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Conceptual Design of a 500 MWe Traveling Wave Demonstration Reactor Plant

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Abstract – *The conceptual design of a 1200 MW thermal, 500 MW electrical, Traveling Wave Reactor (TWR) Plant has been completed with the objective of completing construction and startup by 2020. The reactor is a pool-type, sodium-cooled fast reactor and has been named TerraPower – 1 (TP-1). The TP-1 core will operate for over 40 years without refueling, but provisions have been made to insert highly instrumented test fuel and materials assemblies and to remove them for post irradiation examination and testing. These test assemblies will provide the bases for validating fuel design for future generations of TWRs. The fuel design is supported by an extensive materials and fuel development program.*

TP-1 uses proven technologies for most of the plant with a few notable exceptions. The fuel pins are designed to vent fission product gases to the primary sodium coolant in a controlled manner. Venting the fuel pins enables deep burnups required to sustain the core for over 40 years and greatly reduces the probability of cladding failures. Detailed studies have shown that the only significant impacts on the primary coolant and reactor operation are the fission product noble gases and Cs137. The noble gases decay, except for Kr85m, which is collected on carbon beds and stored in shielded vessels. Cs137 is removed from the primary coolant by reticulated carbon traps at a rate that prevents build up and keeps the primary coolant activity levels similar to fast reactors with unvented fuel. The intermediate heat exchanger is another area of innovation using printed circuit, plate-type modules that form a compact, robust safety barrier between the primary and intermediate coolants, while maintaining very high thermal-hydraulic efficiency. Thermal efficiency is also enhanced by the use of compact helical coil steam generators that yield a gross electrical conversion efficiency of 42%.

The seismically qualified active decay heat removal system is backed up by an independent, passive system using natural convection of ambient air to ensure decay heat removal even under station blackout conditions. Preliminary safety studies have confirmed satisfactory decay heat removal and acceptable reactivity coefficients. Initial identification of design basis and beyond design basis events has been completed. A level 1 Probabilistic Risk Assessment (PRA) is underway such that designers are informed of probability sequences that can benefit from design changes.

An extensive external review of the TP-1 Plant conceptual design has been completed and the initial Technical Baseline of the TP-1 Plant is being established. It is anticipated that preliminary and final design can be completed in about three years which would enable licensing efforts to progress to the point that a construction decision could be made. A successful 2020 startup and demonstration of TWR technology will give energy planners a sustainable nuclear power option that does not require reprocessing, reduces proliferation risk and opens the door for further longer term innovation.

I. INTRODUCTION

Traveling Wave Reactor (TWR) is the terminology used to describe a special class of fast reactors that have the ability to breed and burn *in-situ*, providing the possibility for very long core life and other characteristics not achieved in typical fast reactor designs. The first known proposal of a fast reactor design that could sustain a breed-and-burn condition using only natural uranium or depleted uranium as fuel was made in 1958 by Feinberg[1]. Similar concepts were proposed by Driscoll in 1979,[2] Feoktistov in 1988,[3] Teller in 1995,[4] and van Dam in 2000.[5] Fomin[6] has developed a mathematical treatment of the space-dependant criticality in nuclear burning waves and Sekimoto[7] has demonstrated the strengths of this type of reactor in his CANDLE Reactor configurations.

In 2006, TerraPower launched an effort to develop the first practical engineering embodiment of a breed-and-burn fast reactor, producing a design concept now known as a TWR.[8] Practical TWR designs were hampered initially because of conflicting demands made by the orientation of coolant flow and control systems in a reactor core whose critical burn region moved slowly as a function of time.

During 2008 TerraPower core designers achieved as major breakthrough when they demonstrated in 2-D simulations that a “standing wave” variant of the TWR could be achieved by periodically shuffling fuel within the core. The shuffling process will be described in more detail in Section IV. The standing wave concept was incorporated into a cylindrical core geometry that greatly simplified reactor design and enabled application of a large body of proven fast reactor engineering and design solutions.

