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Next Generation Nuclear Plant (NGNP) Project – Preliminary Assessment of Two Possible Designs

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Abstract - *This paper provides a preliminary assessment of two possible versions of the Next Generation Nuclear Plant (NGNP), a prismatic fuel type helium gas-cooled reactor and a pebble-bed fuel helium gas reactor. Both designs will meet the three basic requirements that have been set for the NGNP: a coolant outlet temperature of 1000 °C, passive safety, and a total power output consistent with that expected for commercial high-temperature gas-cooled reactors.*

I. INTRODUCTION

In the coming decades, the United States, the other industrialized countries, and the entire world will need energy supplies and an upgraded energy infrastructure to meet growing demands for electric power and transportation fuels. The Generation IV project identified reactor system concepts for producing electricity that excelled at meeting the goals of superior economics, safety, sustainability, proliferation resistance, and physical security. [1] One of these reactor system concepts, the Very High Temperature Gas Cooled Reactor System (VHTR), is also uniquely suited for producing hydrogen without the consumption of fossil fuels or the emission of greenhouse gases. DOE has selected this system for the Next Generation Nuclear Power (NGNP) Project, a project to demonstrate emissions-free nuclear-assisted electricity and hydrogen production by about 2017.

The objectives for the NGNP project are [2]

- Demonstrate a full-scale prototype NGNP by the middle of the next decade
- Demonstrate high-temperature Brayton Cycle electric power production at full scale
- Demonstrate nuclear-assisted production of hydrogen (with about 10 % of the heat)
- Demonstrate by test the exceptional safety capabilities of the advanced gas cooled reactors
- Obtain an NRC License to construct and operate the NGNP, to provide a basis for future performance-based, risk-informed licensing of high temperature gas reactors
- Support the development, testing, and prototyping of hydrogen infrastructures such as refueling stations, the "Freedom Car" initiative, petrochemical extension, heavy crude oil or tar sands "sweetening," and other industrial hydrogen applications

The NGNP reference concepts are helium-cooled, graphite-moderated, thermal neutron spectrum reactors with a design goal outlet temperature of 1000 °C or higher. The reactor core could be either a prismatic graphite block type core or a pebble-bed core. The use of a molten-salt coolant is also being evaluated. The NGNP will produce both electricity and hydrogen. The process heat for hydrogen production will be transferred to the hydrogen plant through an intermediate heat exchanger (IHX). The reactor thermal power and core configuration will be designed to assure passive decay heat removal without fuel damage during hypothetical accidents. The fuel cycle will be a once-through very high burnup low-enriched uranium fuel cycle.

This paper provides a preliminary assessment of two possible versions of the Next Generation Nuclear Plant (NGNP), one is a prismatic fuel type helium gas-cooled reactor and one is a pebble-bed fuel helium gas reactor. [3] Both designs are to meet three basic requirements: a

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[b] General Atomics

coolant outlet temperature of 1000 °C, passive safety, and a total power output consistent with that expected for commercial high-temperature gas-cooled reactors. [4] The two efforts are discussed separately below. The analytical results are very promising; however, we wish to caution the reader that future, more detailed, design work will be needed to provide final answers to a number of key questions including the appropriate power level, the inlet temperature, the power density, the optimum fuel form, and others.

II. PRISMATIC BLOCK NGNP

The prismatic NGNP reactor is essentially a large graphite pile composed of hexagonal blocks. Approximately one-third of these blocks are fuel blocks arranged in an annular core, and the remaining two-thirds of the blocks are graphite blocks arranged to form inner and outer neutron reflectors about the annulus. During transients, the graphite reflector mass acts as an important temporal heat sink and storage device to maintain fuel temperatures below values that may damage the fuel (i.e. temperatures above 1600 °C). The blocks are stationary during reactor operation, but at the end of each power cycle, every block can be replaced if needed, thus allowing for the ability to rebuild a new core pile at regular intervals and eliminate the material damage effects due to long-term neutron irradiation and high temperatures. The annular geometry of the core ensures inherent safety under transient conditions by facilitating the conduction and radiation of the decay heat to the containment cavity cooling system.

