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Project No.: 713

November 27, 2001

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Documents Supporting the November 30, 2001 Pre-application Meeting Regarding the Pebble Bed Modular Reactor (PBMR)

Dear Sir/Madam,

Attached are 2 documents that will support the scheduled November 30, 2001 PBMR pre-application meeting between the U.S. Nuclear Regulatory Commission and Exelon Generation (Exelon). Attachment 1 contains the technical information that complements a planned PBMR Pty. presentation regarding operational modes and states. Attachment 2 contains the regulatory position that complements a planned Exelon Generation presentation regarding testing requirements for a combined license.

The information is being submitted in advance of the meeting in order to aid the NRC in their preparation for the meeting and to provide documentation of the presenters' discussions.

If you have any questions, please contact me at (610) 765-5528.

Sincerely,

Kevin F. Borton
Manager, Licensing

Attachments

cc: Thomas King, RES
William Borchardt, Associate Director NRR
John Flack, RES
James Lyons, NRR
Stuart Rubin, RES
Amy Cubbage, NRR

D064

Attachment 1

"PBMR Operational Modes and States"

To be presented by
PBMR Pty.
On November 30, 2001

32 Pages

Presentation to the US Nuclear Regulatory Commission (USNRC) on 30 November 2001

PBMR Operational Modes and States

Presented by Willem Kriel
Manager Module Dynamics and Control Group
PBMR (Pty) Ltd, PO Box 9396, Centurion 0046, South Africa

The purpose of this paper is to document the dialogue accompanying the slide presentation prepared for the US Nuclear Regulatory Commission (USNRC). The presentation is scheduled for 30 November 2001 and will be presented in Washington DC.

The topic is: **"PBMR Operational Modes and States"**

The presentation consists of 91 slides. The dialogue and slide presentation together constitute the entire presentation. Generally it is best if the audience can review the slide presentation and dialogue prior to the presentation. The presentation is designed as an opportunity to, not only present the basic concepts but also to provide clarification to any questions raised during the presentation.

The dialogue is as follows:

1. SLIDE 1

Unlike other topics, plant operations can generally not be presented meaningfully if the subject material is too simplified. A meaningful presentation covers a lot of in-depth concepts. Audience concentration is required throughout the presentation for an optimal understanding of the subject matter. This presentation assumes that the audience has a basic understanding of reactor operations and the basic PBMR design, and that the audience can follow the presentation without the narration of all of the slide material. Proficient understanding of the subject matter may require self-study, and informal follow-up questions may be addressed at a later date as the project matures.

2. SLIDE 2

The objective of the presentation is to inform and educate the US Nuclear Regulatory Commission regarding the operation of the PBMR and to provide information regarding the application of analytical codes used by PBMR to evaluate plant operation. The NRC is requested to provide general comments and feedback regarding the basic modes and states of PBMR operation, and the presented use of analytical codes.

3. SLIDE 3

Please refer to slide.

4. SLIDE 4

Slide 4 indicates the major PBMR plant components and gas systems.

5. SLIDE 5

The efficiency of the ideal Brayton cycle can be improved by using a portion of the heat rejected during the cooling process to preheat the gas before it enters the heat source. Another method of improving the efficiency is to use multistage compression with inter-cooling. The PBMR utilizes both these mechanisms, and the modified cycle on which the PBMR is based is referred to as the recuperative Brayton cycle. Another distinctive characteristic of this recuperative Brayton cycle is the use of helium as the working fluid. A schematic sectional layout of the plant to perform this cycle is shown in slide 5 together with the temperature-entropy diagram of the cycle.

Helium at a relatively low pressure and temperature is compressed by the Low Pressure Compressor (LPC) to an intermediate pressure. The helium is then cooled in an intercooler. As mentioned above the inter-cooling between the two multistage compressors improve the overall cycle efficiency. A High Pressure Compressor (HPC) then compresses the helium to the maximum pressure in the circuit. The helium is then preheated in the recuperator before entering the reactor that heats the helium to the maximum circuit temperature. After the reactor, the hot high-pressure helium is expanded in a High Pressure Turbine (HPT) after which it is further expanded in a Low Pressure Turbine (LPT). The High Pressure Turbine (HPT) drives the High Pressure Compressor (HPC) while the Low Pressure Turbine (LPT) drives the Low Pressure Compressor (LPC). After the Low Pressure Turbine (LPT), the helium is further expanded in the

Power Turbine (PT). The helium is then cooled in the recuperator, after which it is further cooled in the pre-cooler. This completes the cycle.

The heat removed from the coolant before the helium flows into the pre-cooler equals the heat transferred to the helium returning to the reactor inlet. The recuperator uses heat from the cooling process that would otherwise be lost to the ultimate heat sink to heat the gas before it enters the nuclear heat source, thereby reducing the heating demand on the reactor, thus increasing the overall plant efficiency.

6. SLIDE 6

The following dialogue also refers to slide 7.

Controllability and transient performance of the PBMR are ultimately determined by the dynamic characteristic of the Reactor Unit and the Power Conversion Unit. The following dominant system characteristics should be recognized to understand the dynamic response of the PBMR:

- The thermal response of the Reactor is slow. This is because of the large thermal capacity of the graphite-moderated core relative to its heat generation and removal rates.
- The output and speed response of the turbo machinery is very fast compared to the reactor dynamics. This is because of their relatively low inertia compared to the available operating torque.

The difference in speed of response of the Reactor Unit and Power Conversion Unit has a certain advantage but at the same time also creates a demanding co-ordination requirement.

The advantage is that the large thermal capacity of the core allows relatively fast load changes of the system without requiring fast response from the core. In principle, the energy stored in the core can be tapped or additional energy can be stored, with minimal core temperature changes. The reactor also has a negative temperature coefficient, which results in the reactivity and consequently the neutronic power to counteract temperature changes. The reactor is therefore to a large extent self-regulating maintaining a constant reactor outlet temperature for a given control rod or reactivity setting.

As a result of low turbo-machine inertia, sudden loss or partial loss of external electric load will result in immediate acceleration of the turbo-machines to potentially unacceptable rotational speeds. This is most noticeable in the case of the power turbine generator that requires active speed control when the generator is not connected to a stable grid. The high and low-pressure turbo units have the added stability of a compressor and turbine on a single shaft. The compressor improves the stability of a

turbo unit by counteracting power requirements with speed variation. Nonetheless, the turbo unit speeds vary across a wide range due to changes in the Main Power System (MPS) operational conditions.

Thus, in principle the PBMR consist of an inherently stable and slow acting heat source coupled to a very fast acting power conversion unit. The Power Conversion Unit (PCU) requires active control.

The most important PBMR abbreviations and definitions used in this presentation are listed in slides 16 and 17.

The following 4 definitions are clarified in more detail:

Cooling Water Temperature (CWT) is defined as the water inlet temperature of the pre-cooler and the intercooler. The Cooling Water Temperature (CWT) is a function of the site location and the type of ultimate heat sink used. Coolant options for obtaining the required cooling may include wet cooling towers, dry cooling towers or water from dams, rivers, lakes or the sea. The different cooling methods have different approach temperatures and will result in a wide range of water inlet temperatures for the pre-cooler and the intercooler. The Cooling Water Temperature (CWT) also varies during the year as the site temperatures change. The plant is designed to operate at ultimate heat sink temperatures between –2 °C and 38 °C as a result of the wide range of coolant options. A study of the variation of ultimate heat sink temperatures across the world have shown that the ultimate heat sink temperature varies between –2 °C to 38 °C for prospective PBMR sites. The –2 °C lower limit is associated with high salt content cold sea conditions. The Cooling Water Temperature (CWT) is assumed to be 3 °C higher than the ultimate heat sink temperature. This is as a result of the efficiency of the intermediate heat exchangers. The intermediate heat exchangers exchange heat between the demineralised cooling water circuit feeding into the pre- and the intercooler and the seawater. This implies a Cooling Water Temperature (CWT) range of 1 °C to 41 °C. The plant design temperature is 28 °C.

Reactor Outlet Temperature (ROT) is the average helium outlet temperature of the reactor. Under normal power operation the average helium outlet temperature of the reactor is 900 °C, which is also the maximum allowable reactor outlet temperature and results in the highest overall cycle efficiency for any given Cooling Water Temperature (CWT) in the range of 1 °C to 41 °C.

Maximum Continuous Rating (MCR) also referred to as 100% of Maximum Continuous Rating (MCR) is defined as the maximum grid power produced by a PBMR unit at a Cooling Water Temperature (CWT) of 28 °C and a Reactor Outlet Temperature (ROT) of 900 °C. The reactor power at 100 % of Maximum Continuous Rating (MCR) is limited by the maximum reactor power of 268 MW, and the manifold pressure of 7.0 MPa. 7.0 MPa is also the maximum working pressure of the manifold and other Primary Pressure Boundary (PPB) components. A PBMR unit is capable of more than 100 % of Maximum Continuous Rating (MCR) if the Cooling Water Temperature (CWT) is below

28 °C. A unit functioning below 28 °C Cooling Water Temperature (CWT) can produce grid powers in excess of Maximum Continuous Rating (MCR) without exceeding maximum reactor power and pressure limits. This is as a result of cycle efficiency improvements associated with the reduction in Cooling Water Temperature (CWT). If the Cooling Water Temperature (CWT) rises above 28 °C then the PBMR unit will be limited to below Maximum Continuous Rating (MCR) grid power levels. This is as a result of the 7.0 MPa pressure limit of the Primary Pressure Boundary (PPB) and an efficiency reduction in power conversion which is associated with an increase in Cooling Water Temperature (CWT).