Using the cylindrical standing wave concept, TerraPower launched a conceptual design effort in 2009 for an nth of a kind, 1150 MWe commercial TWR power plant. The objective of this conceptual design was to develop a complete plant design that would support a cost and schedule estimate to see if such a plant could be competitive with Generation III, III+ power plants. The results of the design and cost estimate[9] indicated that the “all in” costs were within the range reported for new US nuclear power plants.

The results of the 1150 MWe conceptual design were encouraging, so TerraPower started the conceptual design of a smaller, first of a kind, demonstration TWR designated the TerraPower – 1 (TP-1) nuclear power plant. The conceptual design of TP-1 is the subject of this paper.

II. THE TP-1 MISSION AND GENERAL REQUIREMENTS

The TP-1 Reactor Plant is a multi-mission facility that must provide the capability for the following mission components:

- Produce Electrical Power – levelized cost must be reasonably competitive with other types of nuclear power.
- Confirm TWR Core Performance – comparison of predicted and actual reactor core operation must confirm breed-burn rates.
- Qualify Fuel and Materials – standard and advanced fuel and materials must be irradiated and examined to qualify fuel for future generations of TWRs.
- Demonstrate High Availability – achieve average 90% availability over a 5 year period.
- Prototypic Experience – provide operating experience with pumps, heat exchangers and other large components that are prototypic for large commercial TWRs.

These mission components lead to a number of high level, general requirements that have been imposed on TP-1. Some of the more quantitative requirements are summarized in Table 1.

Table 1 Key General Requirements

General Requirement	Parameter
Plant Design Life	40 years
Core Life (eff. full power years)	40 years
Core Thermal Power	1200 MW
Plant Availability (avg. over 5 yr)	90%
Site Seismicity (IAEA SL-2)	0.15g
General Safety (IAEA Document)	NS-R-1
Safety Analysis (IAEA Document)	NS-G-2.1
Construction Time	48 months

III. Design Description

The TP-1 Reactor Plant conceptual design was completed in November 2010. A description of this design is provided in the sections that follow.

III.A. Reactor Fuel, Core and Control System

Reactor fuel for TP-1 is composed of uranium-zirconium metal alloy fuel slugs with a sodium thermal bond in martensitic stainless steel cladding. The fuel is designed for 35 atom-percent peak burnup and has a 61% smear density to allow for swelling at the design limit burnup. The dimensions of the fuel components are shown in Figure 1. Each fuel pin has a vent assembly at the top to vent gaseous fission products. The vent provides controlled release of the gases while preventing mixing of primary sodium coolant and bond sodium. Venting is used because conventional pressurized gas plenums develop

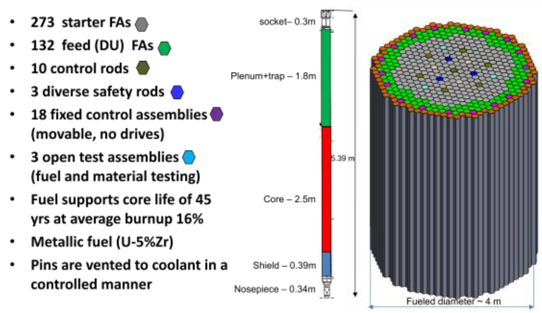

Figure 1 Fuel Pin and Core Description

unacceptably high pressures at high burnup. Venting also greatly reduces the likelihood of clad breach. The consequences of fuel venting are discussed in the Reactor Safety section of this paper. The fuel pin has a wire wrapped helically around the cladding to provide spacing between adjacent fuel pins and to promote coolant mixing.

A fuel assembly is composed of 169 fuel pins contained within a hexagonal duct that channels primary sodium coolant around the fuel pins. The duct contains pressure equalization holes above the top of the fuel and at special load pads near the top to prevent dilation of the duct during full flow conditions. The ducts are 15.99 cm flat-to-flat and have 6 mm spacing between adjacent fuel assembly ducts.

