A containment building structure is provided to contain radionuclides and protect against the effects of a postulated aircraft impact. <u>Emergency Planning</u>: The exclusion zone and emergency planning zones that are currently required by the U.S. NRC to be about ten miles for the plume exposure pathway and about 50 miles for the ingestion exposure pathway provide additional Defense-in-Depth to protect the public health and safety. Given the small size of the AFR-100 and the passive safety behavior, a low release frequency and low doses are expected. It is envisioned that the exclusion and emergency planning zones might be reduced in size relative to current LWRs, provided that revision of current regulations can be achieved.
2) Provide sufficient design information on the shutdown and decay heat removal systems to allow an assessment on their effectiveness and reliability.	<u>Reactor Shutdown</u>: Two independent sets of control rod assemblies are employed for reactivity control and reactor shutdown. The primary system is composed of the ten (10) control rod assemblies. These control rod assemblies can be raised and lowered individually or in a bank configuration for normal operation, load following, and shutdown. The secondary system is composed of three symmetrically spaced mid-core assemblies; this set of control rods is a backup system to allow shutdown from any operating condition to hot standby.

2 Advanced Fast Reactor-100 - Information Tables for TRP Evaluation

Information Requested	Concept Description
	The control assemblies utilize conventional boron-carbide (B_4C) pins with enriched and natural boron. Natural boron was used for the central primary control rod assembly, while 90% enriched boron was used for the other primary control rod assemblies. For the secondary control rod assemblies, natural boron was used. The rod worth requirements allow for a single stuck rod (N-1) in either system. The drive system for the primary control rods is based upon the fully developed FFTF control rod drive system adapted for the AFR-100 design. Upon loss of power, the lead screw is unlatched and the control rods drop by gravity and spring assist. The secondary control rod drive system is mechanically different from the primary control rod drive system and is based upon technology developed for the CRBR and PRISM plants. The detailed design of the control rod structure and drive is varied between the two systems (e.g., different means for allowing insertion in distorted geometry) to avoid common mode failures of the two control systems. These drives and reactivity control systems were heavily tested for FFTF and CRBRP. <u>Normal Decay Heat Removal</u>: The normal coolant path to remove decay heat from the core is through the intermediate sodium loops and through the balance of plant system. Under normal operation at power levels above about 3 % nominal, heat from the core is transferred into the primary coolant, through the intermediate heat transport loops via the intermediate heat exchangers (twisted tube heat exchangers), to the sodium-to-CO_2 heat exchangers (compact diffusion-bonded heat exchangers) and into

2 Advanced Fast Reactor-100 - Information Tables for TRP Evaluation

Information Requested	Concept Description
	the S-CO_2 Brayton cycle system where it is rejected to circulating cooling water. For power levels below about 3 % nominal, transition is made from the S-CO_2 Brayton cycle to an independent supercritical CO_2 shutdown heat removal system. Each of the two intermediate sodium loops is provided with a S-CO_2 shutdown heat removal loop having its own sodium-to-CO_2 heat exchanger, CO_2-to-water cooler, and S-CO_2 pump. Each of the two S-CO_2 shutdown heat removal loops is sized to remove up to 5 % of the reactor nominal thermal power. Cross ties between the two shutdown heat removal loops enable one loop to also remove heat from the other intermediate sodium loop, if necessary. The shutdown heat removal loops are independent of the S-CO_2 Brayton cycle. For normal decay heat removal to work correctly, electrical power is required. <u>Emergency Decay Heat Removal</u>: The shutdown heat removal system is completely independent from the normal decay heat removal (which is through the intermediate heat transport system and balance of plant), and is activated passively only when the normal heat removal system is disabled or as needed during maintenance. The system consists of three redundant loops corresponding to 0.25% of core full power each. Only two loops are required for effective decay heat removal during emergency and ATWS events. Each loop consists of a submerged in-vessel direct reactor auxiliary cooling system heat exchanger (twisted tube heat exchanger) (DRACS), a secondary natural draft heat exchanger (NDHX), an expansion tank, and an exterior stack that forms the natural pathway for dissipating the decay heat to the air atmosphere. Two of the DRACS system can be operated in an active mode to allow

2 Advanced Fast Reactor-100 - Information Tables for TRP Evaluation

Information Requested	Concept Description
	for fine control of decay heat removal. The DRACS systems allow for passive decay heat removal. They are located directly in the sodium cold pool. Moreover, there are no valves or other mechanical devices to isolate the primary coolant. During full power operation, cold pool sodium circulates at a modest flow rate through the shell side of the DRACS. However, when fully activated, the DRACS sodium-to-air heat exchanger spring loaded dampers open, and buoyancy-driven natural convection flow of primary sodium through the DRACS is fully initiated. Inherent Safety Approach The AFR-100 design requirements specify the AFR-100 design shall have inherent means of net negative reactivity feedback and passive decay heat removal sufficient to place the reactor system in a safe stable state for anticipated transient without scram (ATWS) events with minimal damage to the core or reactor system structure.
3) Describe the expected response of the concept to normal and abnormal conditions. For example, describe the concept design and associated instrumentation that will provide operators with longer times than for current generation LWRs for system diagnosis of slowly evolving accidents before reaching safety systems activation and/or exposure of vital equipment to adverse conditions.	The mission of the AFR-100 is to provide electrical power at remote sites with minimal infrastructure support. Therefore, the AFR-100 is designed to simplify operator actions and interaction with the reactor control system. Favorable natural reactivity feedbacks are used to adjust to a range of operating conditions, and to prevent core damage in adverse conditions. The inherent safety capabilities of this approach were demonstrated in the landmark EBR-II passive safety tests in 1986. Furthermore, the primary system cold pool heat capacity greatly dampens any thermal transients in the plant providing time for the plant control system to take remedial action. Preliminary safety analysis indicates that the

2 Advanced Fast Reactor-100 - Information Tables for TRP Evaluation

Information Requested	Concept Description
	inherent neutronic, and passive thermal-hydraulic performance characteristics of the AFR-100 design provide self-protection in beyond-design-basis sequences to limit accident consequences even without activation of engineered systems or operator actions.
4) Describe the design features that will reduce the probability for accidents, including accidents with potentially severe consequences. These design features should provide sufficient reliability, redundancy, diversity, and independence in safety systems to provide for either accident prevention or accident mitigation.	The primary tenet of the natural inherent safety approach of the AFR-100 is to utilize design features for prevention of core damage and related consequences, even for beyond-design-basis events that involve the failure of multiple safety grade systems. In fact, the goal of the AFR-100 design is to protect the plant during these events such that it can be safely restarted thus protecting the plant investment. The preliminary safety analysis shows that the inherent neutronic, hydraulic, and thermal performance characteristics of the AFR-100 design provide self-protection to limit accident consequences *without activation of engineered systems or operator actions*. Specifically, the AFR-100 has the following design and inherent features that minimize the potential for severe accidents and their consequences: 1) Natural and inherent characteristics of the reactor system: small size, pool configuration, large thermal inertia of the primary heat transport system, very low system pressure, total negative reactivity coefficient; the pool configuration completely eliminates loss-of-coolant accidents. 2) Primary heat transport system is maintained even in a reactor vessel faulted condition. 3) Natural shutdown heat removal system provides emergency decay heat removal based upon natural convection flow

2 Advanced Fast Reactor-100 - Information Tables for TRP Evaluation

Information Requested	Concept Description
	through simple decay heat removal loops. 4) Containment function is provided by the upper containment structure, lower containment compartment, and the reactor guard vessel; the guard vessel will contain any potential leaks from the reactor vessel in the event that the reactor vessel develops a leak. 5) The use of the intermediate loop eliminates the potential of the primary sodium coming into contact with water or steam in the steam generator system in fault conditions. 6) Two independent, redundant, and reliable reactivity control rod systems. Although the AFR-100 has a positive void reactivity coefficient, it is recognized that the operating range of sodium-cooled fast reactors is at a very low pressure AND far below the boiling point (>882^{o}C) of the coolant. Indeed, the natural inherent safety approach prevents voiding even for the severe double-fault ATWS accident conditions.
5) Describe the design features that will minimize potential radiation exposures to plant personnel.	Routine radiological exposures should be reduced in the AFR-100 compared to conventional LWR systems. The limited database on SFR systems (e.g., PHÉNIX and FFTF) shows low typical exposures of 1-5 person-rem that increase up to 13-17 person-rem when non-routine maintenance was conducted.[2] Since refueling operations are not regularly conducted in AFR-100, this will significantly and further reduce radiation exposure.