Maximum Continuous Rating Inventory (MCRI) is defined as the mass of helium in the Primary Pressure Boundary (PPB) at 100 % Maximum Continuous Rating (MCR) with a Cooling Water Temperature (CWT) of 28 °C and a Reactor Outlet Temperature (ROT) of 900 °C. Thus, 40 % Maximum Continuous Rating Inventory (MCRI) is the mass of helium in the Primary Pressure Boundary (PPB) when 40 % of Maximum Continuous Rating (MCR) is produced and 100 % Maximum Continuous Rating Inventory (MCRI) is the mass of helium in the Primary Pressure Boundary (PPB) when 100 % Maximum Continuous Rating (MCR) is produced (Assuming that the CWT is 28 °C, the ROT is 900 °C and all bypass valves are closed for both 40 % of Maximum Continuous Rating (MCR) and 100 % Maximum Continuous Rating Inventory (MCRI)). It must be noted that the helium mass corresponding to 50 % Maximum Continuous Rating Inventory (MCRI) is **NOT** half the mass of helium corresponding to 100 % Maximum Continuous Rating Inventory (MCRI). Even though the relationship is very close to a linear relationship, it is not a linear relationship.

The **Reactor Outlet Temperature (ROT)** is controlled. The Reactor Outlet Temperature (ROT) is usually controlled to a specific value and during certain transitions and modes of operation the outlet temperature can be ramped at a given rate of temperature increase or decrease. The Reactor Outlet Temperature (ROT) can be controlled when the reactor is critical or when the reactor is sub-critical. The control differs for these two cases.

When the reactor is critical the Reactor Outlet Temperature (ROT) is controlled using the control rods. The Reactor Outlet Temperature (ROT) is measured together with the reactor neutronic power and the reactor fluidic power. The neutronic power is derived from the neutron flux measurement. The reactor fluidic power is calculated using the reactor inlet and outlet temperature and the helium mass flow rate through the reactor.

When the reactor is sub-critical the Reactor Outlet Temperature (ROT) is controlled using the SBS speed. Only the Reactor Outlet Temperature (ROT) is measured. By changing the SBS speed the mass flow rate through the reactor can be adjusted, which in turn determines the heat removed from the reactor, thereby controlling the Reactor Outlet Temperature (ROT).

The shutdown rods are not used to regulate the temperature directly. They are used during a reactor scram and when the plant is taken to maintenance. The Reserve Shutdown System (RSS) consisting of the Small Absorber Spheres (SAS) are only used

during the maintenance modes, and transitions to and from maintenance modes. When the plant is in maintenance mode the Reactor Outlet Temperature (ROT) must be kept below 400 °C. During maintenance the maintenance valves are used to isolate the reactor from the Power Conversion Unit (PCU). The Core Conditioning System (CCS) is used to cool the reactor during maintenance.

The Inventory Control System (ICS) is used to adjust the inventory in the Primary Pressure Boundary (PPB) for **Inventory Power Control**. Helium is normally injected into the Primary Pressure Boundary (PPB) at the inlet of the pre-cooler and removed at the manifold. Helium can also be injected into the manifold from the Inventory Control System (ICS) booster tank, but the amount of helium is dependent on the pressure in the booster tank. The helium inventory is adjusted so that the helium mass flow rate needed to drive the turbo machinery can produce the required generator output. Pressure ratios and temperatures remain constant and only the gas density and power level are changed. High efficiency can be maintained at all power levels above 40% of Maximum Continuous Rating (MCR) with helium inventory adjustment.

Electrical power generated is measured at the generator terminals. This determines whether helium should be injected or removed from the Primary Pressure Boundary (PPB). The pressure in the manifold is measured to ensure that the Primary Pressure Boundary (PPB) is not over pressurised. The total inventory in the Primary Pressure Boundary (PPB) is estimated to ensure that the minimum and maximum MCRI limits are not exceeded for a given Cooling Water Temperature (CWT).

By using the compressor bypass valves the electrical power generated can also be controlled. The valves used are the Low Pressure Compressor Bypass Valve (LPB), Low-Pressure Compressor Bypass Control Valve (LPBC), High-Pressure Compressor Bypass Valve (HPB) and the High-Pressure Compressor Bypass Control Valve (HPBC). Slide 7 shows single valves, but in practice the bypass configurations consist of multiple valves. PBMR also references the valve sets in singular and not plural. By opening the bypass valves some of the helium that would normally pass through the reactor and turbines is re-circulated through the compressors. The power turbine power is reduced due to the decrease in mass flow rate and the compressors proportionately use more of the available energy. As opposed to inventory control, opening the bypass valves results in changes in the cycle temperatures and the rotational speeds of the turbo-units.

Under normal load ramping between 40 % Maximum Continuous Rating (MCR) and 100 % Maximum Continuous Rating (MCR) the Low-Pressure Compressor Bypass Control Valve (LPBC), and the High-Pressure Compressor Bypass Control Valve (HPBC) valves can be used in addition to the Inventory Control System (ICS). This enables the plant to ramp up and down at 10 % of Maximum Continuous Rating (MCR) per minute without a non-minimum phase effect resulting from injecting helium at the pre-cooler inlet (The non-minimum phase effect results in the power decreasing before increasing due to helium injection upstream of the compressors). Below 40 % Maximum Continuous Rating (MCR) the power is controlled using the Low Pressure Bypass Valve (LPB) and High-Pressure Compressor Bypass Valve (HPB). It is also possible when the Inventory

Control System (ICS) is completely isolated (or for plants that do not have an Inventory Control System (ICS)) to control the power between 0 % Maximum Continuous Rating (MCR) and 100 % Maximum Continuous Rating (MCR) using only the Low Pressure Compressor Bypass Valve (LPB) and High-Pressure Compressor Bypass Valve (HPB).

Speed Control of the power turbine generator is performed when the generator is not synchronised to the grid. The Power Turbine Generator (PTG) is then free to change speed and active control is required to maintain the speed at a specified set point. Speed control is also used during the synchronisation process when the generator is synchronised to the grid.

The speed controller measures the speed of the Power Turbine Generator (PTG) and uses the bypass valves (Low Pressure Compressor Bypass Valve (LPB) and High-Pressure Compressor Bypass Valve (HPB)) and the variable resistor bank (CRB) to control the speed. The Low Pressure Compressor Bypass Valve (LPB) and High-Pressure Compressor Bypass Valve (HPB) change the Power Turbine (PT) fluidic power, which acts as driving force to the Power Turbine Generator (PTG). The Continuous Resistor Bank (CRB) manipulates the absorbed electrical power produced by the generator and therefore produces a loading or braking mechanism to the Power Turbine Generator (PTG). It is also possible that the PBMR house load can contribute to the generator load. This is dependant on the backup power configuration as well as the status of the various breakers.

The Loss of electrical load transient is associated with **Rapid load reduction**. When a rapid reduction of electrical power occurs the Gas Cycle Bypass Valves (GBP) are used. The Gas Cycle Bypass Valves (GBP) consists of 8 valves that open simultaneously in a very short time (0.3 seconds). This results in rapid reduction of power turbine fluidic power.

In the case where the connection to the grid is lost the Gas Cycle Bypass Valves (GBP) are used to prevent over-speed of the Power Turbine Generator (PTG). The Gas Cycle Bypass Valves (GBP) are only opened for a short period of time (about 2 seconds) after which they are closed. This ensures that the Brayton cycle is kept self-sustaining. The Low Pressure Compressor Bypass Valve (LPB) and High-Pressure Compressor Bypass Valve (HPB) and the Continuous Resistor Bank (CRB) are then used to regulate the power turbine generator frequency. In the case where a Power Conversion Unit (PCU) trip occurs the Gas Cycle Bypass Valves (GBP) are left open and the Brayton cycle stops functioning.

In the case of a magnetic bearing failure on the power turbine generator it is required that the speed of the power turbine generator be rapidly reduced. For this reason the peak braking system or Peak Resistor Bank (PRB) is used to provide a braking mechanism for the Power Turbine Generator (PTG). The peak braking system or Peak Resistor Bank (PRB) is a large resistor bank and is used only for a relatively short period of time.

The **Recuperator Inlet Temperature Control** function protects the recuperator from elevated temperatures that could reduce recuperator life. The recuperator is designed for a maximum working temperature of 600 °C. During operation at low power levels (< 40 % MCR) the Low Pressure Compressor Bypass Valve (LPB) and High-Pressure Compressor Bypass Valve (HPB) are used. This results in temperature changes in the system. The low-pressure inlet temperature of the recuperator can increase above the 600 °C limit if cold helium is not mixed into the main gas stream.

The recuperator temperature controller measures the recuperator low-pressure side inlet temperature. The temperature is controlled using the High-pressure Coolant Valve (HCV) and / or the Low Pressure Coolant Valve (LCV). The coolant valves allow cool helium from the manifold to be introduced into the hot low-pressure gas stream upstream of the recuperator.

The **Start-up Blower System (SBS) control** function is required for the following:

- Removal of heat from the reactor in respect of fissile and decay heat
- Conditioning of the Main Power System (MPS)
- Start-up of the Main Power System (MPS)

The Start-up Blower System (SBS) is used to induce and maintain helium mass flow through the Main Power System (MPS) for the above-mentioned functions. A speed request is sent to the Start-up Blower System (SBS), the speed at which the Start-up Blower System (SBS) blowers rotate determines the mass flow through the Main Power System (MPS). Higher-level controllers determine the mass flow rate required which in turn implies blower speed.

Surge control of the Start-up Blower System (SBS) is done using the Start-up Blower System Bypass Valve (SBP) and Start-up Blower System Bypass Control Valve (SBPC). The inlet and outlet pressure and the inlet temperature of the Start-up Blower System (SBS) are measured. The Start-up Blower System Inline Valve (SIV) and the Start-up Blower System Isolation Valve (SBSV) are used to bypass the Start-up Blower System (SBS) or to channel the main helium flow through the Start-up Blower System (SBS) respectively.

The **Reactor Inlet Temperature Control** function is required during the Main Power System Start-up and Synchronisation transition and when the Reactor Outlet Temperature (ROT) is reduced. The reactor inlet temperature is decreased in a controlled manner to increase the thermal energy removed from the reactor. The Start-up Blower System (SBS) must be operational and adjusting the Start-up Blower System (SBS) blower speed controls the mass flow rate through the reactor. The heat removed from the reactor is adjusted by changing the temperature difference over the reactor or the mass flow rate through the reactor.