The reactor core is composed of two fuel assembly types. As shown in Figure 1, fuel assemblies in the active control zone have uranium alloy fuel slugs with beginning-of-life (BOL) enrichment less than 20 % ^{235}U. Fuel assemblies in the fixed control zone have uranium alloy fuel pins that contain only depleted uranium at BOL. The two types of fuel assemblies are initially distributed to achieve criticality and to start the breed-burn process. Excess reactivity in the core increases monotonically and control rods are slowly inserted to offset this increase. At a pre-determined point the reactor is shut down and higher burnup assemblies are moved to the periphery of the core and depleted uranium fuel assemblies replace the high burnup fuel. This process, referred to as "shuffling", occurs about every 18 to 24 months. Shuffling accomplishes three very important functions:

- Reduces Excess Reactivity – maintains operation within the control system capability
- Limits Fuel Assembly Burnup – moving high burnup fuel to the core periphery effectively stops further burnup
- Extends Core Life – periodic shuffling continues until all depleted uranium fuel assemblies have been used, thus several decades of core life is obtained

The control function is accomplished with control rods which are ducted clusters of clad B_4C absorber pins. In the active control zone, these control rods are inserted or withdrawn as needed for core operation. Similar absorber assemblies are placed in the fixed control zone, but they cannot be moved during operation. Their purpose is to control reactivity and power in the peripheral region of the core as high burnup fuel is moved into this region. The assemblies can be relocated by in-vessel handling machines during a shuffling shutdown to optimize core performance.

A dedicated group of three safety rods with diverse system design are used to rapidly shut down the reactor during transient conditions. Their operation will be described in more detail in the "Reactor Safety" section of this paper.

III.B. Reactor Mechanical Systems

There are a number of mechanical systems that directly or indirectly support the reactor core. The core and all primary cooling system components are contained in the reactor vessel with the addition of a number of in-vessel structures. Figure 2 illustrates the reactor vessel, major in-vessel structures and the reactor head. All vessels and structures in direct contact with primary sodium coolant are fabricated

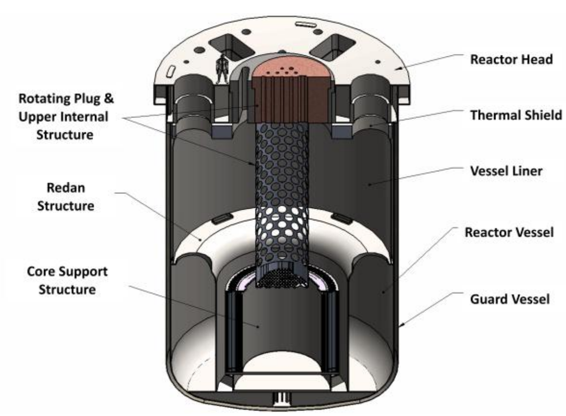

Figure 2 Reactor Vessel and Head Cut Away

from 316H stainless steel, where H indicates a control range of 0.2 to 0.8% carbon. Figure 2 also shows the guard vessel fabricated from 516 carbon steel.

Both vessels and a number of major components are supported from the reactor head. The reactor head is supported by a bearing system interfaced to a concrete ledge in the reactor building. The reactor head is a composite structure fabricated from 304 stainless steel metal plates filled with concrete or metallic shielding material. The reactor head shielding is such that operators can perform maintenance and inspections in containment during reactor operation. The reactor head has two rotatable plugs at the center as shown in Figure 2 that enable access to all removable core components with the in-vessel handling machines.

Reactor containment consists of the guard vessel, the reactor head, the upper containment vessel and a continuity plate that connects the upper containment vessel to the reactor head. Reactor containment also has equipment and personnel air locks and penetrations, some of which require isolation valves. Figure 3 illustrates upper containment with additional mechanical systems required to be located in the upper containment. Also shown are large removable ports at the top of the upper containment vessel that allow access for removal of major in-vessel components such as primary pumps and intermediate heat exchangers.