[2] Operational exposures for advanced concepts are evaluated in the paper "Advanced Reactors and Associated Fuel Cycle Facilities: Safety and Environmental Impacts," Hill, Nutt, and Laidler, *Health Physics*, **100**, 20 (January 2011)

2 Advanced Fast Reactor-100 - Information Tables for TRP Evaluation

Information Requested	Concept Description
	The low radiation doses of the AFR-100 are attributed to low system pressure reactor vessel system with biological shielding incorporated into the reactor vessel head, coolant chemistry control including active cover gas cleanup (as necessary), a sealed coolant piping systems, and the use of a secondary cooling loop. Modern designs also employ remote operations and repair features to further reduce exposure. Because of the low corrosion rate of sodium with the structural materials of the AFR-100, there is little to no crud buildup within the primary system unlike in LWR plants. The impurities in the primary sodium are trapped in the simple cold trap or nuclide trap and thus there are no hot spots in the piping or equipment systems due to activated corrosion product buildup (as in an LWR). Most of the radioactive sodium is contained in the primary reactor vessel, and thus this is not a potential source of radiation exposure to plant personnel given the radiation shielding that is incorporated within the reactor vessel head. The small amount of primary sodium that is pumped from the reactor vessel and to the purification system is contained in shielded piping systems and shielded cells to minimize radiation exposure to personnel. The cold traps and nuclide traps contain integrated component shielding.
6) Describe how incorporation of defense-in-depth philosophy is accomplished in the concept design. Specifically, describe how the multiple barriers are maintained to prevent radiation release, and thereby reduce the consequences of severe accidents.	The AFR-100 safety design approach implements the “defense in depth” strategy by adopting the traditional three levels of safety. The <u>first level of safety</u> is, in part, the selection of fuel, cladding, coolant, and structural materials that are stable and compatible, and provide large margins

2 Advanced Fast Reactor-100 - Information Tables for TRP Evaluation

Information Requested	Concept Description
	between normal operating conditions and limiting failure conditions. 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's 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, eliminates loss-of-coolant accidents, and allows for shutdown heat removal by natural circulation. At the <u>second level of safety</u>, protection is provided by engineered safety systems for reactor shutdown, reactor heat removal, and emergency power. The <u>level 3</u> protections (against beyond-design-basis accidents) for cooling assurance and containment of radioactivity are provided by the reactor guard vessel and the reactor containment building. In case of a leak in the reactor vessel, the guard vessel will collect the leaking primary coolant and maintain sufficient primary coolant inventory within the reactor vessel to maintain the core and primary and decay heat transport paths covered in sodium and thus decay heat will be removed. The AFR-100 has three physical barriers against fission product release: HT-9 fuel cladding, primary containment boundary (e.g., reactor enclosure, IHX tubes), and secondary confinement (containment structure, guard vessel, and intermediate loop

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Information Requested	Concept Description
	piping).
7) Describe the features that could result in a large release of radioactive materials, such as those that would prevent a simultaneous loss of containment integrity (including situations where the containment is by-passed), and the ability to maintain core cooling as a result of an aircraft impact. If prevention of release is not possible under this scenario, identify system designs that would provide a delay in radiological releases to facilitate any required emergency response both on-site and off-site.	The AFR-100 is designed to withstand a directed plane strike on the containment building and survive. The AFR-100 containment building is similar in design to a hardened LWR containment building specifically designed to accommodate an aircraft impact. In addition, the radioactive materials are located in the reactor vessel or the spent fuel storage vessel which is located mostly below grade. There is redundancy in the decay heat removal systems to ensure that decay heat can be removed passively after an aircraft impact, if one of the systems is destroyed.
8) Describe the features that will prevent loss of onsite spent fuel storage capability and facility integrity (if part of the concept design), including consideration of an aircraft impact and other external events.	The AFR-100 core has a 30 effective-full power years core lifetime. The normal deployment of the AFR-100 will be to sites that do not have the infrastructure to support the long-term storage of spent nuclear fuel. The AFR-100 core will be allowed to decay in-situ within the reactor building at the core end-of-life until such time as the fuel is cool enough for removal from the primary reactor vessel and then stored in the fuel storage vessel located within the containment. The storage of spent nuclear fuel (and fresh nuclear fuel waiting for transfer into the reactor vessel) is protected from aircraft impact and other external events by the hardened containment of the reactor building.
9) Identify any R&D that would be needed to bring any of the safety-related technologies used in the design to a sufficient level of maturity to allow for industrial use.	The following R&D should be conducted to bring the safety-related technologies used in the design to a sufficient level of maturity to allow for industrial use: 1. Natural circulation performance of DRACS decay heat removal system. 2. Testing of primary and secondary control rod drive system. 3. Testing of seismic isolation system (as used). 4. Performance of advanced structural

2 Advanced Fast Reactor-100 - Information Tables for TRP Evaluation

Information Requested	Concept Description
	materials for vessels, piping, and heat exchangers. 5. Performance of intermediate heat exchanger technology. 6. Performance and failure modes of sodium-to-S-CO_2 Brayton cycle heat exchangers. 7. Others may be identified as the AFR-100 design matures
Category II – Safety	
1) Types of special nuclear materials present and the security features that provide SNM protection.	The core assemblies are the main special nuclear materials that are present in the AFR-100. The core assemblies contain enriched Uranium (U-235) and transuranics including Pu-239. Appropriate security and safeguards will be installed in the plant to ensure that fuel is protected from theft or diversion. The fuel is located in the primary vessel that is submerged in hot molten sodium. The core refueling system hardware is not located on site due to the long time frames between refueling events (30 years). Thus, there is no effective way for SNM to be removed or diverted from the reactor building. In addition, specialized security and safeguards features will be installed in the plant during refueling events. Fresh fuel can be stored in an adjacent storage vessel of similar construction to the reactor vessel. This storage vessel is located within the reactor building and is used to stage fuel to facilitate rapid refueling once the reactor is shut down for core change-out. Buildup of fission products in the used nuclear fuel makes the used nuclear fuel self-protective and unattractive for handling during theft due to the high gamma activity and a coating of sodium.

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Information Requested	Concept Description
	Reactor operation will be remotely monitored and data transmitted for review and evaluation to assure that no diversion of SNM takes place. Data includes inlet and outlet temperatures and video to assure that no transfers of SNM to or from the reactor occur without authorization and that used fuel is maintained in the cooling area inside of the reactor building. The AFR-100 may also incorporate remote monitoring using antineutrinos. This is a technology that is currently under development and testing, and it may be sufficiently mature when the AFR-100 is deployed. Antineutrinos are produced inside of a reactor core from the beta decay of neutron rich fission fragments from the fission of heavy elements including Uranium and Plutonium; about six antineutrinos are generated for each fission. The antineutrino energy spectra from fission of Uranium and Plutonium differ. Antineutrinos are not attenuated to any significant extent by shielding such that a scintillation detector might even be installed outside of the containment building, if the antineutrino flux at that location remains sufficiently high. The detection rate of antineutrinos thus provides a means to measure the core fission power and directly detect when the reactor is operating and at what power and when it is shut down. The antineutrino production rate depends upon the relative proportion of Uranium and Plutonium and provides a means to determine the production of Plutonium from the rate of change of the antineutrino counting rate. Thus, this approach, in principle, could reveal if Plutonium were being diverted and enable a determination of the burnup (i.e, the generation of higher Plutonium isotopes).

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2) Consideration of security requirements throughout the design process so that security issues (e.g., threats of theft, diversion, and sabotage) can be effectively anticipated, identified, and addressed at an early stage through integrated facility design and engineered security features, and through formulation of response/mitigation measures, ideally with reduced reliance on human actions compared with previous generations.	One of the objectives of the AFR-100 design is to demonstrate improved technologies for safeguards and security. The AFR-100 design requirements specify that the reactor shall be designed to minimize the risk of sabotage or proliferation, either through design features, or by proven safeguards and security techniques, or a combination of the two. Security is treated as an integral part of the design. The inherent and passive safety features of the AFR-100 offer a high level of protection against malevolent events, as well as against accidents. The location of the reactor vessel, the core, and the primary heat transport system below grade within a hardened strong containment structure provides protection against external events and threats. The industrial security and safeguards system of the AFR-100 is designed to protect plant equipment and personnel, and to prevent the theft of special nuclear materials. The system is designed to defend against the design basis threats specified in the regulations. The key requirements for the security and safeguards systems are: 1) allow plant access only to authorized personnel and material; 2) prevent the theft of special nuclear materials; 3) prevent the sabotage of critical plant equipment; and 4) deter, detect, and delay unauthorized activities and assaults on the plant. Adoption of the antineutrino remote monitoring system which can detect diversion of SNM from the reactor core represents the incorporation of an advanced technology early in the design process.
3) Features to prevent/mitigate sabotage threats e.g., loss of integrity of onsite core fuel and	The AFR-100 design takes advantage of the inherent properties of the sodium coolant,

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spent fuel storage, including consideration of an aircraft impact and other relevant attack scenarios.	metallic-fueled core, and pool-type primary system configuration to eliminate specific traditional accident initiators and significantly reduce the likelihood of radionuclide release from specific accident initiators followed by postulated equipment failures or operating errors. A high level of inherent/passive safety is achieved. This behavior also reduces the vulnerabilities to malevolent acts. Cooling of the reactor core following neutronic shutdown is not dependent upon the balance of plant as is the case for many existing Light Water Reactors (LWRs). Thus, damage to or destruction of the turbine generator building or any of the components housed inside of it would not result in inability to remove decay heat from the core. Cooling of the reactor core following neutronic shutdown is not dependent upon maintaining integrity of one or both of the intermediate sodium loops. Damage to or destruction of the intermediate sodium circuit would not result in inability to remove decay heat from the core. Decay heat can be removed through the three independent, NaK, natural circulation emergency decay heat removal loops. Each loop has a twisted tube, DRACS sodium-to-NaK heat exchanger immersed in the cold sodium pool of the reactor vessel and a natural draught NaK-to-air heat exchanger located at the top of a structurally hardened stack on the exterior of the containment building. Two of the emergency decay heat removal natural circulation loops are sufficient to remove decay heat from the core. Thus, damage to or destruction of one of the stacks would not result in the inability to remove decay heat from the core. The three stacks are separated 120 degrees azimuthally around the containment. In the event of an aircraft