The inlet temperature to the reactor is controlled using the Recuperator Bypass Valve (RBP). The Recuperator Bypass Valve (RBP) causes part of the helium to bypass the

high-pressure side of the recuperator resulting in an effectiveness reduction regarding the recuperator performance. At very low mass flow rates the recuperator effectiveness is almost 100%. With an almost 100% recuperator effectiveness almost all the heat removed from the reactor is returned to the reactor by the recuperator. By reducing the effectiveness of the recuperator with bypass flow it is possible for heat removed from the reactor to reach the Pre-cooler. The recuperator bypass flow decreases the reactor inlet temperature and increases the Start-up Blower System (SBS) and Pre-cooler inlet temperatures. The inlet temperature of the Start-up Blower System (SBS) and the reactor outlet temperature are also measured to prevent unacceptably high Start-up Blower System (SBS) inlet temperatures or unacceptably high temperature differentials across the reactor. If the operating limits of these temperatures are encroached then the bypass flow is adjusted.

7. SLIDE 7

Please also refer to the slide 6 dialogue. Slide 7 is a schematic representation of the control elements relevant to the main control functions as listed in slide 6.

8. SLIDE 8

An accurate simulator is essential to determine the transient response of the PBMR plant. By making use of Flownet, which is based on first principles, PBMR is capable of transient simulations of the reactor unit coupled to the power conversion unit. The plant thermo-hydraulics and mechanical interfaces are modelled in Flownet.

All of the plant controllers including advanced controllers are embedded in Simulink. State Flow is used to configure mode logic, control sequencing and interlocks. Flownet, Simulink and State Flow are used for interactive analysis of all the plant dynamics. Accurate sensor and actuator dynamical models are also incorporated in the codes. Coupling all three of these codes together is key to the design of the system as well as providing loading catalogue data for the various components in the system. These coupled codes form the basis of the PBMR plant simulator, which will form the basis of the training simulator.

9. SLIDE 9

The first version of Flownet was released in 1991. The code is used both locally and overseas by leading industrial companies and institutions such as Rolls-Royce in the UK, Cranfield University, the British Defence Research Agency, Eskom, Sastech, CSIR and Iscor. The code has been extensively validated for plant conditions in general industry that are applicable to the PBMR.

The Quality Assurance program for the development of Flownet is in accordance with ISO9001, ANSI/ANS-10.4 and NQA-1. The PBMR verification and validation program for Flownet requires comparisons with analytical data, experimental data as well as benchmarking with other codes.

PBMR, M-Tech and the University of Potchefstroom have embarked on the experimental testing of a three shaft Brayton Cycle model that is representative of the PBMR. This experimental test is one of the experimental tests that have been identified for the validation of the code. The model is not a scale model but it does contain the same Main Power System (MPS) components as the PBMR except for the reactor, which is replaced with a representative electrical heat source. The turbomachines for the model have been sourced and the construction of the model will commence early 2002. The purpose of the model is to provide experimental validation for Flownet and to demonstrate the application of the code. The operation of a three shaft directly coupled recuperative Brayton cycle will also be demonstrated.

The approach followed in Flownet is to model the PBMR Main Power System (MPS) as a network of interconnected components such as pipes, diffusers, heat exchangers, compressors, turbines, orifices, valves and the reactor. Components are connected to nodes at their upstream and downstream ends. Any number of components can be connected at one node. The code is also capable of evaluating gas mixtures. This makes the code very flexible.

Components such as compressors and turbines are modelled as grouped systems. Their performance data (i.e. pressure ratio and efficiency) are obtained through interpolation from standard-format corrected mass flow rate performance maps. The code is capable of extrapolating on the corrected mass flow rate performance maps. All normal operation analyses do not require extrapolating on the corrected mass flow rate performance maps.

Components such as pipes and heat exchangers are modelled as distributed systems by dividing them into a number of smaller elements. The primary pressure losses through pipes, diffusers and heat exchanger flow passages are calculated with the Darcy-Weisbach equation (or an adapted version of the equation in the case of compressible flow), while the secondary losses are modelled through a loss factor.
Pressure losses through valves and orifices are modelled in a fundamental way, although it is possible to interpolate values from pressure drop/flow rate curves supplied as input data. The pressure loss relationships for pipes and orifices are valid up to a Mach numbers of unity. The code, however, correctly predicts the flow rate at supercritical pressure ratios when the flow is choked. This is important when simulating abnormal situations, e.g. a blow-down through a rupture in a pipe.

The pressure drop through the reactor core is calculated with the Ergun equation, which applies to the flow through a packed bed. The reactor is internally divided into two flow paths, one representing the flow through the outer annular fuel sphere region, and the other representing the flow through the inner cylindrical reflector region. Both flow paths are divided into a number of horizontal layers to account for the change in flow properties through the reactor.

The heat transfer in the heat exchangers is calculated using a three-point temperature model in the cross-flow direction. The three temperatures are the average hot stream temperature, the average temperature of the metal separating the hot and cold streams, and the average cold stream temperature.

The reactivity in the reactor is calculated using the point kinetics model. The reactivity is used as an input to the temperature distribution calculation for a representative fuel sphere in each horizontal layer. Dividing the spheres into a number of radial layers, and then solving the temperature in each layer, using a finite difference technique, forms the basis of the reactor model.

The overall network solution algorithm is based on an implicit finite difference technique. With this technique, the nodal pressures and temperatures are iteratively adjusted at each time interval to simultaneously satisfy the continuity and energy equations for each node, as well as the pressure drop/flow rate relationship for each element (component). It is important to note that the continuity equation is in the conservative form, thereby ensuring overall conservation of mass in closed systems.

Flownet can deal with both steady state and transient flows. It also simulates slow and fast transients. In the case of transient flows, the inertia of the fluid as well as the thermal inertia of other components such as heat exchanger walls and fuel spheres in the reactor core are taken into consideration.

A more detailed description of the solution algorithm can be found in the Flownet User Manual.

10. SLIDE 10

This slide summarizes the most important features and attributes of Flownet. Flownet is PBMR Intellectual Property.

Flownet is based on first principles and PBMR uses Flownet as an analysis and simulation tool. Flownet uses implicit solving techniques and is capable of time step variation. The code is also an industrial code and is widely used in general industrial applications.

The code has multi fluid capability and can also accommodate compressible and incompressible fluids. The code also allows for gravitational effects.

The code is written as object orientated coding and makes use of relational databases.

PBMR's vision for Flownet is that it will be the "RELAP" of PBMR type reactors.

11. SLIDE 11

Slide 11 is a schematic representation of the nodes and elements constituting the model of the PBMR Main Power System (MPS) in Flownet.

12. SLIDE 12

Slide 12 is a schematic representation of the PBMR plant process being modelled in Flownet and the PBMR controllers in Simulink controlling the plant. The bottom right graph indicates a grid power set point ramp as a black trace. The resulting plant response is indicated as a blue trace. The graph clearly indicates the plant's ability to follow the power ramp at a rate of 10 % of Maximum Continuous Rating (MCR) per minute from 40 % Maximum Continuous Rating (MCR) to 100 % Maximum Continuous Rating (MCR).

13. SLIDE 13

The graph indicates required grid power set point versus time as a green trace. PBMR considers this a challenging example of Primary Frequency Support (PFS). The PBMR plant response is indicated as a black trace. This graph clearly indicates the plant's ability to provide Primary Frequency Support (PFS).

14. SLIDE 14

The graph indicates required grid power set point versus time as a green trace. PBMR considers this a challenging example of Automatic Generation Control (AGC). The PBMR plant response is indicated as a black trace. This graph clearly indicates the plant's ability to provide Automatic Generation Control (AGC).

15. SLIDE 15

The primary modes and states of operation are shown in slide 15. The entire presentation is summarised in this single slide. Slides 16 to 91 expand or focus on various detail areas of this slide. Keeping this slide accessible for the rest of the presentation and referring to it will help tremendously as far as context and relevance of the following slides are concerned. This slide is the most important PBMR operations navigation tool. All the PBMR modes and states are encompassed in the diagrammatic representation on slide 15. Later on in the presentation this slide is again shown as slide 20. Slides 15 and 20 are identical.

16. SLIDE 16

The most important abbreviations relevant to the presentation are summarised in slides 16 and 17.

17. SLIDE 17

The most important abbreviations relevant to the presentation are summarised in slides 16 and 17.

18. SLIDE 18

Please refer to slide.

19. SLIDE 19

The purpose of a detailed description of the symbology is to clarify intent and to aid understanding and navigation through the concepts depicted in the slides.

Slide 19 is a "key" for the symbology used in the logic diagram slides of this presentation. More specifically the symbology refers to slides 15 and 20.

20. SLIDE 20

Now that the symbology, definitions and abbreviations are clear we can effectively navigate through the main modes, states, transitions and transients of the PBMR. Slide 20 and 15 are identical. The slides that follow focus on the detail modes, sub modes, states, transitions and transients as depicted in slides 15 and 20. Please continually refer to slides 15 and 20 regarding the rest of the presentation to maintain context, relational association and relevance.

21. SLIDE 21

This slide indicates the main modes with the embedded sub modes. The transitions and transients have been removed for clarity.

22. SLIDE 22

Please refer to slide.

23. SLIDE 23

Please refer to slide.

24. SLIDE 24

Please refer to slide.

25. SLIDE 25

Please refer to slide.

26. SLIDE 26

Please refer to slide.

27. SLIDE 27

Please refer to slide.

28. SLIDE 28

Please refer to slide.

29. SLIDE 29

The reactor is sub-critical in this state.

Three reactivity control devices (acting independently or in combination) keep the reactor sub-critical, namely:

- The Small Absorber Spheres (SAS) Reserve Shutdown System (RSS)
- The shutdown rods
- The control rods

In this mode (and all sub-modes) the Start-up Blower System (SBS) is operational. By controlling the speed of the SBS (and therefore the mass flow rate through the core) the reactor temperature is controlled. The heat generated in the reactor in this mode is due to decay heat only.