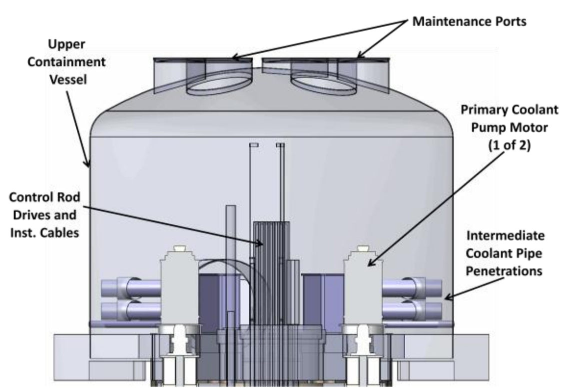

Figure 3 Upper Containment Vessel

Removable core components are accessed by two machines called the straight-pull in-vessel handling machine (IVHM-SP) and the offset-arm in-vessel handling machine (IVHM-OA). The IVHM-SP is mounted on the inner rotating plug and is capable of accessing radial rows 1 through 6 in the core. The IVHM-OA is mounted on the outer plug and can access assemblies in rows 6 through 12. The overlap of the two machines in row 6 enables a "hand off" of assemblies in rows 1 through 5 to the IVHM-OA if they must be moved to outer core positions or storage positions outside the core.

Core components can enter or exit the core from a single dedicated port in the reactor head outer rotating plug, accessible by the IVHM-OA. External access to this port requires that additional equipment be brought into containment by opening one of the upper containment access ports. The handling equipment interfaces to the reactor head via adapters and valves to ensure that there is no air ingress into the reactor vessel. Additional equipment forms a temporary, inerted, confinement boundary to allow insertion or withdrawal of heated sodium bearing assemblies. Aside from the initial core loading operations, these ex-vessel handling operations are infrequent because they are required only for the fuel qualification mission.

There is a large crane hall directly above the upper reactor containment. This crane hall extends over the adjacent Auxiliary Building and the Reactor Maintenance Building to allow components in casks to be transported to specialized cells for temporary storage, sodium removal, disassembly and storage or preparation for shipping to other facilities for further testing. Conditioning for new core components also occurs in this area prior to transport to the Reactor Building and insertion into the reactor vessel.

III.C. Thermal-Hydraulic Systems

The TP-1 reactor uses a number of thermal-hydraulic systems to extract heat energy from the reactor core and transport the heat in a form suitable for conversion into electrical power. The heat transport starts in the reactor core by pumping primary sodium coolant into a plenum at the bottom of the core. The coolant is heated to a nominal 510 °C as it exits the core and flows through four intermediate heat exchanger (IHX) modules. Coolant exits the IHX at a nominal 360 °C and enters the plenum to complete the primary coolant cycle.

The IHX heats intermediate sodium coolant as primary coolant passes through the other side of the heat exchanger and the intermediate coolant flows through pipes to either of two once-through, helical coil, counter flow steam generators. Cooled intermediate sodium flows to mechanical pumps that return the coolant to the IHX to complete the cycle. Intermediate sodium is not exposed to neutron irradiation and, thus, all the intermediate piping, pumps and steam generators remain non-radioactive. Furthermore, the pressure in the IHX is always higher on the intermediate coolant side than on the primary side in all operating and shut down modes. Therefore, if the IHX leaked, only non-radioactive intermediate sodium would leak into the primary system.

The steam generators use the hot intermediate sodium to raise super-heated steam in the tube bundle. This steam is transported to the turbine generator where it spins a multi-stage turbine and generates electrical power. The Turbine Island and its supporting systems will be described in the Balance of Plant section. Heated feed water is returned to the steam generators tube bundles to complete the cycle. Steam generator tube leaks are an obvious concern for this system, so leak detection is provided at very high sensitivity and rapid response time. Detection of small leaks results in operater action that may include a reactor shutdown, followed by valve closure to exclude additional feedwater. If the leak is large enough, it may cause a rupture disk to open which has pipes coupled to a blow down system for capture of sodium-water reaction products.

III.D. Balance of Plant

The TP-1 Balance of Plant (BOP) is composed of a number of buildings, structures and systems that provide non-safety grade functions for a complete, autonomous nuclear power plant. Figure 4 shows a rendering of the plant.