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	impact on the containment, it is unlikely that more than a single structurally hardened stack would be damaged or destroyed. Upon reduction of heat removal from the primary sodium coolant system, the core autonomously shuts down the core power to match the heat removal due to the inherent strong reactivity feedbacks of the fast neutron spectrum metallic-fueled core and sodium coolant. Shutdown of the fission power is not dependent upon the reactor control system. Thus, damage to the reactor control system would not result in inability to shut down the reactor. The pool-type vessel configuration of the primary sodium coolant eliminates primary coolant system piping external to the reactor vessel and the possibility of loss of the primary sodium coolant due to damage to piping. The reactor vessel is surrounded by a steel guard vessel to maintain a sufficient sodium level inside of the reactor vessel in the event of reactor vessel leakage. Further reducing the sodium height inside of the reactor vessel would require penetration of both the reactor and guard vessels which are located below grade inside of the reactor cavity inside of the containment. The low pressure sodium coolant does not flash/vaporize upon loss of the integrity of the primary coolant system boundary as is the case for LWRs. Thus, penetration of the reactor vessel upper closure head would not result in the loss of primary coolant due to flashing, although a portion of the sodium could burn as long as there is access to oxygen at the sodium surface. However, the argon inert gas is heavier than air and thus will minimize and suppress any air ingress. As a result of the passive core cooling and

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	shutdown behavior of the core, damage to or destruction of the control room will not result in the inability to shut down the reactor and remove decay heat from the core. Inability to remove the core decay heat would require disabling of all three emergency decay heat removal loops and to not have electrical power available. In the aftermath of such a postulated attack scenario, heatup and boiloff of the sodium and core uncovering might still be avoided, if consideration were given to the installation of a hardened capability to flood the exterior of the guard vessel in a timely manner with water or a suitable coolant which could be provided via truck from outside of the containment. Removal of decay heat from used fuel that is temporarily stored in the storage vessel can also be assured by the incorporation of analogous independent, NaK, natural circulation emergency decay heat removal loops. The used fuel natural circulation emergency decay heat removal loops can also utilize the three structurally hardened stacks of the core emergency decay heat removal loops.
4) Features to eliminate or reduce the potential theft of nuclear materials.	The nuclear materials for the AFR-100 are located within the reactor core and in the storage vessel located within the reactor building. The safeguarding of nuclear materials is provided by the AFR-100's Industrial Security and Safeguards System. Interior and exterior security perimeter fencing provide passive barriers to reduce potential theft of nuclear materials. The security building provides controlled access to the plant site to prevent inadvertent access, industrial sabotage, or theft of nuclear materials. All personnel must pass through this building and be checked by the

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	associated security systems for ingress and egress to sensitive plant structures and areas, or areas where radioactive materials are stored. The plant security system is monitored and operated from this building. A truck trap is located adjacent to this building that allows for security force control and containment of trucks requiring access to the site for deliveries or pickups. Radiation portals and monitors are placed at strategic locations at the security building and around the perimeter of the site to detect the movement of radiological and nuclear materials. The AFR-100 has a long-lived core and thus nuclear materials are moved approximately every 30 years. although staging of fresh nuclear fuel will happen just before the required refueling event. During a refueling event, special equipment is deployed to the site to perform the refueling of the spent AFR-100 fuel from the primary reactor vessel and emplacement within the storage vessel. Without this specialized equipment, nuclear fuel cannot be removed from the reactor vessel or from the location where spent nuclear fuel is stored until it is cooled sufficiently and ready for on-site movement or off-site shipment. This specialized equipment consists of the in-vessel refueling machine which is used to transfer the nuclear fuel from the core to the in-vessel transfer position, the ex-vessel fuel unloading machine that is used to transfer the spent nuclear fuel from the in-vessel transfer position through the reactor vessel head and into the storage vessel, the same ex-vessel fuel unloading machine that is used to transfer the cooled spent nuclear fuel from the storage vessel to the on-site fuel transfer cask, the on-site fuel transfer cask that is used to transfer the fuel from within the

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	reactor building to the cleaning station and staging area where the spent nuclear fuel will be shipped to an off-site fuel supply station or reprocessing station, and the certified shipping cask that will be used for off-site fuel shipments.
5) Any features that will require R&D to bring to maturity.	The following system will require technology research and development to bring to maturity: 1. Advanced compact in-vessel fuel handling machine 2. Ex-vessel refueling machine 3. Remote Monitoring Technology 4. Under-sodium viewing technology 5. Non-destructive examination technology 6. Others may be identified as the AFR-100 design matures
Category III – Ability to Improve Uranium Resource Utilization and Minimize Waste Generation	
1) Uranium resource utilization	
a. Uranium enrichment required (compared to existing LWR systems)	The AFR-100 as currently designed uses low enriched metallic alloy uranium fuel, 13.5% average enrichment LEU. The AFR-100 is capable of using U-Pu-Zr, or a U-TRU-Zr based metallic fuel in the existing reactor configuration. It was decided that fueling the AFR-100 with LEU-based fuel initially would provide it with a higher fuel technology readiness level and better prospects for deployment internationally than fueling the reactor with Pu-based fuel that may be difficult to ship and transport to remote international locations.
b. Design features, if any, that reduce uranium consumption.	The fast neutron spectrum core with a higher enrichment than is typical of LWRs and a 30-year core lifetime requires a much larger fissile working inventory than LWRs but the initial loading of fissile is less than a LWR's consumption of U-235 over 30 years for the

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	same energy delivery.
c. Is the use of reprocessed fuel planned for the ARC system in its fuel cycle?	The base fuel for the AFR-100 is LEU U-Zr based fuel. However, the AFR-100 can use reprocessed fuel in the form of U-TRU-Zr, if necessary. In addition, the AFR-100 plant design can be converted from a U or Pu based core to a Th based core, if the thorium-based fuel is more readily available. [3]
d. What are the expected conversion ratio capabilities for the proposed ARC design? Can the ARC system be used for fissile material breeding?	The conversion ratio of the AFR-100 is ~0.80. The AFR-100 system can be used for fissile material breeding if needed. The conversion ratio is strongly dependent on the fuel form and the core configuration. The U-Zr binary metallic fuel is considered as the primary fuel form, while the U-Pu-Zr and U-TRU-Zr ternary metallic fuels are considered as the alternative fuel forms. Based on the flexibility of the fuel forms in the AFR-100, a wide range conversion ratio from a low value (~0.5, which is a burner) to a high value (> 1.1, which is a breeder) can be achieved without serious performance deterioration.
e. What are the R&D needs?	The AFR-100 was developed for aiming at a small grid, transportation from pre-licensed factory for affordable cost, and a long operation period (cycle length) without a frequent refueling period. The operating cycle length is dependent of the achievable burnup of the metallic fuel. The higher burnup fuel is better to increase the operation period without refueling. Thus, R&D on the high burnup fuel is needed. However, the major goals of the AFR-100 could be achievable based on current metallic fuel irradiation experiences. In addition, R&D is needed for the advanced shielding materials within the core.
2) Estimate of waste generation (qualitatively compared to a once-through LWR).	

[3] Include the Thorium study citation.

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a. Ability to transmute long-live products in spent fuel; those produced in situ during reactor operation.	The fission product production rate (i.e., FP mass per unit energy generation) is comparable regardless of the neutron spectra in reactors. However, the fast spectrum reactor is favorable to transmute the long-lived higher actinides because of the higher fission-to-capture ratios of most isotopes.
b. Mass (qualitatively compared to an LWR discharge) of high longevity/high heat materials (example transuranics) requiring long-term geologic isolation.	For the once through option, the spent fuel mass is inversely proportional to the discharge burnup. The discharge burnup of the once through AFR-100 option is a factor of two to three higher than that of a typical LWR. So, the generated TRU mass is much smaller than an LWR. The disposal of TRU (except for the losses during reprocessing) could be avoided when the continuous recycling of the AFR-100 spent fuel is allowed.
c. Mass (qualitatively compared to an LWR discharge) of low heat, long-lived materials (examples Carbon 14, Technetium 99, Iodine 129)	The production rate of Tc-99, I-99 could be different depending on the neutron spectrum and fissile isotope. However, in general, the masses are comparable regardless of the neutron spectra in reactors.
d. Mass (qualitatively compared to an LWR discharge) of low heat, low longevity materials (Class A, B, C, low-level waste (LLW)).	The generated LLW is generally comparable between the once-through and continuous recycling fuel cycle options because the D&D is the major contributor to the LLW volume. Thus, it is expected that the generated LLW from the AFR-100 fuel cycle is comparable to that from the LWR.
e. R&D needs to facilitate transmutation or other waste management goals.	High burnup metallic fuel to maximize the operation period without refueling frequently. Primarily, advanced cladding materials that can withstand high irradiation levels at reactor temperatures. R&D on metallic fuel separation is also needed when the recycling of AFR-100 used fuel is considered.