In this slide the Brayton cycle is described as being not self-sustaining. This means that the thermodynamic cycle requires external assistance to operate in the form of externally induced mass flow of the working fluid. Even though the Brayton cycle requires assistance to circulate working fluid the Brayton cycle can still contribute considerably to

the fluid circulation. If the Brayton cycle is said to be self-sustaining then the thermodynamic cycle operates and stays in operation without external support such as that which can be provided by the Start-up Blower System (SBS).

30. SLIDE 30

Full shutdown (2a) occurs before or after maintenance of the Main Power System (MPS). The Reserve Shutdown System (RSS), the shutdown rods and the control rods are fully inserted and the reactor outlet temperature is less than 550 °C.

Intermediate shutdown (2b) occurs after a Reactor SCRAM transient or after an Insert Shutdown Rods Transition. The shutdown rods and the control rods are inserted. The outlet temperature of the reactor is not permitted to fall below 550 °C (to be verified). At this temperature and with both sets of rods inserted the reactor will be sub-critical even at zero xenon levels.

Partial Shutdown (2c) occurs after a Control rod SCRAM transient or after the Insert Control Rods Transition. The control rods are inserted. The outlet temperature of the reactor is not permitted to fall below 750 °C (to be verified). At this temperature and with the control rods inserted the reactor will be sub-critical even at zero xenon levels.

31. SLIDE 31

Please refer to slide.

32. SLIDE 32

Please refer to slide.

33. SLIDE 33

Please refer to slide.

34. SLIDE 34

The Power Conversion Unit (PCU) Start-up and Synchronisation Transition goes directly from MPS Ready (3b) to Power Operation (5), bypassing PCU operational (4), as synchronisation to the grid takes place before the Brayton cycle is fully self-sustaining.

35. SLIDE 35

Please refer to slide.

36. SLIDE 36

In this mode or state a remote grid operator can automatically adjust the plant grid power to a power set point. This functionality of being capable of submitting the plant to automatic power set point adjustments is fundamentally different from existing reactor designs.

37. SLIDE 37

The Reactor Outlet Temperature (ROT) may be adjusted at a given rate.

If the plant was operated at high power levels before a power reduction, then xenon concentration increases. The increased Xenon concentration level may imply that sustained critical operation of the reactor unit and plant at 900 ■■C is not feasible. This will occur when the amount of reactivity that the control rods can compensate for at 900 ■■C, is less than the negative transient reactivity associated with the high levels of xenon. That is, the control rods are nearly completely withdrawn. Under these conditions, the reactor will enter the floating reactor temperature controller mode. This controller will allow the Reactor Outlet Temperature (ROT) to float (in the allowable range 750 to 900 °C), thus keeping the reactor critical. This reactor sub-mode will be triggered when the rods are fully withdrawn. Double shifting of the PBMR plant typically necessitates this mode of operation.

38. SLIDE 38

The reactor outlet temperature will be controlled to 900°C and the power delivered to grid will be between 40% and 100% of Maximum Continuous Rating (MCR). In this mode and state of operation, the plant is capable of providing power to the grid, as well as supporting ancillary services.

The ancillary services supported in this mode and state of operation are:

- Load following. The power supplied to the grid can automatically be adjusted to match the signal dispatched by the system operator within a scheduling period.
- Automatic Generation Control (AGC) also known as Regulation or Secondary frequency support. The power supplied to the grid is automatically adjusted in accordance with control signals issued by the system operator. The grid power of the plant can be adjusted between 40 and 100% of Maximum Continuous Rating (MCR) by using the Inventory Control System (ICS) to manipulate Main Power System (MPS) helium inventory levels at a rate of up to 10% of Maximum Continuous Rating Inventory (MCRI) per minute. The bypass valves are simultaneously used to compensate for the non-minimum phase response caused by the injection. Remote control by the grid system controller remains subject to limits determined by the PBMR's own control system, such as ramp rates and maximum power.
- Primary frequency support, also known as governing. The power supplied to the grid is adjusted by the plant in response to the rotational frequency of the Power Turbine Generator (PTG). This is characterised by the "droop" function when operating outside a specified frequency dead-band.

This mode or state will also be subjected to Runback conditions. Runbacks are pre-programmed transitions within a specific mode and state to ensure sustainability of operation given a sub-system capacity reduction. This results in a degraded mode of operation where all the control capabilities for the plant still exist and are operational, but subject to additional limitations. A typical example would be the operation of plant when only half the ultimate heat sink system pumps are in operation. All the control software, controller and mode and state logic is active and operational, but the allowable power that can be delivered to the grid is limited. It is limited physically by the heat removal capacity of the heat exchangers as well as being limited in the controller software by changing the maximum allowable power request.

39. SLIDE 39

Modes 1 to 5 were presented but mode 4 was omitted. Slide 39 shows the transitions and transients associated with mode 4, which is known as PCU Operational. The reason for leaving mode 4 till last is that it is an unusual mode that is not frequently entered. This mode or state will generally only be entered through a Loss of load transient. The plant operator can enter the mode through operator intent through the Controlled grid

separation transition. It is unlikely that the plant operator will wilfully separate the generator from the grid and opt to keep the Brayton cycle self-sustaining. In this mode the house load and resistor bank load the generator. Generally if the operator intends separating from the grid the Brayton cycle need not be self-sustaining and the operator should use the PCU disengage transition.

40. SLIDE 40

In this mode of operation the reactor power levels are typically between 25 and 100 MW thermal. The reason for this is that the house load and resistor bank are the only loads to the generator since no power is exported to the grid unless the unit is regionally islanded through grid fragmentation. The bypass valves are utilized to control the speed and fluidic power associated with the power turbine. The use of bypass valves reduces the efficiency of the plant.

41. SLIDE 41

Please refer to slide.

42. SLIDE 42

Slide 42 is a summary of the modes and sub modes presented in the preceding slides. The most important plant and subsystem states are presented in the table. The table is referred to as the Main Power System State Matrix.

43. SLIDE 43

Slide 43 details the time fractions associated with each mode.

44. SLIDE 44

The 6 transitions associated with a self-sustaining Brayton cycle are the simplest transitions to explain and will be dealt with first.

Please refer back to slides 15 or 20, which depict the Main Mode diagram.

In the **Controlled grid separation transition** the Power Turbine Generator (PTG) power is reduced and breakers required to separate the Power Turbine Generator (PTG) from the grid are opened. The operator will choose whether the house-load will be supplied from the generator or an external source. The relevant breaker is set to trip when the power through it is less than a set threshold. The operator adjusts the base-load controller set point to the power level required. Once the Main Power System (MPS) is disconnected from the electrical network, the Power Turbine Generator (PTG) speed controller becomes active and regulates the generator frequency to 50Hz.

The **Synchronisation transition** is the transition from the PCU operational (4) mode to Reduced-capability Operation (5a). In this transition, the generator frequency, voltage and phase of the voltage is synchronised with the external grid following which the power to the grid is increased by removing the resistor load and closing the bypass valves.

The **Close-down** transition is from the PCU operational (4) mode to MPS ready (3b). There are two different Close-down transitions that may be followed. The Close-down transitions are dependant on whether house load power is supplied by an external source or by the Power Turbine Generator (PTG) before initiation of the transition. The differences will not be discussed in this presentation. For both Close-down transitions the Brayton cycle is still self-sustaining at the start of the transition and by using the speed controller the speed of the Power Turbine Generator (PTG) can be reduced in a controlled manner. However, once the Power Turbine Generator (PTG) has attained the required speed, the Low Pressure Compressor Bypass Valve (LPB) and High-Pressure Compressor Bypass Valve (HPB) can be fully opened. The Brayton cycle will cease to be self-sustaining. As soon as it is safe to operate the Start-up Blower System (SBS), the blowers will be started and the "conditioning" controller will be initialised.

The **Increase capability** and **Reduce capability** transitions are associated with the average reactor outlet temperature and grid power level. If the average reactor outlet temperature is 900 °C and the grid power is equal to or above 40% of Maximum Continuous Rating (MCR) then the plant is in Normal power operation (5b). If the aforementioned is not true and the plant is synchronized and the Brayton cycle is self-sustaining then the PBMR is in Reduced-capability operation (5a).

The **PCU disengage** transition is from Reduced-capability Operation (5a) to MPS ready (3b). The Brayton cycle is still self-sustaining at the start of the transition and by using the speed controller the speed of the Power Turbine Generator (PTG) can be reduced in a controlled manner. However, once the Power Turbine Generator (PTG) has attained the required speed, the Low Pressure Compressor Bypass Valve (LPB) and High-Pressure Compressor Bypass Valve (HPB) can be fully opened. The Brayton cycle will cease to be

self-sustaining. As soon as it is safe to operate the Start-up Blower System (SBS), the blowers will be started and the "conditioning" controller will be initialised.

45. SLIDE 45

Slide 45 depicts the De-Fuel transition. A detail discussion of this transition is outside the scope op this presentation. The De-Fuel transition is effectively a presentation in itself.

It is suffice to say that during this transition the pressure in the Primary Pressure Boundary (PPB) will be 1 Mpa and the temperature in the reactor must be less than 250°C. Generally the reactor is De-Fuel by exchanging fuel elements with graphite elements.

46. SLIDE 46

Slide 46 depicts the Re-Fuel transition. A detail discussion of this transition is outside the scope op this presentation. The Re-Fuel transition is effectively a presentation in itself.

It is suffice to say that during this transition the pressure in the Primary Pressure Boundary (PPB) will be 1 Mpa and the reactor could be fuelled whilst being critical at very low power levels. Generally the reactor is Re-Fueled by exchanging graphite elements with fuel elements.

47. SLIDE 47

This is a transition from Closed maintenance (1b) to Open maintenance (1a).

48. SLIDE 48

Please refer to slide.

49. SLIDE 49

This is a transition from Open maintenance (1a) to Closed maintenance (1b).