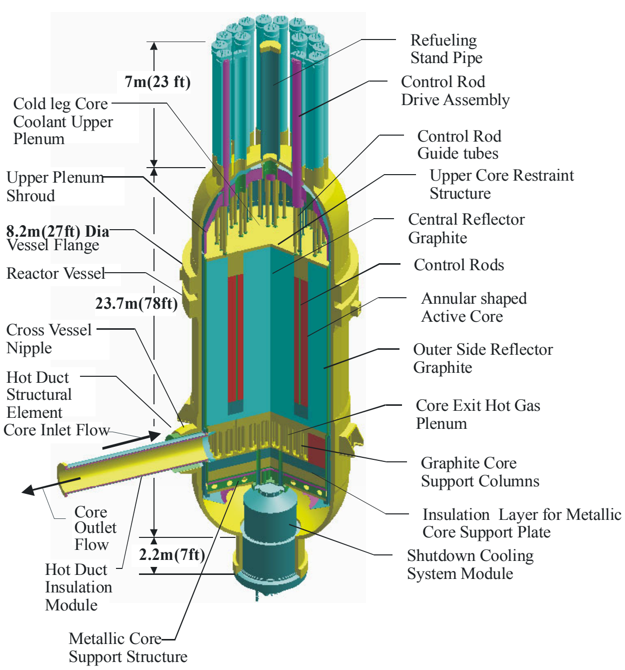

Fig. 1. GT-MHR reactor vessel cutaway showing the arrangement of the reactor components

The prismatic NGNP is an evolutionary design with roots stemming in the Fort Saint Vrain high-temperature gas-cooled reactor design and the recent General Atomics Very High Temperature Gas-cooled Reactor (VHTR) design submittal to the Generation IV Roadmap, which was based on their gas turbine-modular helium reactor (GT-MHR) design shown in Figures 1 and 2. [5] Modifications of the GT-MHR design have been developed in order to meet the NGNP design requirement of inherent safety and the NGNP design goal of a 1000 °C outlet helium gas temperature.

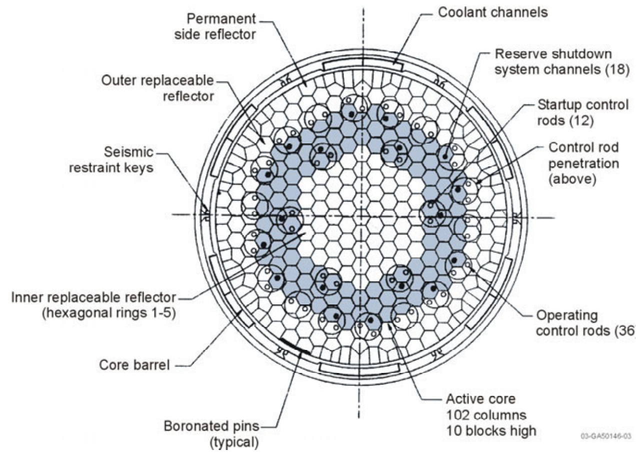

Fig. 2. Cross sectional view of the GT-MHR and NGNP cores.

II.A. Prismatic Block Thermal—Hydraulic Design

Parametric thermal-hydraulic design studies were performed using the POKE [6] computer code[c]. POKE was used to calculate the flow distribution in 1/3 of the core; the temperatures of the coolant, graphite, and fuel at each axial block location for each column; the axial pressure distribution in each column; and the overall pressure drop across the core. The POKE modeling of the heat transfer within a block is shown in Fig. 3. The power distribution was based on 3-D core-physics calculations for the 600-MWt GT-MHR, fueled with low-enriched uranium and operating at the middle of an equilibrium cycle. The primary purpose of these studies was to investigate design options for the prismatic NGNP that would allow (1) an outlet temperature of 1000 °C, (2) the lowest possible inlet temperature, and (3) the highest possible overall core power, while maintaining the peak fuel temperatures during normal operation at an acceptable level of about 1250 °C. (A general "rule of thumb" is that fuel performance and fission-product release in a high-temperature gas-cooled reactor with SiC TRISO coated

[c] The NGNP thermal-hydraulic design analyses were performed by John Bolin, Matthew Richards, Alan Baxter and Malcoom LaBar at General Atomics.

fuel will be acceptable if the peak fuel temperature during normal operation remains below about 1250 °C.)