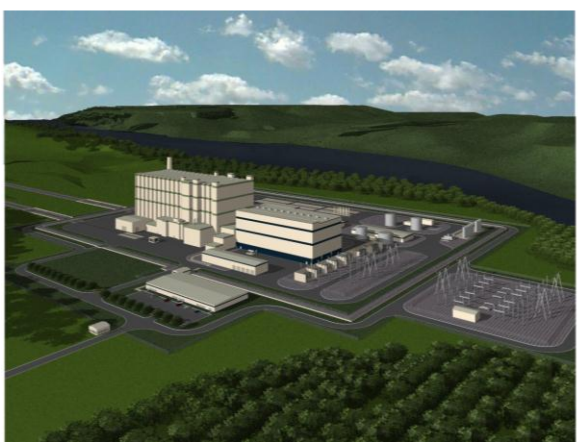

Figure 4 TP-1 Plant Rendering

Some of the more important functions of the BOP are:

- Steam Rankine Cycle for converting heat energy to electrical power,
- Transformers and switchyard for transmission of power to the grid,
- Emergency Diesel Generators to provide investment protection for the plant,
- Physical security for the plant,
- Fire detection and suppression systems,
- Radwaste facilities for processing and storing radioactive waste,
- Warehouses and facilities for plant maintenance.

The TP-1 Plant conceptual design assumes once-through fresh water cooling for its heat sink. After site selection it may be necessary to modify this part of the BOP design to accommodate site specific conditions.

IV. REACTOR SAFETY

Reactor safety for the TP-1 Plant is based on strategies that have proved successful for previously operated fast reactors combined with state-of-the-art analytic methods to ensure the highest levels of public protection. TP-1 is similar to other pool-type, sodium-cooled fast reactors in that Loss of Coolant Accidents (LOCAs) are eliminated from the design bases. This means that the primary design basis accidents (DBA) are Loss of Flow (LOF) and Loss of Heat sink (LOHS). Transient overpower (TOP) events are much less frequent in TP-1 because interlocks prevent excessive rod withdrawal during operations.

TP-1 decay heat removal after shutdown is normally accomplished by circulating both primary and intermediate sodium coolants with pump pony motors that are connected to the emergency diesel generators. If pony motors are unavailable, natural circulation is sufficient to remove decay heat. The heat is dissipated through the steam generators using a large dedicated inventory of feedwater. In the rare event that this system is disabled, decay heat is removed by a passive decay heat removal system called the Reactor Vessel Air Cooling System (RVACS). RVACS draws ambient air down four coaxial chimneys and circulates the air through a baffle structure surrounding the guard vessel, exhausting the heated air through the chimneys. During normal operation RVACS maintains the concrete in the Reactor Building at safe temperatures. If the normal decay heat removal system fails, the primary sodium coolant temperature rises causing it to expand and flow over a liner inside the reactor vessel. The heated primary sodium flows down the inside surface of the reactor vessel radiantly heating the adjacent guard vessel. The RVACS has sufficient capacity to remove the peak decay heat requirement that occurs about 24 hours after shutdown. RVACS has no moving parts and requires no operator actions.

One difference from previous fast reactor operation is the venting of fission gases from individual fuel pins described in the Design Description section. The primary consequences of venting have been identified as a small amount of ^{137}Cs and ^{134}Cs in the primary sodium coolant and ^{85m}Kr in the argon blanket gas. The cesium isotopes are continuously removed from the sodium with reticulated vitreous carbon traps and stabilized when the traps become saturated. The krypton gas is initially collected on cryogenic carbon beds and pumped to high-pressure storage vessels. These storage systems become distributed radioactive source terms in the reactor safety analyses. No significant engineering problems have been identified for fuel venting or the systems required to maintain the primary coolant and blanket gas at levels similar to unvented fuel conditions.

IV.A. Probabilistic Risk Assessment

A detailed Level 1 probabilistic risk analysis (PRA) is in progress based on the conceptual design. The preliminary results confirm that seismic events dominate the core damage frequency as has been seen in other sodium-cooled fast reactor PRAs.[10] The main sequence leading to core damage and a late large release is a seismic event that disables all decay heat removal capability. This leads to bulk primary sodium boiling approximately one day after the initiator.