50. SLIDE 50

Please refer to slide.

51. SLIDE 51

The transition is from Fuelled maintenance (1) to Shutdown (2).

52. SLIDE 52

Please refer to slide.

53. SLIDE 53

This transition is from Shutdown (2) to Fuelled maintenance (1).

54. SLIDE 54

Please refer to slide.

55. SLIDE 55

This transition is from Shutdown (2) to Reactor ready (3a). The reactor is made critical during the transition. The Small Absorber Spheres (SAS) and the shutdown rods will be removed during the transition depending on whether the transition is Reactor Startup (a), (b) or (c). The control rods and reactor temperature will be used for reactivity adjustment.

The Start-up Blower System (SBS) must be operational to ensure that accurate temperature measurements and control are possible. In this transition the Reactor Outlet Temperature (ROT) controller is switched from the average Reactor Outlet Temperature (ROT) decay heat controller to the Reactor Outlet Temperature (ROT) reactivity controller that manipulates the control rods.

Before initiating the transition, the operator must confirm that the protection bypass that allows SBS/CCS operations at low Reactor Outlet Temperature (ROT) temperatures is activated.

Nuclear heating must not exceed 20 MWt before all the SAS channels have been emptied.

56. SLIDE 56

Please refer to slide.

57. SLIDE 57

Please refer to slide.

58. SLIDE 58

Please refer to slide.

59. SLIDE 59

The operator will initiate the transitions.

Before the Reactor Outlet Temperature (ROT) of 550 ..C is reached the operator must ensure that the Reserve Shutdown System (RSS) is fully inserted and shall then bypass the interlocks that disallow the operation of both the Start-up Blower System (SBS) and Core Conditioning System (CCS) at low Reactor Outlet Temperature (ROT). This is relevant to the Insert RSS transition.

60. SLIDE 60

This transition is from Reactor ready (3a) to MPS ready (3b). The main goal of this transition is to get the MPS to specified temperature levels. This is necessary to reduce temperature differentials when the Brayton cycle is started.

61. SLIDE 61

The Start-up Blower System (SBS) is used to control reactor fluidic heat by adjusting the mass flow rate through the reactor. The Reactor Outlet Temperature (ROT) reactivity controller will control the Reactor Outlet Temperature (ROT). The heat removed from the reactor is used to heat the components. The hot gas coming from the reactor is mixed with cold gas from the manifold using the High-pressure Coolant Valve (HCV). In this manner the recuperator low-pressure inlet temperature is controlled. The plant is conditioned for start-up when the Main Power System (MPS) is within the start-up limits. If the plant parameters do not remain within start-up limits the plant will revert back to Reactor Ready (3a) mode or state through the .NOT. Conditioned transition.

62. SLIDE 62

This transition is from MPS ready (3b) to Reduced-capability operation (5a). The Brayton cycle becomes self-sustaining in this transition. This is achieved by using the Start-up Blower System (SBS) and the gas cycle valves. Speed control of the Power Turbine Generator (PTG) is achieved using the resistor bank and by-pass valves.

63. SLIDE 63

Please refer to slide.

64. SLIDE 64

Please refer to slide.

65. SLIDE 65

Slides 65 through 71 detail PCU start-up and synchronisation transition parameters. These slides are included in the presentation to show how important parameters respond as the plant moves through this transition. These slides also indicate typical results that are obtained using Flownet and Simulink to analyse the transition.

66. SLIDE 66

Please refer to slide.

67. SLIDE 67

Please refer to slide.

68. SLIDE 68

Please refer to slide.

69. SLIDE 69

Please refer to slide.

70. SLIDE 70

Please refer to slide.

71. SLIDE 71

Please refer to slide.

72. SLIDE 72

The four PBMR transients that will be dealt with in the last section of this presentation are:

- Loss of load transient
- PCU trip transient
- Control rod SCRAM / Reverse transient
- Reactor SCRAM transient

73. SLIDE 73

The plant enters this transient when a loss of load is detected in Power operation (5). A Loss of load transient is caused by loss of load on the generator. From a safety viewpoint, only the GBP action is required to prevent over-speed of the generator.

When a loss of load condition is detected the Gas Cycle Bypass Valves (GBP) will open within 0.3 seconds with no feedback control. Opening the valves will ensure that the Power Turbine Generator (PTG) does not over-speed. The Gas Cycle Bypass Valves (GBP) will close as soon as the rotational acceleration is negative. This will ensure that the Brayton cycle remains self-sustaining. Simultaneously the Low Pressure Compressor Bypass Valve (LPB), High-Pressure Compressor Bypass Valve (HPB) and High-pressure Coolant Valve (HCV) are opened and the resistor bank load is set to maximum. The low- and high-pressure bypass valves and resistor bank are used to control the Power Turbine Generator (PTG) rotational frequency to 50 Hz. The High-pressure Coolant Valve (HCV) will ensure that the recuperator low pressure inlet temperature does not exceed 600 °C. The electrical protection may operate to trip the generator circuit breaker or the High Voltage (HV) breaker, depending on the nature of the initiating event. The Reactor Outlet Temperature (ROT) reactivity controller will control the Reactor Outlet Temperature (ROT).

The High Voltage (HV) breaker will be opened when the following condition is true:

Generator breaker closed .AND. speed > 52.5 Hz.

This is to prevent sustained operation on a grid fragment that comprises of more than one generator running at too high a frequency.

74. SLIDE 74

Please refer to slide.

75. SLIDE 75

Please refer to slide.

76. SLIDE 76

Slides 76 through 83 depict the Loss of load transient. These slides are included in the presentation to show how important parameters respond as the plant moves through this transition. These slides also indicate typical results that are obtained using Flownet, Simulink and State Flow to analyse the transition.

The graphs depicted in slides 77 through 83 are for a 302 MW thermal PBMR plant exporting 120 MW electrical to the grid. All other references in the presentation are for a 268 MW thermal PBMR plant exporting 106 MW electrical to the grid.

Slide 76 indicates the State Flow control logic. Slide 76 also serves as a key for the graphs depicted in slides 77 through 82.

At **point A** the loss of load condition is detected and the Gas Cycle Bypass Valves (GBP) are opened within 0.3 seconds.

At **point B** the Gas Cycle Bypass Valves (GBP) are closed because the rotational acceleration of the Power Turbine Generator (PTG) has become negative. Simultaneously the Low Pressure Compressor Bypass Valve (LPB), High-Pressure Compressor Bypass Valve (HPB) and High-pressure Coolant Valve (HCV) are opened and the resistor bank load is set to maximum. The low- and high-pressure bypass valves and resistor bank are used to control the Power Turbine Generator (PTG) rotational frequency back to 50 Hz. The High-pressure Coolant Valve (HCV) is controlled so as to maintain the recuperator low-pressure inlet temperature below 600 °C.

The electrical protection may operate to trip the generator circuit breaker or the High Voltage (HV) breaker, depending on the nature of the initiating event.

The Reactor Outlet Temperature (ROT) reactivity controller controls the Reactor Outlet Temperature (ROT).

At point C the PBMR plant parameters are stable, the Power Turbine Generator (PTG) frequency is 50 Hz and the speed controller controls the speed of the Power Turbine Generator (PTG). The PBMR plant is now in PCU Operational (4).

77. SLIDE 77

The graph indicates grid power or generator external load versus time as a black trace. The fluidic power driving the Power Turbine (PT) is indicated as a blue trace. Before point A the PBMR plant is operating steadily at 50 Hz. The fluidic power driving the Power Turbine (PT) is higher than the grid power or generator external load because of the generator losses. The Power Turbine Generator (PTG) rotational speed is stable at 50 Hz.

The external load is lost just prior to detection at point A. Between points A and B, the fluidic power driving the Power Turbine (PT) is substantially more than the resistor bank load on the generator and subsequently the Power Turbine Generator (PTG) rotational speed increases. Between points B and C the opposite is true and the Power Turbine Generator (PTG) rotational speed is brought back to 50 Hz.

The Power Turbine Generator (PTG) rotational speed versus time is indicated in slide 78.

78. SLIDE 78

The Power Turbine Generator (PTG) rotational speed versus time is indicated in slide 78.

79. SLIDE 79

The maximum and minimum PBMR Main Power System (MPS) helium pressures versus time are indicated in slide 79.

80. SLIDE 80

The reactor mass flow rate versus time is indicated in slide 80.

81. SLIDE 81

The graph indicates reactor core neutronic power versus time as a black trace. This graph is a powerful and clear demonstration of the inherently stable and slow acting properties of the PBMR reactor.

The fluidic power or heat removed by the helium coolant is indicated as a blue trace. The heat removed by the coolant is closely coupled to the mass flow through the reactor as would be expected. The response of the heat removed by the helium coolant curve closely resembles the reactor mass flow rate versus time trace indicated in slide 80.

Slide 81 highlights the slow acting, stable properties of the reactor or heat source (black trace) versus the very fast acting power conversion unit (blue trace).

82. SLIDE 82

The Reactor Outlet Temperature (ROT) versus time is indicated in slide 82. The Reactor Outlet Temperature (ROT) drops to 865 °C before recovering to approximately 900 °C. The temperature reduction is a result of the helium system pressure reduction indicated as a black trace in slide 79.

83. SLIDE 83

Slide 83 indicates the map movement of the Low Pressure Compressor (LPC) operating point on the Low Pressure Compressor (LPC) map. The movement on the map is indicated as a black trace. During the transient the Low Pressure Compressor (LPC) pressure ratio and corrected mass flow rate are reduced.

If the time allocated for the presentation is sufficient a demonstration of a coupled Flownet, Simulink and State Flow analysis will be demonstrated. All three codes will be coupled and will be run on the same laptop to illustrate the analysis of a Loss of load transient. This demonstration highlights the interactive nature of the coupled codes and

the interfaces developed to make such powerful couplings. The demonstration also highlights the transient analysis capabilities and speed of Flownet with respect to complex network models.