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Category IV – Operational capabilities and aspects such as control strategies, operating modes (e.g., base load versus load following capability), maintenance and inspection requirements, refueling interval, etc.	
1) Electricity generation capabilities, including flexibility in electricity generation including the proposed concept capability for load following (the capability to adjust generation as demand for electricity fluctuates). Limitations, if any, on such operation arising from considerations of fuel performance, reactivity limitations, mechanical and thermal stress in materials and components should be addressed.	The AFR-100 is designed to provide electricity on the grid or electricity at remote sites that are off-grid. The AFR-100 with its S-CO_2 Brayton cycle power converter can be operated in either a base-load electricity production mode or in a load following mode from 10% to 100% electrical power production capacity.
2) Other features allowing utilization beyond base-load electricity production, for example process heat generation, high temperature operation for hydrogen production via chemical splitting, etc.	The current maximum core outlet temperature of the AFR-100 is 550°C. This outlet temperature can provide heat for chemical and other processes beyond just electricity production. Examples of these process heat applications are: Besides electricity production, nuclear energy is an excellent source of process heat production for various industrial applications. Depending on the output temperatures, these applications include district heating, water desalination, synthetic and unconventional oil production, oil refining, biomass-based ethanol production, and in the future hydrogen production. Different designs of reactors produce the required temperatures for the various processes. The lower range up to about 200 to 300° C includes industries such as seawater or brackish well water desalination, pulp and paper, or textiles. Chemical industry, oil refining, oil shale and sand processing, and coal gasification are examples of industries with temperature requirements of up to the 500 to 600° C level. Desalination to produce potable water is a

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	natural application for the AFR-100 with its S-CO_2 Brayton cycle power converter. The S-CO_2 exits the low temperature recuperator and enters the cooler at a temperature of 90 °C. Instead of rejecting heat from the S-CO_2 at this temperature and below to circulating water, a portion of the thermal energy of the CO_2 can be utilized as the heat source for a desalination plant. If desired, the recuperator design could be modified to provide a greater exit temperature.
3) Features expected to improve availability in operation as estimated from the system's capacity factor, frequency of outages for refueling, and other planned outages (compared with those for a LWR). Include in the analysis of operational availability/dependability, elements from the information requested under Category V that follows, which includes concept maturity and operating experience (if available) associated with the proposed concept.	The AFR-100 will have the capability to achieve a capacity factor that is greater than 90%. The fuel cycle length is very long – 30 effective full power years – and thus the only reason for taking the plant down is for maintenance on the nuclear and non-nuclear systems and components. The long fuel cycle length was selected specifically to allow the AFR-100 to provide power to the grid in a dependable and reliable manner with very little interruption. In-service inspection approaches for the AFR-100 are established based on the following general guidelines, which are consistent with the ASME code Section XI: In-service inspection will be conducted during refueling or service outage; available examination technologies are adopted; access provisions are provided for all components of the reactor system; design provisions are made to reduce possible areas and modes of failures; all liquid metal and cover gas containment boundaries will be continuously monitored for leakage; continuous monitoring of reactor operating variables will be utilized to supplement or replace other inspection methods, whenever possible; and problem areas observed during the normal in-service inspection will be further investigated.

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	The in-service inspection approaches cover both normal and supplemental inspections. The supplemental inspections may be follow-up examinations in the event the normal inspections indicate deviations from normal operating or structural conditions.
4) Maintenance and Operation – Are there features that will allow easier maintenance to reduce that duration and frequency of outages. Are there special requirements for maintenance and inspection that are different from current LWRs (simpler or more complex)?	The technical requirements of the AFR-100 specify that: the plant operating procedures and diagnostics shall be automated to the extent required to minimize the operating and maintenance (O&M) cost, public risk, and operator exposure to radiation; the design shall minimize required maintenance and facilitate maintenance when needed; the design shall minimize the manpower required for maintenance and minimize the skills required to keep the plant maintained; all systems and components shall be designed so that routine maintenance activities may be performed during operations; and remote maintenance shall be minimized. Ease of operation, inspection and maintenance will be an important factor in making design decisions. Because of some basic differences between LWRs and the AFR-100, e.g., use of sodium and pool configuration for the AFR-100, there are special requirements for maintenance and inspection for the AFR-100 such as inspection of the outer surface of the reactor vessel and inner surface of the guard vessel, performing necessary ISI of the interior structures of the reactor vessel, and handling of sodium during refueling or sodium component repair (when needed). The operation and maintenance of a sodium-cooled fast reactor, such as the AFR-100, may initially appear more complex than an LWR, but personnel who have operated EBR-II and FFTF claim from their

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	experience operating both types of reactors, that operating and maintaining sodium-cooled fast reactors are actually simpler than LWRs for many reasons. However, the AFR-100 does have some innovative systems and components which will require technology maturation and demonstration such as the S-CO_2 Brayton Cycle energy conversion system, primary electromagnetic pumping, advanced refueling technologies, among others. The technical requirements specify that: the plant operating procedures and diagnostics shall be automated to the extent required to minimize operating and maintenance (O&M) cost, public risk, and operator exposure to radiation; the design shall minimize required maintenance and facilitate maintenance when needed; the design shall minimize the manpower required for maintenance and minimize the skills required to keep the plant maintained; all systems and components shall be designed so that routine maintenance activities may be performed during operations; remote maintenance shall be minimized; and the maintenance plan shall ensure that all failures with a probability of occurrence greater than 10^{-4} per reactor year shall have a specific plan for corrective action. Failure with a probability of occurrence of less than 10^{-4} events per reactor year shall have a feasible means of corrective action.
5) Describe design efforts to provide reliable equipment in the BOP to increase the reliability of safety systems (or safety-system independence from BOP).	An AFR-100 technical requirement states: the plant design shall include reliable equipment for the balance of plant (or safety-system independence from balance of plant) to reduce the number of challenges to safety systems. The adopted balance of plant is the innovative S-CO_2 Brayton cycle that is

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	currently under development by the U.S. Department of Energy. The Brayton cycle system does not provide a safety function for the reactor plant and is independent from the safety systems of the AFR-100. However, the reliability of the S-CO_2 Brayton Cycle system will be demonstrated as DOE matures the technology. In addition, the S-CO2 Brayton cycle is expected to be a simpler overall system compared with the steam cycle (backup option for AFR-100) because it is a single phase fluid system.
6) R&D needs to achieve reactor performance goals.	The thermodynamics of the S-CO_2 Brayton cycle is currently being demonstrated at small-scale (0.78 MWt) heat input. At this scale, the turbomachinery (turbines, compressors, and alternators) are subject to severe scale distortions requiring the application of special turbomachinery technologies that are not representative of those at full scale (e.g., 100 MWe). Analysis of the results from such small-scale testing also involves modeling challenges and uncertainties, in part, due to the lack of specific instrumentation on the loop. The S-CO_2 Brayton cycle needs to be demonstrated at larger scale with the same turbomachinery technology (e.g., oil-lubricated fluid film tilting pad bearings, dry gas liftoff seals, axial flow turbine) that would be incorporated in the full-scale system. Ongoing R&D focused upon the reliable design of compact sodium-to-CO_2 heat exchangers needs to be completed to provide an understanding of sodium plugging phenomena in small sodium channels, the ability to drain and refill small sodium channels, as well as the stresses and potential damage from inadvertent freezing and remelting of the sodium. Ongoing tests on interactions between sodium and CO_2 and

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	sodium and sodium-CO_2 reaction products under conditions typical of postulated compact sodium-CO2 heat exchanger failure need to be completed. Tests of the thermal shock response of compact sodium-to-CO_2 heat exchangers need to be carried out such that the thermal shock phenomena can be reliably predicted and incorporated into the design process.
Category V – Concept maturity, operating experience, unknowns and assumptions (e.g. availability of advanced materials, fuels, etc. currently under development)	
1) Description of the general level of the concept design maturity (pre-conceptual, conceptual, or detailed). Identify the proposed schedule for completion of the design, duration of construction (as a function of availability of funds, if appropriate), and initial operation of the proposed concept.	The general level of the AFR-100 design maturity is a pre-conceptual design. The design is considered an evolution of former U.S. fast reactor plants and designs including EBR-II, FFTF, CRBRP, LSPB, PRISM Mod A and B and thus contains technology that has a greater design maturity than pre-conceptual, however, in accordance with DOE O 413.3, the stage of the design is considered pre-conceptual.
2) Description of Technology Readiness Levels (based on DOE TRL definition in DOE G 413.3-4, U.S. Department of Energy Technology Readiness Assessment Guide) of major technologies and system and their relation to previous operating reactors. Identify the overall TRL of the proposed concept, which should be based on the TRL of the least ready major technology or system.	The lowest TRL level for the various systems and components is TRL 3 - Active research and development (R&D) is initiated. TRL 3 (This includes analytical studies and laboratory-scale studies to physically validate the analytical predictions of separate elements of the technology. Examples include components that are not yet integrated or representative tested with simulants. Supporting information includes results of laboratory tests performed to measure parameters of interest and comparison to analytical predictions for critical subsystems. At TRL 3 the work has moved beyond the paper phase to experimental work that verifies that the concept works as expected on simulants. Components of the technology are validated, but there is no attempt to integrate the components into a complete system.