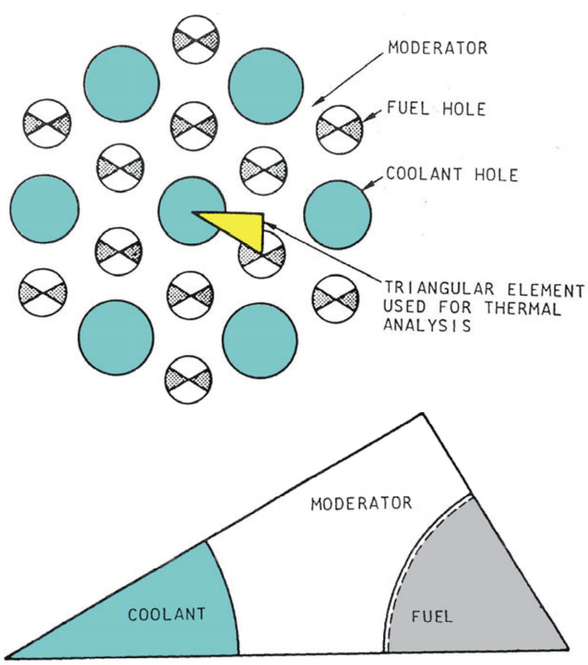

Fig. 3. Unit cell used for the thermal analysis of the NGNP.

Table 1. Effects of controlling flow distribution in the NGNP.

	None	Optimized by POKE	Optimized by POKE
Inlet Temperature (°C)	641	641	491
Flow Rate (kg/s)	320	320	226
Average Outlet Temperature (°C)	1000	1000	1000
Max Fuel Temperature (°C)	1309	1204	1239
Max Outlet Temperature (°C)	1124	1030	1042
Core Pressure Drop (psid)	10.0	14.5	6.9

The study began with an analysis of the current 600 MWt GT-MHR design operating with a coolant inlet temperature of 491 °C, an average coolant outlet temperature of 850 °C, a coolant flow rate of 320 kg/s, a bypass flow fraction of 0.2, and conventional column-by-column refueling. Two major design modifications were then evaluated: reducing the bypass flow and better controlling the inlet coolant flow distribution to each block column. Reducing the bypass flow fraction from 20 to 10% reduces peak fuel temperatures by about 50 °C and reduces coolant channel hot streaks by about 75 °C. Controlling the inlet flow distribution has an even more dramatic effect on reducing the maximum fuel temperatures and coolant hot streaks as shown in Table 1. The results indicate that a NGNP with these or other potential design modifications can have an outlet temperature of 1000 °C and fuel temperatures similar (same peak temperatures, slightly higher volumetric average temperatures) to the GT-MHR design. Also, controlling the flow distribution allows for reducing the coolant inlet temperature and coolant flow rate, such that the operating temperature for the reactor vessel (490 °C) and the core pressure drop for the NGNP would be about the same as that for the reference GT-MHR.

Taller and higher-power reactor cores were also evaluated with the POKE computer code. The power density was kept the same as that for the 10-block-high, 600-MWt core, since this parameter has a strong effect on core temperature response during accident conditions. Both 12-block-high (720 MWt) and 14-block-high (840 MWt) cores were evaluated. For the higher-powered cores, the coolant flow rate was increased in proportion to the power level, in order to maintain the same coolant temperature rise as the 600 MW_t core. It was determined that the higher-powered cores will operate with about the same fuel and graphite temperatures as the 600 MWt core.