84. SLIDE 84

This transient occurs when there is a fault within the Power Conversion Unit (PCU). The plant will default to this transient when a Power Conversion Unit (PCU) trip signal is generated. Excessive vibration or temperature increases due to the total loss of Active Cooling System (ACS) cooling are typical trip signals, which will activate this transient. This transient will default to MPS ready (3b) mode and state after the Brayton cycle has ceased to be self-sustaining.

The generator breaker should trip on reverse power detection. The Gas Cycle Bypass Valves (GBP) will be opened to dissipate the power turbine fluidic power. The Gas Cycle Bypass Valves (GBP) will remain open until the Brayton cycle has shut down completely. The resistor bank will be utilised to load the Power Turbine Generator (PTG) until it has decelerated to an acceptable level. During the transient the Start-up Blower System (SBS) will be activated to maintain flow through the reactor.

85. SLIDE 85

Please refer to slide.

86. SLIDE 86

The transient is initiated with the Control Rod SCRAM / Reverse Transient Signal (CRS). Excessive reactor outlet temperatures and neutron flux levels and rate of neutron flux increase are typically the parameters that could initiate a reverse or a SCRAM. This transient will end in Partial shutdown (2c) after the Brayton cycle has shut down. The reactor will be sub-critical.

The Start-up Blower System / Core Conditioning System Inhibit for low Reactor Outlet Temperatures (ROTs) is automatically activated when the SCRAM transient is initiated or in the case of the reverse transient. The inhibit is activated when two control rods have reached their fully inserted position. A Control rod SCRAM / Reverse transient is accomplished by gravitationally fully inserting the control rods into the reactor in the

case of the SCRAM condition and fully inserting them in a controlled manner using the motor drives in the case of the reverse condition. A controlled shutdown of the Power Conversion Unit (PCU) will take place with the operation of the Low Pressure Compressor Bypass Valve (LPB) and High-Pressure Compressor Bypass Valve (HPB). This causes the speeds of the turbo units and the power being delivered by the Power Turbine Generator (PTG) to be reduced. The generator breaker will trip on reverse power conditions. The resistor bank will be utilised to load the Power Turbine Generator (PTG) until it has decelerated to an acceptable level.

The SBS will be started and the Reactor Outlet Temperature (ROT) control will be passed from the Reactor Outlet Temperature (ROT) reactivity controller to the Reactor Outlet Temperature (ROT) decay heat controller.

87. SLIDE 87

Please refer to slide.

88. SLIDE 88

The transient is initiated with the Reactor SCRAM Transient Signal (RS). This SCRAM will only occur if the previously initiated control rod SCRAM has not had the required result. This transient will end in the Intermediate shutdown (2b) mode or state after the Brayton cycle has shut down. The reactor will be sub-critical.

The Start-up Blower System / Core Conditioning System Inhibit for low Reactor Outlet Temperatures (ROTs) is automatically activated when the transient is initiated. The shutdown and control rods are gravitationally fully inserted into the reactor. The Gas Cycle Bypass Valves (GBP) will be opened to dissipate the power turbine fluidic power. The Gas Cycle Bypass Valves (GBP) will remain open until the Brayton cycle has shut down completely. The operation of the Gas Cycle Bypass Valves (GBP) causes the speeds of the turbo units and the power being delivered by the Power Turbine Generator (PTG) to be reduced. The generator breaker will trip on reverse power conditions. The resistor bank will be utilised to load the Power Turbine Generator (PTG) until it has decelerated to an acceptable level. The Start-up Blower System (SBS) will be started and the Reactor Outlet Temperature (ROT) control will be passed from the Reactor Outlet Temperature (ROT) reactivity controller to the Reactor Outlet Temperature (ROT) decay heat controller.

89. SLIDE 89

Please refer to slide.

90. SLIDE 90

Please refer to slide.

91. SLIDE 91

Please refer to slide.

End of presentation.

Attachment 2

"Testing Requirements for Issuance of a Combined License"

To be presented by
Exelon Generation
On November 30, 2001

13 Pages

TESTING REQUIREMENTS FOR ISSUANCE OF A COMBINED LICENSE

Presented by Exelon Generation
November 30, 2001

1.0 Introduction and Purpose

The purpose of this paper is to summarize the requirements related to testing, including the testing requirements for issuance of a combined construction permit and operating license (COL). In particular, this paper demonstrates that NRC regulations do not require full-scale prototype testing for issuance of a COL. This paper also explains that a substantial amount of operating experience and test data already exist on pebble bed reactors and fuel, and that Exelon plans to determine whether additional tests are needed to support licensing of the Pebble Bed Modular Reactor (PBMR). Exelon requests feedback from the NRC regarding the positions in this paper related to testing.

2.0 Summary of Regulatory Requirements for Testing

Under the applicable provisions in 10 CFR § 50.57 and § 52.83, the NRC may issue an operating license (OL) or a COL only if the proposed facility will operate in conformity with the provisions of the Atomic Energy Act and the rules and regulations of the NRC, and if there is reasonable assurance that the activities authorized by the license can be conducted without endangering the health and safety of the public. 10 CFR Part 50 and Part 52 do not require the performance of tests as a prerequisite to issuance of an OL or COL. However, an applicant may elect to perform tests in order to support the findings required by Sections 50.57 and 52.83.

In some cases, there may be sufficient information for the NRC to make the required findings under 10 CFR § 50.57 and § 52.83, but additional testing may be desirable to provide additional assurance or confirm some of the bases for the findings. Such testing is generally designated as "confirmatory" testing and can be performed before or after issuance of the license.

In contrast, 10 CFR Part 50 does require certain tests during construction, startup, and operation. Under 10 CFR Part 52, some of the construction and pre-operational tests are designated as inspections, tests, analyses, and acceptance criteria (ITAAC) for a COL. Under 10 CFR § 52.103, the ITAAC must be satisfactorily completed in order for a licensee to load fuel. Additionally, 10 CFR Part 52 states that certain testing may be required as a prerequisite to issuance of a design certification. Table 1 identifies the type of testing of structures, systems, and components (SSCs) required by NRC's regulations, and summarizes when the testing must be performed.

TABLE 1
TYPES OF TESTS REQUIRED BY NRC REGULATIONS

TYPE OF TESTING	PREREQUISITE TO COL ISSUANCE ?	APPLICABLE TO THE PBMR ?
1. **Quality Assurance (QA) Inspections and Tests** – NRC regulations require inspections and tests to ensure that construction of a nuclear plant satisfies quality standards: 10 CFR Part 50, Appendix B, Criteria II and X; 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 1; and 10 CFR § 50.55a(a)(1).	No. By definition, these tests can only be performed on the as-constructed plant following issuance of the COL. Additionally, neither the cited regulations nor 10 CFR § 50.34 requires a COL application to provide the results of these tests.	Yes.
2. **Qualification Tests** – NRC regulations require qualification tests and/or analyses to ensure that procured components will be able to perform under applicable environmental and seismic conditions: 10 CFR § 50.49(f); 10 CFR Part 50, Appendix S; 10 CFR Part 100, Appendix A.	No. Qualification tests must be performed prior to installation, not issuance of the COL. Additionally, neither the cited regulations nor 10 CFR § 50.34 requires a COL application to provide the results of these tests.	Yes.
3. **Proof Tests** – Criterion XI of Appendix B to 10 CFR Part 50 states that proof tests shall be performed as appropriate prior to installation. Additionally, Criterion III states that where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, it shall include suitable qualification testing of a prototype of the item under the most adverse design conditions.	No. Proof tests must be performed prior to installation, not issuance of the COL. Additionally, neither the cited regulations nor 10 CFR § 50.34 requires a COL application to provide the results of these tests.	Yes.

TYPE OF TESTING	PREREQUISITE TO COL ISSUANCE ?	APPLICABLE TO THE PBMR ?
4. **Preoperational Tests** – NRC regulations require preoperational tests to ensure that the functional performance of the as-built plant conforms to the design: 10 CFR § 50.34(b)(6)(iii); 10 CFR Part 50, Appendix B, Criterion XI.	No. By definition, these tests can only be performed on the as-constructed plant following issuance of the COL. Additionally, neither the cited regulations nor 10 CFR § 50.34 requires a COL application to provide the results of these tests.	Yes.
5. **Tests of particular types of structures, systems, and components (SSCs)** – NRC regulations require tests of the reactor coolant pressure boundary and containment to ensure the adequacy of fabrication and construction: GDC 14 and 30; 10 CFR Part 50, Appendices G and J.	No. By definition, these tests can only be performed on the as-constructed plant following issuance of the COL. Additionally, neither the cited regulations nor 10 CFR § 50.34 requires a COL application to provide the results of these tests.	Yes.
6. **Inspections, Tests, Analysis, and Acceptance Criteria (ITAAC)** – 10 CFR § 52.47(a)(1)(vi) and § 52.79(c) state that design certification and COL applications must contain proposed tests, inspections, analyses, and acceptance criteria to provide reasonable assurance that the plant is built and will operate in accordance with applicable requirements.	No. By definition, these tests can only be performed on the as-constructed plant following issuance of the COL. Additionally, neither the cited regulations nor 10 CFR § 50.34 requires a COL application to provide the results of these tests.	Yes.
7. **Startup and Power Ascension Tests** – NRC regulations require startup and power ascension tests to ensure that the functional performance of as-built plant conforms to the design. 10 CFR § 50.34(b)(6)(iii); 10 CFR Part 50, Appendix B, Criterion XI.	No. By definition, these tests can only be performed on the as-constructed plant following issuance of the COL. Additionally, neither the cited regulations nor 10 CFR § 50.34 requires a COL application to provide the results of these tests.	Yes.