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	Modeling and simulation may be used to complement physical experiments. So, the AFR-100 is at TRL 3 due to the various innovations that have been incorporated into the overall plant to increase its performance, effectiveness, and economics.
3) Applicable experience from other reactor systems (test, research, demonstration reactors, naval reactors, foreign reactors) such as design elements, component testing and demonstration.	The AFR-100 has its roots in the various research and test reactors developed by the U.S. DOE and its predecessor agencies. The design experience and lessons learned from previous fast reactor systems such as EBR-II, FFTF, CRBR, LSPB, SAFR, PRISM Mod A and B have played a major role in the design evolution of the metallic fueled, sodium-cooled AFR-100. For example, metal fuel design and analysis is based upon EBR-II experience. The approach to designing an inherently safe sodium-cooled fast reactor is based upon experience from EBR-II. Core restraint design and analysis is based upon work conducted for FFTF and CRBR. Reactivity control systems are based upon research and development work conducted during the FFTF, CRBR, and ALMR program. Balance of plant technology is based upon work currently being conducted by U.S. DOE.
4) Status of applicable design and analysis tools, including safety analyses.	The AFR-100 design has used computer codes that are maintained by the DOE National Laboratories, such as DIF-3D, REBUS-3, ETOE-2, MC2-2, VARI3D, ORIGEN-2, TWODANT, and NJOY. The SAS4A/SASSYS-1 computer code was used to assess the AFR-100 system performance. Many of these codes have existed for a long time and their capabilities have been well established. In addition, the NUBOW-3D code was used to analyze the core restraint system design. ANSYS has been used for structural and stress analysis of components. SOFIRE-II has been used for sodium fire

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	analysis. The S-CO_2 Brayton cycle for the AFR-100 has been designed using the Plant Dynamics Code (PDC) developed at ANL especially for S-CO_2 cycle steady state design and performance analysis and system level transient analysis. The PDC has been used to develop an automatic control strategy for the S-CO_2 cycle coupled to the SFR system. The code is in the process of being validated through on-going analysis and comparison with the small-scale S-CO_2 cycle tests.
5) Discussion of the assumptions made regarding the expected concept performance (associated with unique or unproven aspects of the design) and the basis of those assumptions, including identification of uncertainties.	For the reactor core, the expected performance and assumptions are provided in the AFR-100 report, with regard to the core parameters, conversion ratio, and performance and preliminary safety analysis. These analyses were performed using the analysis tools discussed in Item 4) above. The AFR-100 uses twisted tube heat exchanger technology. This technology has been used in the chemical industry, but has not been deployed in the sodium fast reactor technology. The analyses for the sizing and performance of the twisted tube heat exchangers is based upon Russian literature, but will have to be confirmed by testing and evaluation at the appropriate scale. The AFR-100 uses electromagnetic pump technology for the primary and secondary heat transport systems. This technology is based upon the technology developed for the PRISM reactor concept during the Advanced Liquid Metal Reactor program. Significant research and development was conducted on this technology including the construction and testing of a single stator EM pump that was tested at Argonne and a double stator annular linear induction pump tested at the Energy Technology Engineering Center

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	(ETEC). However, EM pump technology will still need to be matured to the degree that its long term performance will be understood in the primary sodium environment where it will be under high temperatures and neutron and gamma radiation damage. The AFR-100 uses a passive core restraint design that is based upon the work conducted at FFTF and CRBRP. However, the overall core reactivity response during overheating events will need to be confirmed by analysis and/or testing.
6) Identification of major technology issues, R&D needs to address design and operational uncertainties, and technology gaps.	The following system will require technology research and development to bring to maturity: 1. Advanced compact in-vessel fuel handling machine 2. Ex-vessel refueling machine 3. Remote Monitoring Technology 4. Under-sodium viewing technology 5. Non-destructive examination technology 6. Advanced Structural materials 7. Advanced compact Na-to-Na heat exchangers, Na-to-NaK heat exchangers, Na-to-S-CO_2 heat exchangers 8. Advanced S-CO_2 Brayton Cycle technology 9. High temperature metallic fuel 10. Vented fuel
7) Estimated time frame to develop the needed information identified in Item 6 above.	The goal of DOE's R&D program is to have technologies ready for commercial deployment by the 2050 timeframe. Thus, the technologies required for the AFR-100 will need to be matured from the TRL3 technology readiness level to TRL8 or 9 in approximately 30 years.

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Category VI – Fuel Cycle Considerations	
1) Ore mining and conversion requirements (qualitatively compared to the once-through LWR cycle, including enrichment)	For the once-through option, the AFR-100 needs about 13.5% enriched uranium, which requires a factor of 3.2 more natural uranium compared to LWR. However, for a recycling option, the required natural uranium drops by a factor of 20 compared to LWR's.
2) Fuel fabrication (compared with LWR fuel).	The AFR-100 uses binary (U-Zr) metallic fuel. The core can utilize U-Pu-Zr and/or U-TRU-Zr fuel as necessary. although the decision was made to initiate the AFR-100 on U-Zr fuel. The fabrication process for the AFR-100 metallic fuel is much different from the LWR oxide fuel. Metallic fuel is fabricated using a melting and casting technology such as a casting furnace and could be based upon the casting furnace technology developed and used at EBR-II during its 30-years of operations. The fabrication of metallic EBR-II fuel was well developed during the 30-years of operations of EBR-II.
3) Fuel form experience base (as needed for licensing/certification) for fuel forms different from current UO2 fuels (LWRs, HWRs, etc).	More than 130,000 metallic rods/pins were irradiated in EBR-II and FFTF and qualified to 10 % burnup, demonstrated to 20 % burnup with HT9 (or D9) cladding up to 31% Pu loading
4) Are the systems currently used for managing used fuel/waste in LWRs applicable?	Used LWR nuclear fuel is typically discharged from the reactor vessel of the LWR and placed into a spent fuel pool where it decays in storage for many years until it is transferred to dry cask storage. Used SFR fuel is discharged from a reactor and depending upon its decay heat load, may require storage in a vessel filled with sodium before it can go to inert gas or air storage. Similar dry cask storage is an option for the AFR-100 spent nuclear fuel once the sodium on the external surfaces of the spent core

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	assemblies is removed. No, but the AFR-100 can be used to support managing the LWR used fuel with its recycling option.
5) Is a reprocessing capability required? If yes, what type of technology is needed? Has it been proven/demonstrated? And what are the waste forms?	For the AFR-100 U-Zr core, spent nuclear fuel reprocessing technology would not be required to fuel the reactor core (this is similar to the current situation with LWRs). However, reprocessing technology would be required to reprocess the spent nuclear fuel coming from the AFR-100 and to produce the materials required for a Pu or TRU based core or for recycle of the useable fissionable materials to another AFR-100. The AFR-100 could also be operated in a once-through cycle if needed. AFR-100 can adopt either once-through or recycling options. For the recycling option, a reprocessing capability is required. In particular, electrochemical separation (pyro-processing) technology is required and the pyroprocessing technology was proven and demonstrated at both laboratory and engineering scales. If pyroprocessing technology is used, then the typical waste forms generated from the pyroprocessing technology are a metallic waste form consisting of the cladding that surrounds the metallic fuel and a ceramic waste form that contains the non-recyclable salt and fission products generated from the pyroprocessing technology. Purified heavy metal is generated from this process and is sent to a casting furnace for melting, casting into fuel slugs, and fabrication into pins for use in another fast reactor or an AFR-100.
6) Discuss any unique features/aspects of processing/storage/transportation of used fuel, high level waste (HLW), or LLW.	So, the AFR-100 plant will generate spent nuclear fuel and some low level waste, but no high level waste (from reprocessing). The spent nuclear fuel exiting the reactor vessel

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	will require cleaning to remove the sodium that will be wetted to the surface. The spent fuel can be stored in an air atmosphere after it is removed from the reactor vessel and the residual sodium has been removed. So, dry storage, pending shipment to a reprocessing facility, is an acceptable method for storing the spent nuclear fuel. Transportation of the spent nuclear fuel will be via an approved shipping cask that is suitable for this fuel type.
Category VII – Assessment of market attractiveness (e.g., efficiency, revenue generation benefits, application beyond electricity generation, etc.)	
1) Energy products of the concept (e.g., electricity production, desalination, process heat, hydrogen production, etc.) and its power (thermal, electric) output and/or product output.	The primary mission of the AFR-100 is to produce electricity on the grid at locations that do not require a large monolithic nuclear reactor. The AFR-100 with a core outlet temperature of 550 ^{o}C can provide fresh water through desalination processes, process heat for various applications, power for hydrogen production, and other non-electrical uses of this nuclear heat source. The core power level is ~250MWth and the electrical power generated is 100MWe.
2) Expected thermal to electric conversion efficiency, and overall multi-use plant efficiency	The core power level of the AFR-100 is 250MWth and the electrical power generation is 100MWe. The overall electric conversion efficiency with the S-CO_2 Brayton cycle power converter is 42%. The overall multi-use plant efficiency would be higher. In order to determine the overall multi-use plant efficiency (electricity and transmutation of TRU), it is necessary to analyze the cost and effort required to dispose of LWR TRU, including that of permanent spent-fuel disposal in geological formation, the energy value gained by conversion in fast reactors, and the resulting value of the spent fuel which will be discharged from the AFR-100. This is an

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	analysis that compared the once-through fuel cycle with a closed fuel cycle. Such an analysis was not performed at the design level of the AFR-100.
3) Revenue generation benefits or advantages	The AFR-100 will provide electrical power or process heat at remote sites without access to a major electrical grid thereby removing dependencies upon generators that require the frequent resupply of fuel (e.g., diesel generators dependent upon diesel fuel transport by aircraft). The load following capability of the AFR-100 makes it ideal for such a remote small grid on which it is the major, if not the only, power generator. The 30-year core lifetime eliminates the need for frequent fuel resupply. The remote sites are locations at which consumers are used to paying relatively high electricity rates. Once the AFR-100 is constructed and generating power on the grid, the price of the fuel (which is currently low for nuclear reactor fuel) will have been paid for over the 30-year effective full power years of the AFR-100 operations. The overall plant is designed for a 60-year life such that the overall initial plant investment will generate power for 60 years. Thus the revenue generation benefits are high.
4) Estimated siting requirements (e.g., less water usage or accident consequences may favorably impact siting requirements).	The AFR-100 is designed to produce electricity and will impose minimal siting requirements. Using the S-CO_2 Brayton Cycle technology, the AFR-100 can be adapted to dry heat rejection, although that will impact somewhat the thermal efficiency of the overall thermodynamic cycle. The AFR-100 is designed to accommodate the most severe of reactor accidents while reducing the risk of core degradation to the degree that it is in the residual risk category. The plant is designed to accommodate these very severe accidents to the degree that the plant investment is protected and will be able to be restarted.