II.B. High And Low Pressure Conduction Cooldown Accident Analyses

Analyses were performed to determine the peak reactor vessel and fuel temperatures during high and low pressure conduction cool-down (HPCC and LPCC) accidents and thereby identify the allowable core power.[d] The calculations were done with the RELAP5-3D/ATHENA computer code. [7] Fig. 4 illustrates the convective, conductive, and radiative heat transfer modeled between the various structures and the coolant. The reactor fuel was modeled as being in 102 blocks on each level (see Fig. 2), with 10, 12 or 14 levels in the active core (the 10-block high core is the base case). The block height was 0.793 m, yielding an active core height of 7.93, 9.52, or 11.10 m. The core outer diameter was 4.8393 m. The inner ring contains 30 assemblies, and the middle and outer rings each contain 36 (the six corner assemblies in the outer ring are not fueled). For the initial calculations, estimated radial power factors of 1.10, 0.92, and 1.00 were used for the inner, middle, and outer rings, respectively, and a symmetric chopped cosine axial power profile was used, with a peak-to-average ratio of 1.2. For the maximum power calculations, radial power factors of 0.98, 1.10, and 0.91 were used, with a chopped cosine axial power shape that was slightly skewed toward the top of the core and had a peak-to-average ratio of 1.3. These values were calculated by General Atomics for an equilibrium cycle GT-MHR core.

[d] The NGNP high and low-pressure conduction cooldown analyses were performed by Paul Bayless at the INEEL.

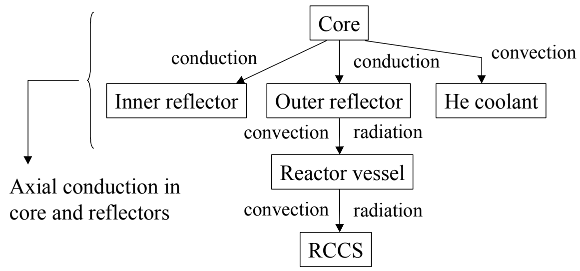

Fig. 4. Heat transfer interactions in the RELAP5-3D/ATHENA NGNP model.

The reactor cavity cooling system (RCCS) designed by General Atomics and Bechtel National was modeled as shown in Fig. 5. [8] Air at 43 °C enters the inlet plenum above the downcomer from the surrounding environment, then flows through the downcomer (which is attached to the containment wall) to the bottom of the reactor compartment, where it is distributed to the riser channels. The hot air leaving the risers is collected in a plenum, then discharged back to the atmosphere. Emissivity values of 0.8 were used for the core barrel, reactor vessel, and RCCS structures. An emissivity of 0.1 was used for the RCCS downcomer wall facing the reactor vessel because it has a reflecting surface with 3 inches of insulation behind the surface.

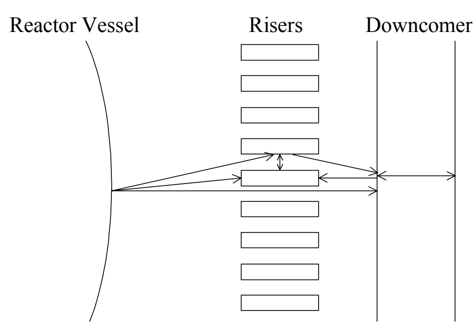

Fig. 5. Radiation paths between the reactor vessel and RCCS. [10]

The RELAP results were first benchmarked against previous high- and low-pressure conduction cooldown transient calculations performed at General Atomics for the GT-MHR. When the appropriate decay heat curve was used, the peak fuel temperatures calculated by RELAP were only slightly below the values reported by General Atomics. The small differences are attributed to the somewhat better convective heat transfer in the bypass regions calculated by RELAP. The code and model were then used to perform analyses of the transient response of the NGNP prismatic core design and determine the effects of core geometry on the peak reactor vessel and fuel temperatures.

A series of calculations was then performed to address how changing the core geometry would impact the transient temperature response of the reactor. The LPCC results are presented here because those results were more severe than the HPCC results. The fueled annulus was kept three blocks wide, but the specific rings occupied by the fuel were varied, as was the total height of the core. The thicknesses of the upper, lower, and outer reflectors were left unchanged from that of the GT-MHR; in the core, only the inner reflector thickness changed as the fuel rings were moved. Outside the core, the core barrel and reactor vessel diameters also changed as the active core diameter varied. The results of the core configuration studies showed that moving the fuel out one ring could significantly reduce the peak fuel temperatures during conduction cooldown transients. However, there are neutronic and manufacturing issues associated with the larger core diameters that need further evaluation if this approach is to be pursued. While the potential reductions are not as large, a more expedient means to reduce the peak transient temperatures is to increase the core height.