Unprotected transients are rare events that can be challenging for fast reactors because of the positive sodium void worth. Some measures have been taken in TP-1 to make these events even rarer, although they are already expected to occur at a frequency less than 10^{-7} per reactor year. In addition to the reduction in the TOP frequency using interlocks to prevent excessive control rod removal during operation, the primary pumps are designed to not trip until there is a confirmation of rod insertion. By confirming the insertion of the rods before tripping the pumps, the primary ULOF initiator becomes a much rarer failure of a single pump followed by a failure to scram. Additionally, the loss of a single pump with failure to scram will still have one pump running. The loss of both pumps due to a common cause failure is expected to be another order of magnitude lower. Finally, analysis of a previous sodium-cooled fast reactor PRA similar to TP-1 has shown that nominal inherent negative reactivity feedback is of relatively low risk importance. Without inherent feedback, the frequency of an energetic event increases by a factor of four.[11] The factor of four is an acceptable risk increase because the base core damage frequency is less than 10^{-9} per reactor year.

IV.B. Seismic Analysis

The importance of seismic events has motivated us to perform seismic analysis of key nuclear island systems at the conceptual design phase. The purpose of the analysis is to predict the seismic response of the TP-1 reactor head and reactor vessel including the reactor guard vessel. The results of this preliminary analysis are being used to identify potential issues in these components or structures coupled to them. The reactor is analyzed for the IAEA SL-2 design basis earthquake. The zero period ground acceleration as defined in ASCE 4-98[12] is 0.15g in all three ordinate directions. Scaling the standard spectrum shape from NRC Regulatory Guide 1.60[13] by the 0.15g factor, a contractor determined the in-structure response spectra at a location corresponding to where the lip of the reactor head interfaces with the building foundation.

The analysis methodology is in accordance with the ASCE 4-98 standard. Specifically, the frequency based method is used because the components of interest are considered to be linear and elastic. A quasi-static model has been developed in accordance with ASCE 4-98 to compare predications with the frequency based predications. Having results from both methods provides information that is useful in planning the methodology for future analyses. No time-history analyses were required for this study, but this technique may be used in the future for non-linear systems such as interactions of fuel assemblies within the core, sloshing fluids, or the concrete and steel reactor head. The finite element program ANSYS is used to model the reactor head, reactor vessel, and guard vessel. The reactor head is modeled as a composite of solid stainless steel elements and concrete volumes. The reactor vessel and guard vessel are modeled as stainless steel shell elements.

The intermediate heat exchangers, primary sodium pumps, large and small shield plugs, and upper internal structure are attached to the reactor head. They are modeled as point masses with moments of inertia about their centroids. Because these designs are at the conceptual level, their geometric properties are approximated using their preliminary CAD models. All connections and joints are idealized in the model. That is, welds and bolted connections are assumed to have full strength. None of the models consider operating condition loads:

fluid flow, rotating pump shafts, etc. These analyses will be continually refined during the preliminary design phase.

IV.C. Preparations for Licensing

At the start of conceptual design it was not clear where TP-1 might be sited and what regulatory agency would be responsible for licensing. Therefore, TerraPower adopted the IAEA safety series requirements[14] as the most general framework for licensing TP-1 subject to adjustments for specific site regulatory requirements. These IAEA safety requirements are at such a high level that implementation is not as straightforward as it is with the large body of prescriptive regulations developed by the US Nuclear Regulatory Commission.

With the assistance of licensing consultants, a roadmap for implementing the IAEA safety requirements was developed for TP-1. In the process several safety issues important to TP-1 were identified and these issues will be proposed for pre-application discussions with the regulatory agency responsible for licensing TP-1.

Much work remains in the area of licensing, but initiating the process coupled with ongoing reactor safety analyses is expected to keep pace with the aggressive design and construction goals for TP-1.

V. CONCLUSION

Completion of the TP-1 conceptual design has provided a self-consistent, complete plant design that can serve as the starting point for preliminary and final design. No obstacles have been identified that would preclude start of construction by 2015 and initial criticality by 2020.

Preliminary TP-1 safety studies indicate the reactor will be as safe or safer than the current generation of light water reactors subject to confirmation with qualified calculational methods. Level 1 PRA is providing guidance to engineers to achieve a risk-informed design. Use of consensus codes and standards for structures, systems and components important to safety under IAEA general licensing requirements has established a basic design suitable for global licensing.

TerraPower and its partners look forward to the construction and startup of TP-1 as the world's first operational traveling wave reactor.

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