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	The design requirements specify that as a minimum, the safety grade portions of the AFR-100 plant shall be designed to withstand a 0.2 to 0.3 g safe shutdown earthquake (SSE) seismic event without loss of function. The minimum operating basis earthquake (OBE) design value shall be ~1/3 of the SSE value. The facility will also need to have an electrical power grid infrastructure that supports a 100MWe power reactor, but the assumption is that this infrastructure would already exist for the extant high cost electrical generation equipment in the region.
5) Environmental impacts under normal/abnormal conditions, including severe accident conditions, and from spent fuel arrangements (as compared with current LWRs).	The AFR-100 design will meet the siting regulations specified in NRC's 10CFR100. It is expected that the environmental impact of the AFR-100 per MWt generated will be less than the current fleet of LWRs due to the robustness of the design and its ability to accommodate very severe accidents (such as a full-scale station blackout) with investment protection.
6) Competitiveness or international markets/export potential. Specifically, what concept features could make it desirable to a foreign customer?	It is expected that the AFR-100 will be highly competitive in the SMR international and export markets where the current electrical or power generation is from carbon based fuels. Foreign customers are interested in reliable, dependable, and cost effective power that is sustainable and very safe. The AFR-100 concept will deliver electrical power for 30 effective full power years without refueling in a very safe stable reliable manner and can be used to not only produce electricity but other energy-related products as needed such as fresh water from desalination processes. In addition, the AFR-100 has the potential to provide very stable power to military bases and other critical facilities and infrastructure

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	that require long-term stable electrical power. The AFR-100 is designed to provide safe, clean, affordable and proliferation-resistant nuclear power to energy markets in both the developed and the developing world. The non-proliferation features of the ARC reactor make it suitable for deployment anywhere in the world.
7) Derived technologies arising from concept development.	Advanced fuel handling systems and equipment that can be used for other Advanced SFRs. Advanced Brayton Cycle Technology – if the advanced Brayton cycle energy-conversion cycle can be proven and demonstrated for the AFR-100 application, it could serve in other energy-related industries such as the power converter for fossil energy and concentrated solar power to increase power generation efficiencies of those power plants and reduce the cost of the balance of plant equipment due its compactness. It can also be utilized for waste heat recovery from conventional power plants thereby increasing the efficiency of those plants. It may also contribute to a perception of safer sodium-cooled fast reactors by eliminating the potential of sodium-water interaction, which could occur in reactors using a conventional Rankine steam cycle energy conversion system. Advanced self-cooled electromagnetic pumps that can be useful for other technologies. Advanced structural materials that can be used for other high temperature applications. In-service inspection and repair technologies for liquid metal applications

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8) Unique features	The AFR-100 has the following unique features: 1. Metallic fuel that can be recycled increasing the utilization of uranium resources by a factor of fifty or more and operates for 30 years without refueling. 2. Compact sodium-cooled pool plant geometry which allows 100% of the primary heat transport system to be completely submerged in one single vessel alleviating loss-of-coolant events 3. Self-cooled EM pumps used in the primary heat transport system. These EM pumps are more reliable and require less maintenance compared with mechanical pumps. 4. Passive decay heat removal that does not require use of IHTS or balance of plant. 5. Passive core restraint system that provides proper reactivity feedback during overheating events. 6. Compact refueling system that reduces the overall size of the reactor vessel. 7. Compact IHX and DRACS heat exchangers that use the twisted tube heat exchanger technology. 8. Advanced structural materials that minimize the amount of steel and other materials used in the AFR-100 SSCs. 9. Advanced shielding materials that reduce the overall size of the core. 10. Compact heat exchangers used between the IHTS and the S-CO_2 BOP. 11. The supercritical CO_2 Brayton cycle energy conversion system. 12. Sodium coolant which allows for high temperatures, very compact core, very large core power densities, very low pressures compared with LWR technology.

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9) Expected time frame of introducing the concept to the market	The AFR-100 is at a pre-conceptual design stage but includes a lot of technology that has already been demonstrated. With the innovations included in the AFR-100 to reduce its cost and increase its competitiveness, it is expected that with a concerted level of focus, effort, and funding, that the AFR-100 could be introduced into the market in 15-20 years.
Category VIII – Economics (including construction, manufacturing, and operating costs)	
1) The materials and features of proposed modules that would improve the concept economics.	The advanced technologies adopted for the AFR-100 are chosen to reduce the cost of the AFR-100 or to increase its performance. Either way, the chosen advanced technology improves the economics beyond the conventional fast reactor technology. See above for the advanced technology options.
2) Concept improved economics by design compared with LWRs (e.g. length of piping, electrical cables, values, number of loops, pool design, etc.	It is expected that the economics of the AFR-100 would be similar to or smaller than a comparably sized LWR. It is expected that the AFR-100 will have improved costs for operations and maintenance given the fact that the core is a long-lived core and processes and equipment are relatively simple compared with LWR technology. In addition, it is expected that the Brayton cycle technology will be simpler to operate compared with a conventional steam plant once the technology has been developed.
3) Cost of nuclear fuel	Based upon discussions at Argonne, it is expected that the cost of the fuel will be in the $5,000 per kg rate. We expect that the cost to manufacture metallic fuel will be less expensive than an oxide based fuel.
4) Cost of major components. Need for special materials and/or construction methods (how many large vessels and pipes have to be fabricated, and how large would they have to be?)	The cost estimates for the major areas are located in the Item #7 below. We don't currently have a cost estimate by components. The technical hardware and various systems and components will be fabricated in suitable modules and shipped to

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	the site for installation. Construction of the nuclear island, balance of plant building, and other major structures will be part conventional construction and facility pre-fab equipment.
5) Estimated construction schedule (as compared with LWRs).	Because of the small reactor plant size and the fact that most of the equipment will be fabricated in a factory, it is expected that the first plant will take approximately 5 years to construct. However, the Nth of a kind reactor plant will take approximately 3 years to construct once the skills of the site and factory engineers and workers improve. We expect the Nth of a kind reactor to be the fourth or fifth reactor constructed.
6) Need for special skill sets and/or procedures required for construction (and their availability) and operations (and their availability).	The AFR-100 will need the same skill sets and procedures for the construction as those used for constructing an LWR. Historically, the operations and maintenance personnel who operated the EBR-II and FFTF reactors came from the U.S. Naval Nuclear Program. There will be special procedures for the operations and maintenance of the AFR-100 that will be unique to the AFR-100 compared with LWR technology.
7) Estimated overnight capital cost	The anticipated overnight capital cost for the AFR-100 based upon the PRISM prototype cost estimate with adjustments for power level and inflation is broken down as follows: Land and Land Rights – 11.2M$ Structures and Improvements – 84.03M$ NSSS Equipment – 274.05M$ Reactor Plant Equipment (less NSSS) 21.61M$ Turbine Plant Equipment – 36.87M$ Electric Plant Equipment – 29.90M$ Miscellaneous Plant Equipment – 14.61M$ Heat Rejection – 4.65M$ Construction Services – 45.31M$ Engineering and Home Office Service – 44.22M$ Field Office Services – 19.42M$

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	Contingency (30%) – 248.34M$ Total – 1,146M$ $/KWe – 11,460 dollars/KWe This cost does not include the cost for designing and licensing the AFR-100 through detailed design which is estimated to cost about – 1.088M$ The first of a kind cost (different from the first prototype) and Nth of a kind costs are expected to be about $1,058.5M (FOAK) and $0.8379M (NOAK) based upon PRISM cost figures corrected for power and inflation.
8) Estimated yearly operational cost (accounting for decommissioning and waste management)	This figure has not been determined.
9) Estimated cost of electricity	This figure is has not yet been determined.
Category IX – Potential Regulatory Licensing environment (advantages and uncertainties/risks)	
1) A description of the licensing approach envisioned for the proposed concept. This would include the general applicability of current regulatory requirements (10 CFR50, 10CFR52) and guidance documents (e.g., NUREG-0800 and Regulatory Guides) to concept design, construction, and operating licensing.	The approach to licensing of the AFR-100 would be to first obtain a license from the NRC for a prototype demonstration of the technology including full power inherent safety testing. Once the prototype AFR-100 was successfully demonstrated, there would be licensing for a standard plant – AFR-100.
2) Concept design/operational features that may positively impact licensing requirements (enhanced passive safety, low-pressure operations, etc.)	The AFR-100 reactor and the primary and intermediate heat transport systems are located mostly below grade level. All of the nuclear and some of the non-nuclear components are located on a seismically-isolated basemat (as an option) called a nuclear island which can isolate the nuclear structures from large seismic events. The primary heat transport system is configured as a pool-type arrangement (the arrangement is similar to that used