TYPE OF TESTING	PREREQUISITE TO COL ISSUANCE ?	APPLICABLE TO THE PBMR ?
8. **Inservice, Surveillance, and Periodic Tests** – NRC regulations require inservice, surveillance, and periodic tests. 10 CFR §§ 50.36(c)(3) and 50.55a(f), and various GDC and other regulations.	No. By definition, these tests can only be performed on the as-constructed plant following issuance of the COL. Additionally, neither the cited regulations nor 10 CFR § 50.34 requires a COL application to provide the results of these tests.	Yes.
9. **Tests for Design Certification of Advanced Reactors** - 10 CFR § 52.47(b)(2) states that an application for design certification of a non-light water reactor must demonstrate the performance of each simplified, inherent or passive safety feature through either analysis, appropriate test programs, experience or through acceptable testing of an appropriately sited, full-size, prototype of the design.	No. As explained elsewhere in this paper, these tests are only a requirement for design certification and not to support licensing. Additionally, neither the cited regulations nor 10 CFR § 50.34 requires a COL application to provide the results of these tests.	No. As explained elsewhere in this paper, these tests are only a requirement for design certification and are not needed to support licensing.

With the exception of the last type of tests discussed in Table 1 (i.e., tests that may be needed for design certification of advanced reactors), all of the tests described in Table 1 pertain to procurement, construction, or operation of a reactor. Thus, by their very nature, such tests do not need to be (and, with the possible exception of the qualification and proof tests, cannot be) completed prior to issuance of a COL. The COL application for the PBMR will describe the test programs for complying with the regulatory requirements for procurement, construction, and operational tests.

3.0 Applicable Regulations, Guidance and Precedents on Full-Scale Prototype Testing

3.1 NRC Regulations

As summarized above, 10 CFR Part 52 requires full-scale prototype testing in some cases for design certification, but does not require full-scale prototype testing to support issuance of a COL.

In particular, Subpart B to Part 52 governs design certification of standard designs. Subpart B contains provisions related to full-scale prototype testing. For standard designs that differ significantly from evolutionary light water reactors (LWRs) or that utilize simplified, inherent, passive, or other innovative means for accomplishing safety functions, 10 CFR § 52.47(b)(2)(i) requires that an applicant utilize one of the following two methods to demonstrate the acceptability of the standard design:

A) "[A]nalysis, appropriate test programs, experience, or a combination thereof." If this method is used, the following factors must be also present: 1) interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof; 2) sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions; and 3) the scope of the design is complete except for site-specific elements such as the service water intake structure and the ultimate heat sink.

or

B) "[A]cceptable testing of an appropriately sited, full-size, prototype of the design over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions."

Thus, for design certification of an advanced reactor such as the PBMR, the regulations do not necessarily require full-size prototype testing; such testing is only necessary if the

acceptability of the design cannot be demonstrated through "analysis, appropriate test programs, experience, or a combination thereof."

Subpart C to Part 52 sets forth the requirements and procedures applicable to issuance of COLs for nuclear power plants. Subpart C does not require full-scale prototype testing. In particular, the various types of technical information required for a COL application are set forth in 10 CFR § 52.79, which does not mention full-scale prototype testing. Although Section 52.79(b) does incorporate certain portions of Subpart B by reference, it does not incorporate those portions pertaining to full-scale prototype testing. Specifically, Section 52.79(b) states: "If the application does not reference a certified design, the application must comply with the requirements of § 52.47(a)(2) for level of design information, and shall contain the technical information required by §§ 52.47(a)(1)(i), (ii), (iv), and (v) and (3), and, if the design is modular, § 52.47(b)(3)." None of these referenced subsections discusses full-scale prototype testing. Instead, the prototype testing requirement is set forth at Section 52.47(b)(2)(i)(B), which is not one of the subsections enumerated in Section 52.79. Thus, while some parts of Section 52.47 have been identified as applicable to the COL requirements in Subpart C, the portion containing the full-scale prototype testing requirement has not.

Except for its references to Section 52.47, Subpart C states that a COL application shall be subject to the same technical standards applicable to a construction permit (CP) or OL. (10 CFR § 52.81). Full-scale prototype testing is not required for a CP or OL under 10 CFR Part 50 or other applicable regulations. Since full-scale prototype testing is not required for a CP or OL, it also is not required for a COL.

The fact that Section 52.79 does not currently reference Section 52.47(b)(2)(i) and does not require full-scale prototype testing was no mere oversight - - it was intentional. The Statements of Consideration for both the proposed Part 52 (53 Fed. Reg. 32060 (August 23, 1988)) and final Part 52 (54 Fed. Reg. 15372 (April 18, 1989)) clearly indicate that design certification and licenses are to be treated differently with respect to prototype testing. For example, in issuing the proposed and final versions of Part 52, the Commission stated:

- "Certification of a reactor design which differs significantly from a reactor design which has been built and operated may be granted only after the design has been shown to be sufficiently mature." (53 Fed. Reg. at 32063-64)

- In order to demonstrate maturity, "[p]rototype testing is likely to be required for certification of advanced non-light water designs." (54 Fed. Reg. at 15375)

- In contrast, the Commission recognized that it may "licens[e] the prototype for commercial operation." (54 Fed. Reg. at 15374)

- Furthermore, the Commission expressly rejected a proposal that would allow a COL to be issued only for a standard design, stating: "The final rule does not contain this restriction because there may be circumstances in which a combined license would

properly utilize a non-standard design and because such a restriction would mean, among other things, that every prototype would have to be licensed in a fully two-step process." (54 Fed. Reg. at 15383)

- Thus, "[i]t is well to remember also that, under the rule, prototype testing is only required for certification or an unconditional final design approval, if at all." (54 Fed. Reg. at 15374).

In summary, 10 CFR § 52.47(b)(2) contains provisions for full-scale prototype testing prior to certification of a standard design, because the Commission wanted to ensure that only mature designs are certified. In contrast, the Commission deliberately did not impose such a requirement for COLs, because it wanted the flexibility to license a non-standardized plant that may not have a mature design. In fact, the Commission wanted to be able to issue a COL for the prototype plant itself.

As indicated above, the Commission has stated that full-scale prototype testing will likely be required for design certification of advanced reactors. The reason for such a requirement is readily apparent - - a certified design is effective for 15 years, may be incorporated by reference by any license applicant without further review and approval by the NRC, and is subject to broad protection against backfits under the change control process in 10 CFR § 52.63. Because a certified design is not subject to further NRC review and approval and has broad backfit protection, there is a sound basis for requiring that the maturity of the design be fully demonstrated before certification (e.g., by restricting design certification of advanced reactors to those designs that have successfully completed prototype testing).

In contrast, the arguments for requiring full-scale prototype testing to support *certification* of advanced reactors do not apply to *licensing* of advanced reactors. Unlike a design certification, licensing represents approval of only a single facility. Licensing of subsequent facilities, even if identical in design, is still subject to NRC review and approval including possible design changes to account for any unfavorable results of startup and power ascension testing and operating experience from previously licensed facilities. Furthermore, unlike a design certification, the NRC has fairly broad authority under 10 CFR § 50.109 to impose backfits on a licensed facility to account for any unfavorable results of startup and power ascension testing and operating experience. Finally, in lieu of full-scale prototype testing, the NRC has authority to impose special license conditions that might not be necessary or appropriate if applied to all plants with a standard design (e.g., a license condition can require special design, procedural, or testing provisions to provide adequate protection of safety until the design is demonstrated to be safe through testing or operation). Thus, unlike a design certification, there is no compelling reason to require full-scale prototype testing to support licensing of an advanced reactor.

3.2 <u>NRC Guidance</u>

NRC guidance documents also support the conclusion that full-size prototype testing is not needed to support issuance of a COL.

SECY-91-074, *Prototype Decisions for Advanced Reactor Designs*, does not require full-scale prototype testing to support issuance of a COL. Instead, it only discusses prototype testing to support design certification. In particular, SECY-91-074 establishes an approach called "certification-by-test" in order to demonstrate that advanced reactor designs "are sufficiently mature to be certified."

Furthermore, even with respect to design certification, SECY-91-074 does not necessarily require a full-scale prototype test. Under the guidance of SECY-91-074, a full-scale or prototype test is <u>only</u> required if scale testing or partial plant tests are impossible. *See* SECY-91-074, Enclosure 2, p. 8 and Figure 1. In this regard, SECY-91-074 states:

> "The types of possible testing include tests of components, systems, simulators, non-nuclear of nuclear test loops, and comprehensive prototypes for determining proof of principle. The applicant may consider the least burdensome type of testing that provides the safety-related insights required to substantiate the applicant's bases. For instance, the applicant may consider component testing first and only consider the most burdensome type of testing (the testing of a full-scale prototype) as a last resort." (*Id.*, p. 1)

The statement of consideration for the Commission's 1986 Policy Statement on Regulation of Advanced Nuclear Power Plants (51 Fed. Reg. 24,643) also addresses prototype testing. It states:

> "The Commission requires proof of performance of certain safety-related components, systems, or structures prior to issuing a license on a design. For LWR's this proof has traditionally been in the form of analysis, testing, and research development sufficient to demonstrate the performance of the item in question. Similar proof of performance for certain components, systems or structures for advanced reactors will also be required. The requisite proof will be design dependent. Therefore, the Commission's specific assessment of a safety technology development program for an advanced reactor design, or of the possible need for a prototypical demonstration of that design can be determined only by review of a specific design. However, the Commission favors the use of prototypical demonstration facilities as an acceptable way of resolving many safety related issues."

This statement clearly indicates that while the Commission favors full-scale prototype testing for advanced reactors, the need for such testing is design dependent and that proof of performance of structures, systems, and components may be in the form of analysis, testing, and research development similar to the proof traditionally offered for LWRs.

NUREG-1226, *Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants* (June 1988), p. 5-10, also notes that the NRC's Advanced Reactor Policy Statement "does not require a priori that a prototype test reactor be constructed and operated." Although the NRC staff favors the use of prototype full-scale test facilities to demonstrate those features that are fundamental to the safety performance of designs that significantly depart from proven technology, the NUREG also states that:

> "The staff will make a case-by-case judgment about the need for a prototype test to resolve safety issues considering such factors as:
> (1) Departure from proven technology
> (2) Uncertainties in performance and how they could be reduced
> (3) Degree of defense-in-depth, and
> (4) Other R&D programs planned in support of the design."