The reactor power that can be obtained for different core heights without exceeding a peak transient fuel temperature of 1600 °C during the transient are shown in Figures 6. With a coolant inlet temperature of 490 °C and a 10% nominal core bypass flow, it is estimated that the peak power for a 10-block high core is 686 MWt, for a 12-block high core it is 786 MWt, and for a 14-block core it is about 889 MWt. However, the mechanical and neutronic stability of cores longer than 10 blocks high has not been studied. The Fort Saint Vrain operating experience suggests that such long fuel block columns could potentially move (fluctuate) laterally. The feasibility of laterally supporting the fuel columns between the column ends to prevent lateral column movement has not yet been fully determined.

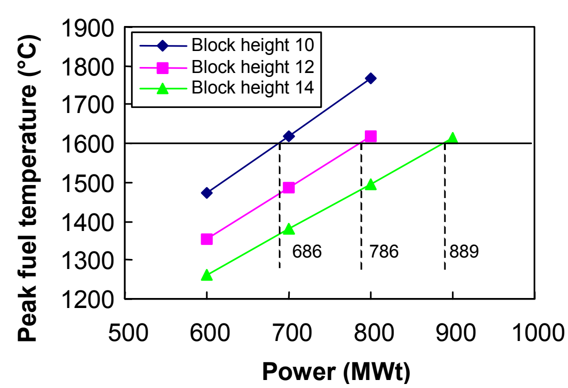

Fig. 6. Maximum fuel temperatures for the LPCC transient with 10% core bypass.

The peak reactor vessel temperature at a given power decreased slightly as the core height increased, as expected, because the power density decreased and more surface area was available to transfer the heat from the reactor vessel to the RCCS. The peak reactor pressure vessel temperature during the low-pressure conduction cooldown event for the three different cases remains below 560 °C, as indicated in Fig. 7.

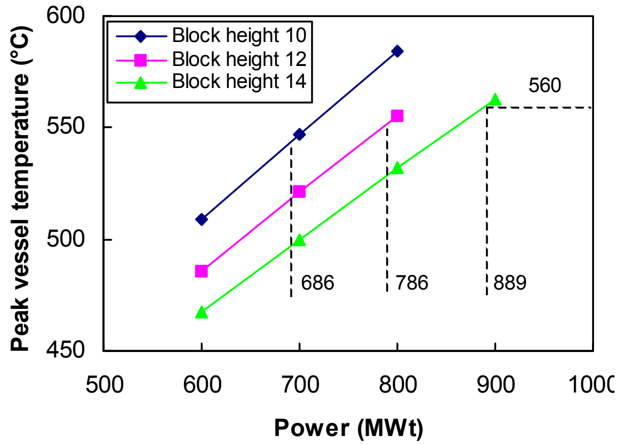

Fig. 7. Maximum reactor vessel temperatures for the LPCC transient with 10% core bypass.

Figures 8 and 9 show the peak fuel and reactor vessel temperatures from most of these calculations. The timing of the peak temperatures appears to be affected more by the total power than by the core height, with higher powers yielding later peak temperatures, for both the fuel and the reactor vessel. For a given power level, the peak temperature occurs later for taller cores. Also, note the relatively long time required for the temperature increases during the LPCC event. The peak fuel temperatures are not reached until about 50 to 60 hours and the peak vessel temperatures are not reached until about 80 hours.

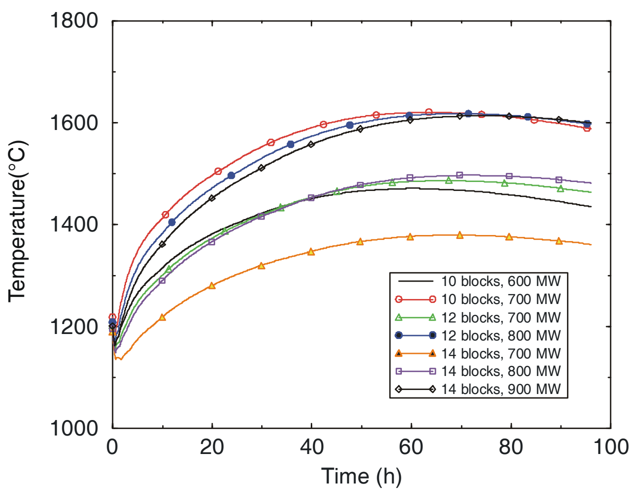

Fig. 8. Peak fuel temperatures during the LPCC transient with 10% core bypass.