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	successfully in EBR-II, PHÉNIX, Superphenix, PFR, and the BN600 reactor plants) with the reactor core, primary pumps, intermediate heat exchangers, and direct auxiliary reactor auxiliary cooling system (DRACS) heat exchangers all immersed in a pool of sodium within the reactor vessel. The pool-type arrangement was selected because of its inherent simplicity and safety. All primary coolant piping is within the sodium pool, which greatly reduces the possibility of loss of coolant, and the sodium pool provides a large thermal inertia in the system. The pool system generally is less sensitive than the loop system to plant upsets associated with loss of forced cooling. This reduced sensitivity allows greater opportunity for inherent passive safety features such as natural convective flow and negative reactivity feedback to protect the core. The major design considerations relate to the thermal inertia, the rate of pump coast down, and the reactivity feedback characteristics of the system. In addition, the reactor vessel is a simple structure having no penetrations. The hot sodium at the core outlet temperature is separated from the cold sodium at the core inlet temperature by the core cover and outlet pipe. The reactor vessel is exposed only to cold sodium throughout its service life, so it is not subjected to severe thermal transients. A guard vessel surrounds the reactor vessel and is provided as an additional passive safety feature in case of a leak of the reactor vessel. If the reactor vessel were to leak, the resulting “faulted level” in the primary reactor vessel will ensure that the core decay heat can be rejected from the vessel. Thus, all

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	of the primary heat transport system and emergency decay heat removal components are located below the faulted level to ensure that the core decay heat can be rejected during a faulted condition. The use of a supercritical carbon dioxide Brayton energy conversion system offers improvements to safety because the potential for a sodium-water chemical reaction is eliminated. In an emergency, natural convection flow can remove the decay heat. There is no need for pumps or electricity for emergency decay heat removal. The reference design provides for in-vessel fuel storage cooled by natural convection alone.
3) Concept design/technology/operational features that have not been subject to the licensing process for the current fleet of LWRs.	The AFR-100 is similar technologically to the Sodium Advanced Fast Reactor (SAFR) and PRISM reactor plants. During the Advanced Liquid Metal Reactor program, preliminary safety information documents (PSIDs) for SAFR and PRISM were submitted to the Nuclear Regulatory Commission for review. Preliminary Safety Evaluation reports were prepared by the NRC staff that commented on the design status at that time. The design and operational features of the AFR-100 are similar in nature to SAFR and PRISM, but are substantially different from LWR technology. The AFR-100 has not been submitted to the NRC for licensing review.
4) Applicability of current codes and standards and possible development required.	The AFR-100 design shall comply with all applicable codes and standards issued by the American National Standards Institute (ANSI), the American Nuclear Society (ANS), the American Society of Mechanical Engineers (ASME), and the Institute of Electrical and Electronics Engineers (IEEE).

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	During the years of the U.S. Sodium-Cooled Fast Reactor development program, industry representatives developed an American National Standards Institute standard, ANSI/ANS-54.1-1989, for safety design criteria applicable to liquid metal-cooled reactors. This standard is in the process of being updated. This set of criteria followed the organization and intent of 10CFR50 Appendix A, but modified certain criteria details for applicability to the low pressure, chemically reactive liquid metal coolant environment in an SFR. There is an ongoing effort within the ANS to update the ANS-54.1 standard. Along with this effort, there is a separate effort at U.S. DOE to create Advanced Reactor Design Criteria (ARDC) that would facilitate the licensing of the AFR-100. The U.S. Department of Energy has developed a set of safety design criteria to apply to DOE reactors that are exempt from NRC regulation. Such reactors include one-of-a-kind designs built for research, and other special purpose reactors. The DOE criteria are similar in organization and intent to the 10CFR50 Appendix A criteria, with some variations to address generically the design variations of the DOE reactors. The current codes and standards are adequate for the design, construction, and licensing of the AFR-100 in the U.S.
5) Applicability of current analysis tools and date (new R&D needed).	Many of the currently used NRC licensing analysis tools (for LWR licensing) are not applicable to sodium-cooled fast reactors. Several non-system codes, such as siting-related analyses are applicable to the AFR-100, but the AFR-100 source term would be different from typical LWR source terms.

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	The Balance of Plant (BOP) with supercritical CO_2 Brayton cycle is still under development and has not been scaled to even a relatively small commercial scale for the AFR-100 power needs. Currently a test loop is operating at 0.78MWt scale. Analysis tools for the BOP will have to be verified and validated to reliably predict the performance of the BOP. Physics, fuel behavior, and thermal-hydraulics (e.g. SAS4A/SASSYS-1) analysis tools will have to go through V&V, although SAS4A/SASSYS-1 has been validated for some transients by modeling EBR-II and FFTF transient tests. Additionally, temperature criteria for assessments of cladding damage thresholds have been established by results from testing of metallic fuel in EBR-II and in the Transient Reactor Test (TREAT) Facility. Major R&D program will be needed to qualify these analysis tools.
6) Knowledge base and skills of NRC staff to address concept design and licensing.	The NRC has an excellent knowledge base and skills in water reactor technology. These skills are an excellent starting point for understanding the advanced concepts like the AFR-100 and the ability to license these concepts. During the 1990's, the NRC had personnel who reviewed two previous licensing applications for advanced sodium-cooled fast reactors, that is the PRISM and SAFR reactor plants. General Electric submitted a PSID for the PRISM reactor plant and Atomics International submitted a PSID for the SAFR reactor plant. NRC reviewed these applications and submitted two PSERs, NUREG-1368 (for PRISM) in 1994 and NUREG-1369 (for SAFR) in 1991. It is obvious that the NRC had a very competent

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	staff in those days who could readily handle the licensing evaluation of a sodium-cooled fast reactor. During the 2010 and 2011 time frame, the DOE conducted a series of training courses on sodium-cooled fast reactors overview, safety, core physics, coolant, fuels, and other fuel cycle related topics. In the audience were mostly personnel who had worked on prior fast reactor programs such as CRBR and other programs. So, the NRC does have personnel who have experience in SFR-related technology and its evaluation. In addition, interactions between NRC staff and foreign regulators in Japan, France, and Russia could have helped maintain the necessary knowledge base and skills of NRC staff (in sodium-cooled fast reactors). It is expected that there will be some need to upgrade skills of the staff to support the licensing of a fast reactor, but it should not be too challenging. In addition, the NRC uses other national resources (for example from the National Labs) to support licensing efforts and there are quite a number of personnel outside of the NRC staff who can and will support the licensing efforts of an advanced reactor.
7) Estimated validation and verification efforts (tests and computer codes).	The NRC has a suite of codes that it uses (or its vendors use) to support the licensing efforts for light water reactors. We have not quantified the effort required to validate and verify the NRC computer codes to support fast reactor licensing efforts.
8) Identification of any additional regulatory activities or products, such as previous NRC reviews or research efforts that could enhance the licensability of the concept.	Four previous licensing applications in the U.S. to the NRC of sodium-cooled fast reactors (PRISM NUREG-1368, SAFR NUREG-1369, Clinch River Breeder Reactor (CRBR) and the Fast Flux Test Facility

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	(FFTF)). In addition, there is the Experimental Breeder Reactor-II plant, the Transient Reactor Test Facility (TREAT), as well as foreign fast reactors such as PHÉNIX, SUPER PHÉNIX, Prototype Fast Reactor, BN-600, MONJU, BN-800, and Prototype Fast Breeder Reactor (PFBR) is relevant to the regulatory activities needed to support the licensability of the AFR-100.
9) The effect of unique fuel configurations on the licensing requirements for storage of spent nuclear fuel. In addition to the relevant regulatory requirements in 10CFR20, 10CFR50, and 52, the participant should address any unique issues of how the requirements of 10CFR72 would impact the long-term storage of spent nuclear fuel.	Spent fuel from the AFR-100 consists of individual fuel assemblies with fuel pins. The size of an individual assembly is comparable to that from a LWR. Following decay of afterheat and removal of sodium, the assemblies are coolable by air. Sufficiently unique features relative to spent LWR fuel assemblies have not been identified.
Category X – Nonproliferation	
1) Characterization of the fresh and spent/used fuel	For the AFR-100 U-Zr core, the initial core load is 23.9 metric tons of heavy metal with 3.22 MT of U-235 and 20.7 MT of U-238. The U-Zr core is 10% zirconium alloyed with the uranium. After 30 effective full power years of operation, the AFR-100 core will be allowed to decay in storage for a few months and then will be discharged to a storage tank filled with sodium until the spent nuclear fuel can be stored in an air environment. The isotopics of the spent nuclear fuel will be (for whole core): U-234 – 1.02E-3 MT U-235 – 1.39 MT U-236 – 3.09E-1 MT U-237 – 7.56E-5 MT U-238 – 18.4 MT Np-237 – 3.77E-2 MT Np-239 – 6.55E-4 MT Pu-238 – 6.79E-3 MT Pu-239 – 1.16 MT