Furthermore, guidance applicable to issuance of a CP or OL does not require full-scale prototype tests. Although some of the guidance does discuss prototype testing, these references generally occur in the context of seismic or environmental qualification testing of individual components, or apply to specific systems or structures (e.g., Regulatory Guide 1.20, *Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing,* calls for vibration testing of a prototype of the reactor internals; and NUREG-0800, Standard Review Plan 4.2, Section II.C.2, calls for prototype structural tests for control rods and fuel assemblies).

Finally, NRC Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition*, explicitly states that special, unique, or first of a kind design features may be verified through preoperational or startup tests:

> "14.1.2 Plant Design Features That Are Special, Unique, or First of a Kind
>
> A summary description of preoperational and/or startup testing planned for each unique or first-of-a-kind principal design feature should be included in the PSAR. The summary test descriptions should include the test method and test objectives."

Similarly, NUREG-0800, Standard Review Plan 14.2, Section III.8 recognizes that the initial test program in Final Safety Analysis Reports may include provisions for "testing for special, unique, or first-of-a-kind design features." Thus, NRC guidance clearly allows for testing of unique and first-of-a-kind design features through the startup and power ascension test program, and does not require prototype testing.

3.3 Precedents

NRC precedents confirm that full-scale prototype testing is not required to support licensing of non-LWRs or advanced reactors.

For example, a construction permit and operating license was issued for Peach Bottom Unit 1, which was a 40 MWe prototype high-temperature gas-cooled reactor (HTGR). Peach Bottom Unit 1 was licensed, despite the fact that there was no previous experience with high-temperature gas-cooled reactors in the United States.

Similarly, a construction permit and operating license was issued for the Ft. St. Vrain Nuclear Generating Station, a 330 MWe HTGR demonstration unit that was substantially larger than Peach Bottom Unit 1 and contained significant design changes. Ft. St. Vrain was licensed without a prior full-scale prototype test. Instead, the construction permit for Ft. St. Vrain relied upon test elements irradiated in other reactors at expected temperatures, not prototype data, to demonstrate its fuel's safety. *E.g.*, Preliminary Safety Analysis Report, Ft. St. Vrain, Docket 50-267, p. 1.4-2 (Sept. 1966). Other examples of non-full scale prototype tests used by Ft. St. Vrain included relying on studies of steam-graphite reactions, hydraulic tests of prestressed concrete reactor vessels, and prototypes of individual components only, such as control rod drive mechanisms and helium circulators. *Id.* at 1.4-3 to 1.4-4. The Atomic Energy Commission approved issuance of the operating license for Ft. St. Vrain based upon factors such as the results experiments on the fuel particles and tests of prototype fuel, analytical models of the core physics, and computer programs and half-scale thermal-hydraulic tests of helium flow through a portion of a reactor core. Safety Evaluation for Fort St. Vrain (January 20, 1972), pp. 9-10, 13-14.

Likewise, for the Clinch River Breeder Reactor Plant, data gathered from a test reactor, rather than a full-scale prototype, was used as the basis for portions of its Safety Evaluation Report for the construction permit. Clinch River relied heavily on data obtained from the Fast Flux Test Facility for its mechanical systems, core, instrument and electrical systems, auxiliary liquid metal, and other design aspects. Safety Evaluation Report (Clinch River Breeder Reactor Plant), Docket No. 50-537, NUREG-0968 at 1-7 (March 1983).

The NRC also determined that full-scale prototype testing would not be needed for design certification of the CANDU 3 reactor (an advanced reactor using heavy-water technology). In SECY-90-133, *Prototype Requirement for CANDU 3 Design* (April 6, 1990), the NRC stated that "a prototype for the CANDU 3 design will not be required because the CANDU 3 design is based on proven heavy water technology in Canada and because plans exist to construct a commercial CANDU 3 reactor" in Canada. Instead, in SECY-93-092, *Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements*, p. 4, the NRC determined that novel safety features of the CANDU 3 and other advanced reactors could be demonstrated through analysis, test programs, experience, or a combination of these methods.

3.4 Other Considerations

The NRC could create a logically impossible situation if it were to require completion of full-scale prototype testing prior to issuance of a COL. In such a situation, the applicant could not get a license to build, start-up, and test the reactor until completion of prototype testing, but prototype testing could not occur without the license. Thus, if full-scale prototype testing were required to support issuance of a license, the applicant would be subjected to a *Catch-22*. Clearly, this is untenable and is not the intent of the regulations.

This principle is exemplified by NRC's handling of prototype testing for the Modular High Temperature Gas Reactor (MHTGR). In the Preapplication Safety Evaluation Report for the MHTGR, NUREG-1338 (December 1995), p. 4-5, the NRC stated that full-scale prototype testing might be necessary for design certification of the MHTGR. However, prior to performance of the prototype testing, NRC indicated that it would issue a license for the prototype MHTGR "based on a higher postulated fuel failure and lower leakage containment" than proposed for design certification of the MHTGR.

As indicated by the NRC's position on the MHTGR, in a licensing proceeding, the Commission has alternatives other than the methods specified in Section 52.47(b)(2)(i) (i.e., other than analysis, testing, experience, or prototype testing) for ensuring the safety of the plant. In particular, the Commission may impose special design or monitoring requirements, or may establish special license conditions, to ensure the safety of the plant pending completion of startup and power ascension testing. While such requirements would not be appropriate to apply on a generic basis to all plants through a design certification, they could be applied in a licensing proceeding as an alternative to Section 52.47(b)(2)(i) for ensuring the safety of a licensed plant.

* * *

In summary, NRC regulations, guidance, and precedent all indicate that full-scale prototype testing may be necessary in some cases for design certification of an advanced reactor but is not necessary to support issuance of a COL for an advanced reactor. For some advanced reactors, such as the CANDU 3, NRC has stated that the design could be certified without testing on a full-scale prototype, given prior relevant operating experience with similar designs. Additionally, the Commission's policy statement on advanced reactors explicitly recognizes that the need for prototype testing can only be determined based upon a review of a specific design. Therefore, it is clear that full-scale prototype testing is not required in all cases for design certification of advanced reactors, and there is no requirement for full-scale prototype testing to support issuance of a COL.

4.0 Testing for the PBMR

4.1 Existing Experience and Test Data for Pebble Bed Reactors

As summarized below, the PBMR has significant testing and operating experience that can be applied to support its design:

- U.S. HTGRs, such as Peach Bottom Unit 1 and Ft. St. Vrain, which demonstrated in general the safety of high-temperature gas-cooled reactor technology.

- The AVR, a 15 MWe pebble bed reactor, operated in West Germany for 20 years. This experience produced extensive data, allowing not only empirical and experimental validation, but also analytical modeling.

- The THTR, a 300 MWe demonstration pebble bed reactor, operated in West Germany for 3 years. This full-size reactor provides valuable operating experience and test data for a reactor that is larger than the PBMR.

- Another pebble bed reactor, the HTR-10, has recently begun to operate in China.

- A number of tests were performed on the AVR fuel.

- Several pebble flow experiments (including experiments with some full-scale models) have been conducted to verify pebble flow models for the AVR, THTR, and HTR reactors.

The PBMR will be very similar in design to the other pebble bed reactors. In fact, the PBMR fuel will have the same specifications as the AVR fuel, while applying innovations that have enhanced safety and function (e.g., controls rods within the core barrel rather than through the fuel spheres). Therefore, existing experience and test data can be used to support licensing of the PBMR.

4.2 Exelon's Plans to Evaluate Whether Additional Testing Is Needed to Support Licensing of the PBMR

The COL application will use the existing body of experience and test data as part of the basis to support licensing of the PBMR. Additionally, Exelon will develop a process for determining whether additional testing is needed to support licensing of the PBMR. In particular, Exelon will develop criteria to determine whether existing information is adequate to support licensing, or whether there is a need for any of the following actions to address any areas in which existing information is not sufficient to support licensing:

- additional analysis using existing information, or
- additional full-scale or partial-scale testing of particular structures, systems, or components (including the fuel), or
- special design provisions or license conditions.

Exelon will provide these criteria to the NRC for its review and feedback. Based upon these criteria, Exelon will then determine whether there is a need for additional actions and will request feedback from the NRC on the plans for these actions.

Currently, it is expected that a PBMR demonstration unit will be constructed in the Republic of South Africa (RSA), and that the demonstration unit will be subject to testing. However, much of this testing is scheduled to occur after licensing of the PBMR in the United States. Furthermore, testing of the RSA demonstration unit does not appear to be needed to support licensing of the PBMR in the United States. If Exelon's conclusions change based upon implementation of the process discussed above, Exelon will notify the NRC.

Assuming that its conclusions do not change, Exelon will designate the testing on the PBMR demonstration unit as confirmatory in nature; i.e., the testing will confirm the adequacy of the design as based upon existing operating experience with pebble bed reactors and existing test data.

5.0 Conclusions

NRC regulations and guidance do not require completion of full-scale prototype testing prior to issuance of a COL. Furthermore, even with respect to design certification of advanced reactors, NRC's regulations and guidance only require prototype testing if existing analyses, testing, and experience are not sufficient to demonstrate the safety of the design.

In the case of the PBMR, there is extensive test data and experience based upon operation of the AVR and THTR in Germany and the HTR-10 in China. Exelon will develop a process for determining whether additional testing is needed to support licensing of the PBMR.

6.0 Request for NRC Feedback

Exelon requests the NRC to provide feedback on the following issues related to this paper:

1. Does NRC agree that its regulations do not require completion of full-scale prototype testing as a prerequisite to issuance of a COL for an advanced reactor?

2. Does NRC agree that an advanced reactor may be licensed based upon experience, test data, analyses, and/or special design or license conditions?

3. Does NRC agree with Exelon's proposal to develop a process and criteria for identifying whether additional testing, analyses, or special design or license conditions are necessary for licensing of the PBMR?