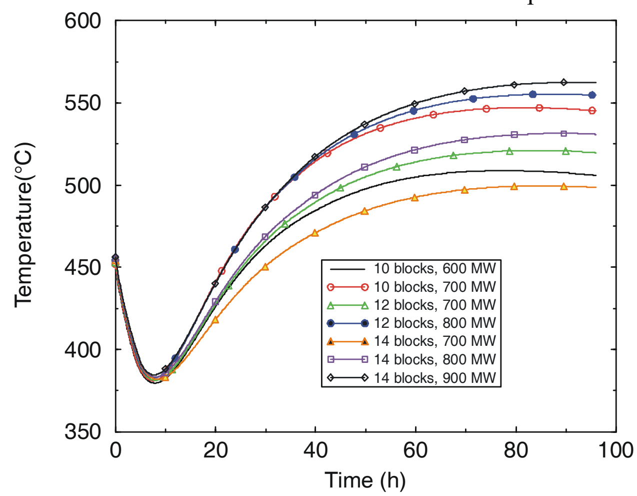

Fig. 9. Peak reactor vessel temperatures during the LPCC transient with 10% core bypass.

II.C. NGNP Prismatic Core Neutronic Design

The reactor physics computer codes MCNP, ORIGEN2, MOCUP and NJOY were used to perform the NGNP neutronic point design neutronic analyses.[e] The MCNP (Monte Carlo N-Particle) code[9] is a general purpose, continuous energy, generalized geometry, coupled neutron-photon-electron Monte Carlo transport code. The geometry capability allows for very explicit, three-dimensional representations of the reactor core and prismatic block details. All of the fuel rods (but not individual fuel particles), coolant channels, and other core features were explicitly defined in the MCNP-ORIGEN block models as shown in Fig. 10. The ORIGEN2 code[10] was used to calculate the complex time-dependent and coupled behavior of both radioactive and stable isotopes under flux irradiation or power production time profiles. This includes the isotopic buildup due to production and destruction mechanisms, which include transmutation (radiative capture), fission, threshold particle reactions, and radioactive decay processes. The MOCUP code[11] was used to link the input and output files from the MCNP and ORIGEN2 codes in order to perform time-dependent burnup or depletion calculations. The NJOY nuclear data processing system[12] was used to produce point-wise and multi-group neutron and photon cross-sections from the ENDF/B evaluated nuclear data.

[e] The NGNP neutronics calculations were performed by James Sterbentz and Robert Sant at the INEEL.

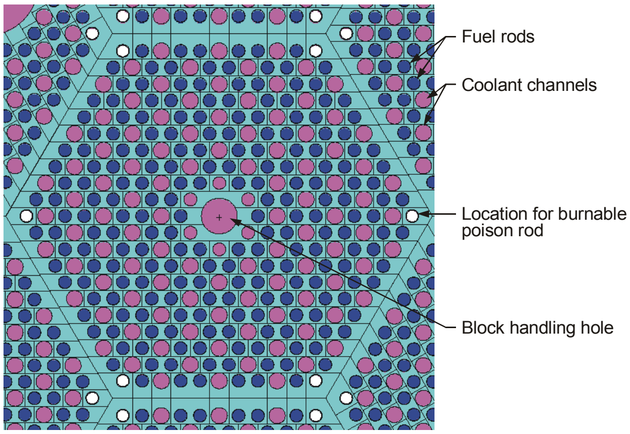

Fig. 10. MCNP infinite lattice model showing a standard NGNP hexagonal fuel block. The six locations on the corners of the hexagonal block can be used to hold burnable poison rods.