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	Pu-240 – 8.31E-2 MT Pu-241 – 3.36E-3 MT Pu-242 – 1.97E-4 MT Am-241 – 1.03E-3 MT Am-242m – 2.43E-5 MT Cm-242 – 1.15E-5 MT Mixed Fission Products – 2.52 MT The spent nuclear fuel will have a very high radiation dose rate per assembly.
2) Other design characteristics that impact the materials control and accounting system (and whether significant development of a materials control and accounting methodology will be needed).	So, the AFR-100 has a long-lived core. The goal for material control and accounting will be that the fuel will reside in only a few locations: 1. As core assemblies within the reactor vessel that is filled with molten sodium. 2. As fresh core assemblies staged on site in the spent fuel pool that is filled with sodium. The fresh core assemblies will be staged to reduce the amount of time required for the (every 30 year) refueling. 3. As spent core assemblies that are decaying in storage (a sodium filled spent fuel pool) until such time as they are cool enough for transport. 4. In transition from the spent fuel pool (as a core assembly) to a washing station and to a fuel shipping cask for transport back to a processing or long-term storage facility.
3) Operational concept for the design as may impact proliferation risk	The operational concept for the design is very proliferation resistant. Special nuclear materials are either in the reactor vessel submerged under sodium, located in the storage vessel submerged under sodium, or in the process of transit to shipment off-site with appropriate additional guards and supervision to eliminate the divergence of any individual core assembly. Spent core

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	assemblies will be self-protecting due to the external radiation dose.
4) Relevant integral nonproliferation and security perspectives (e.g., material attractiveness of fuel considered in the context of anticipated security features/operational concept).	Due to the buildup of fission products in the metallic fuel or the deliberately designed incomplete separation of fission products from recycled metallic fuel, the fuel would be unattractive for handling in a theft scenario due to the high gamma activity.
Category XI – Research and Development	
1) A description of the key R&D needs that could be reasonably supported by a national laboratory	<u>Advanced Structural Materials</u> – the advanced materials used for the AFR-100 reactor vessel and heat exchangers are currently being pursued by the Advanced Reactors Concepts program in DOE. <u>AFR-100 Fuels</u> – The fuels development for the AFR-100 will need to be conducted to support licensing efforts and to support <u>Electromagnetic Pump Technology</u> – Self-cooled EM pump technology research and development was started during the ALMR program and has continued on-and-off for a number of years. ANL developed, with GE, a self-cooled EM pump that was tested at Argonne. In addition, GE and Toshiba tested a large-scale self-cooled EM pump that was tested for about 2,000 hours at ETEC. <u>Compact Sodium-to-Sodium Heat Exchanger Technology</u> – Very compact intermediate and DRACS heat exchangers are used to reduce the overall size of the primary reactor vessel. This technology needs to be matured to the degree that it can be used effectively in a sodium reactor environment. <u>Advanced Sodium Technology Development</u>

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	– For example, a compact fuel handling mechanism concept was developed for the AFR-100. This system will allow for the reduction in the overall primary plant size reducing the cost for the overall plant. This work in currently on-going at Argonne. Other advanced sodium technologies will be deployed in AFR-100 as the design matures. Ultra-high temperature nuclear instruments for detecting source, intermediate, and power range nuclear detection. Expected temperatures for the AFR-100 NI's are about 550C depending upon the ultimate location. Seismic Isolation Technology can be tested at a National Lab or University setting. S-CO_2 Brayton Cycle Power Conversion System – The S-CO_2 Brayton cycle needs to be demonstrated at larger scale at which the same turbomachinery technologies (e.g., oil-lubricated fluid film tilting pad bearings, dry gas liftoff seals, axial turbine) can be incorporated as would be used for the full-scale system. Ongoing R&D focused upon the reliable design of compact sodium-to-CO_2 heat exchangers needs to be completed to provide an understanding of sodium plugging phenomena in small sodium channels, the ability to drain and refill small sodium channels, as well as the stresses and potential damage from inadvertent freezing and remelting of sodium. Ongoing tests on interactions between sodium and CO_2 and sodium and sodium-CO_2 reaction products under conditions typical of postulated compact sodium-CO_2 heat exchanger failure need to be completed. Tests of the thermal shock response of compact sodium-to-CO_2 heat exchanger modules as well as compact

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	recuperator and cooler modules need to be carried out such that the thermal shock phenomena can be reliably predicted and incorporated into the design process. R&D is needed on approaches and technology to control the transport of tritium from the core to the S-CO_2 Brayton cycle. Nuclear reactions inside of fuel and boron carbide absorbers produce tritium that migrates through cladding into the primary sodium, diffuses through the IHX tube walls, enters the intermediate sodium, and can diffuse through the walls of the sodium-to-CO_2 heat exchangers. Tritium that collects in the CO_2 could be adsorbed on structural surfaces such as turbine and compressor blades where a significant uptake of tritium could pose a maintenance hazard for workers, or tritium that collects in the CO_2 could be released to the environment in the event of CO_2 release. One potential approach is the application of a barrier coating in the sodium-to-CO_2 heat exchangers to reduce the tritium diffusion rate into the CO_2. A second potential approach is preferential adsorption of tritium in a tritium trap to remove it from the intermediate sodium or the CO_2. Cold trapping of sodium traps tritium and hydrogen as hydride but does not completely remove them below a certain level. R&D is needed on approaches to control tritium with S-CO_2 cycles. It will be necessary to monitor to detect any leakages in sodium-to-CO_2 heat exchangers because sodium-CO_2 reactions can produce solid reaction products as well as gaseous carbon monoxide (CO). While sodium oxide that forms would be trapped in the intermediate sodium cold traps, carbon monoxide, carbonates, oxalates, and other

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	species would not be cold trapped and their behavior in flowing sodium as well as their removal from the sodium could be problematic. R&D is needed on the detection of CO_2 leakage into sodium such that the reactor can be shut down and the source of leakage isolated. One potential approach is acoustic detection of the sounds produced by CO_2 exiting microcracks as bubbles or jets. R&D is needed on potential approaches for trapping solid reaction products such that they do not pose a risk of plugging the sodium channels in a sodium-to-CO_2 heat exchanger. R&D is needed on accommodating gaseous CO, if CO bubbles don't react with sodium to form sodium oxide as they are transported through sodium over time. An alternative approach to leak detection is an alternative sodium-to-CO_2 heat exchanger design. Heatric Division of Meggitt (UK) can manufacture compact diffusion-bonded heat exchangers with small (e.g., 0.1 mm) sentinel channels between the channels of the two heat exchange fluids. The sentinel channels could be filled with a flowing gas such as helium which would be monitored for the presence of CO_2 and sodium indicating a leak. The sentinel channels would increase the heat exchanger size by 10 to 20 percent implying a somewhat greater cost but one which might turn out to be less than the cost associated with implementation of a different detection approach. Testing of model compact diffusion-bonded heat exchangers with sentinel channels needs to be carried out. Control strategy development for automated

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Information Requested	Concept Description
	control of the S-CO_2 Brayton cycle and the sodium-cooled fast reactor. Advanced Instrumentation – such as reliable sodium level probe detectors In-service Inspection and Repair technology Infrastructure to support the testing of the various advanced systems and components. Advanced modeling and simulation is expected to be useful for the overall analysis of the AFR-100 plant work. The ability to reliably simulate plant phenomena and operating conditions is an essential component of advanced reactor research and development. The purpose of modeling and simulation R&D is to raise the technical readiness of chosen concepts to a level to support commercial deployment. Much of the simulation work needed to support that effort can be effectively provided by concept-optimized system analysis tools. The system analysis codes run much faster than higher-fidelity simulation tools that rely on super-computers, and usually can be executed on desktop servers and work-stations. The systems analyses can capture integral transient effects for the entire plant, as opposed to high resolution analyses focusing only on specific components or local phenomena. Therefore, in a systems analysis, the major physics of the plants and integral effects are captured, albeit with some uncertainties if the geometry is only coarsely modeled or if the correlations have not been validated for the operating conditions. For design optimizations and sensitivity studies, system codes coupled with appropriate subgrid physics, correlations, or higher-fidelity tools can provide the information that is needed for an advanced concept in the

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	conceptual and preliminary design phases. Even though the physics may be common (neutron transport, incompressible fluid flow), the method implementations need to be optimized for specific concepts and operating modes (e.g. SAS4A/SASSYS-1, DIF3D, and REBUS). Advances in computing power and algorithms for solving complex systems of equations are enabling high fidelity simulations of advanced reactors which promise a greater understanding and predictability of phenomena and scenarios. Much development work is needed, however, before these codes can be used to their full potential and on a routine basis. Furthermore, validating high performance multiphysics, multiscale simulations remains a somewhat unresolved challenge. Development and testing of these tools is best conducted in concert with development, testing, and validation of the existing system and core analysis codes that will continue to be an essential part of R&D activities.
2) Identification of the general costs for the identified R&D.	Given the nature and breadth of the research and development proposed for the AFR-100 work, it is expected that the cost for the necessary research and development and the infrastructure to support the research and development would be in the ROM range of \$500M-\$750M before the advanced technology will reach TRL7 where an integrated demonstration can be performed in the prototype AFR-100.
3) Identification of the time frame in which the R&D is needed.	In accordance with DOE's Nuclear Energy Research Roadmap, the AFR-100 technology will need to be ready for commercialization (TRL8+) by the 2050 time frame. Thus the R&D is needed in the time from of 2014 to 2030.
4) Relative prioritization of potential R&D activities.	Because reactor plants are integrated machines, the relative priority of the R&D activities is about the same. However, some of these advanced technology choices can be

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	adjusted depending upon the deployment timeframe of the AFR-100. For example, straight tube heat exchanger technology could be used in lieu of twisted tube heat exchanger technology for the IHX and DRACS heat exchangers. The challenge would be that the AFR-100 reactor vessel would have to be physically larger in order to accommodate the conventional technology. Conventional structural materials could be used in lieu of the advanced structural materials. In this case, the amount of structural materials would increase by about 40%. Other examples can be given.

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