The core models shown in Fig. 11 were 1/6-core radial wedge models with reflective boundary conditions applied to the azimuthal planes. Both 1/6-core single block and full core height (including top and bottom reflector blocks) models were run. The reactivity was about the same for the 1/6-core single block and full core height model. The MCNP neutronic evaluations corroborated the results of the previous General Atomics annular GT-MHR design.

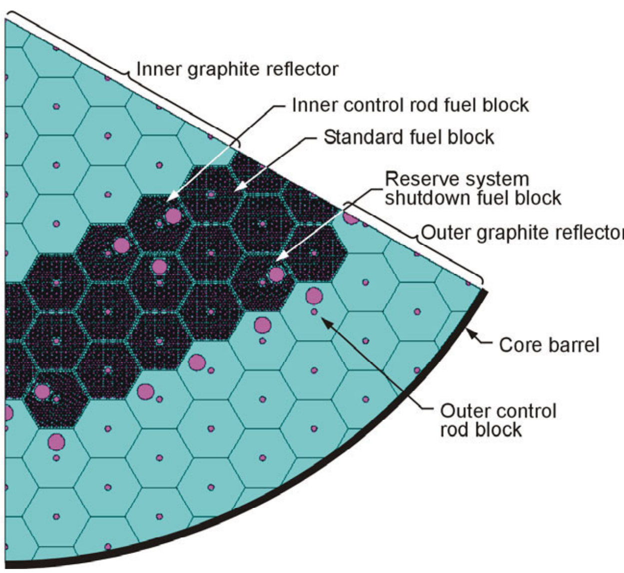

Fig. 11. MCNP model of a 1/6-core NGNP.

The initial core loading achieves a 420-540 effective full power day (14-18 month) design burnup with an initial effective enrichment of about 10 wt% U-235 uniformly distributed across the 3-ring annular core. The equilibrium cycle reload core requires an enrichment of about 15 % U-235. The core also exhibits strongly negative Doppler and isothermal temperature coefficients of reactivity over the burnup cycle. Also, there is a negligible core reactivity change in the event of a rapid loss of the helium gas. However, water or steam ingress can be a problem. Water or steam ingress into the core coolant channels produces a small reactivity effect up to a water density of approximately 0.001 g/cc (18.1 Kg of H_20 in 18 million cc of coolant channels). Greater quantities of water or steam ingress cause a significant reactivity increase as shown in Fig. 12. Complete flooding results in a reduction in reactivity.

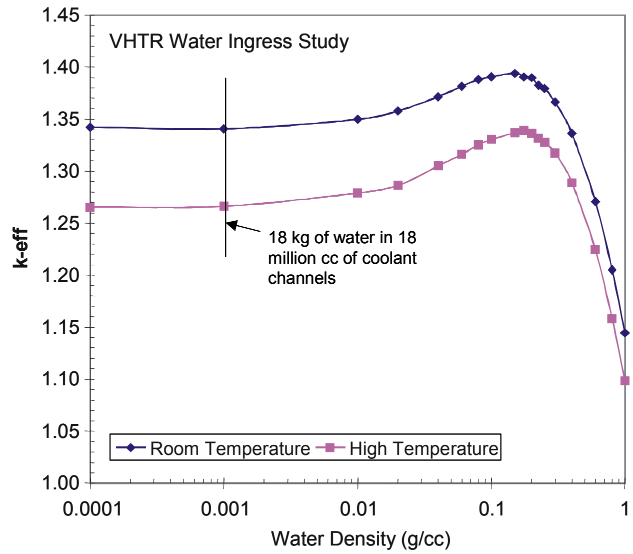

Fig. 12. Core k-effective as a function of water density in the NGNP coolant channels.

An evaluation of the effects on the NGNP reactivity of varying the particle packing fraction and uranium enrichment was conducted. Fig. 13 shows the calculated results. It is apparent that as the enrichment is increased the k-infinity value increases as expected. However, as the packing fraction increases the k-infinity values decrease. This effect can be exploited for the goal of increasing the NGNP power cycle length. The larger packing fractions allow heavier U-235 loading with suppressed reactivity due primarily to thermal neutron self-shielding. Hence, at beginning-of-cycle the reactivity is held down by the self-shielding and later released as the cycle or burnup progresses.

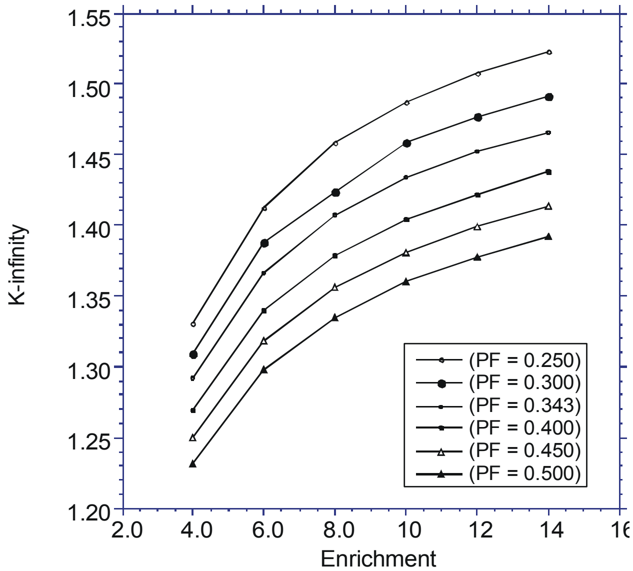

Fig. 13. Standard block lattice k-infinity versus fuel enrichment and particle packing fraction (PF).

An important issue involving the performance of the fuel particles under normal operating conditions is the power peaking of the fuel rods at the annular core interfaces with the inner and outer graphite reflectors, but primarily at the inner reflector interface. Sustained reduction of this power peaking over the power cycles will improve the fuel performance. Fortunately for the prismatic NGNP design, there are a number of possible solutions that can effectively solve this problem. These solutions involve: (1) use of the allocated B_4C burnable poison rod locations in the fuel blocks, (2) graded particle packing fractions in the fuel rod Rows 1, 2, 3, and 4 nearest the interface, (3) graded fuel enrichments in these same Rows 1-4, (4) use of burnable poison (e.g. B_4C) particles in the compacts, and (5) B_4C loaded in various ways in the graphite reflector blocks in Rings 5 and 9 near the reflector/core interfaces. The results of including B_4C burnable poison rods at the core/inner reflector interface, graded particle packing fractions in the fuel rods near the fuel/inner reflector interface, and graded enrichment in the fuel rods near the fuel/inner reflector interface have been assessed to date. The results show that the radial power peaking can be reduced from 1.6 (no mitigation actions) to about 1.3.

III. NGNP PEBBLE-BED REACTOR DESIGNS

The pebble-bed NGNP essentially consists of an annular vat filled with fuel spheres, or “pebbles,” that are dropped in at the top and removed at the bottom, so that they flow slowly through the core region. This design configuration introduces several unique advantages compared to batch fueled reactor designs. Continuous online refueling reduces the frequency of required shutdowns. Reactor shutdown is required only when the portions of the reflectors near the core need to be replaced or when the power conversion equipment needs refurbishment. Very little excess reactivity is needed in the core, which greatly reduces the magnitude of the reactivity insertion accident (RIA), makes proliferation attempts easy to detect, and significantly reduces the reactivity insertion from water ingress. Every pebble reaches its burnup limit before being discharged from the fuel loop, resulting in very effective fuel utilization. The enrichment is much lower (about 8% U-235) and it is easier to make pebbles than to make compacts and machine blocks; therefore, the fuel costs will be significantly lower. The peak fuel temperatures during normal operation in the pebble-bed reactor are calculated to be somewhat lower than in the block reactor. And finally, the fuel duty on individual fuel elements is milder because of the continuous movement of the fuel; the stress imposed by core hot spots is shared among many thousands of elements each of which only spends of fraction of their residence time at any given location.

Pebble bed reactors of 300 MWt or less had been shown analytically to be passively safe, but the ability of a pebble-bed NGNP of 600 MWt or higher to preserve passive safety had not previously been shown. The pebble-bed NGNP design was developed along two parallel paths. On one path, a reactor module of 300 MWt similar to the South African PBMR [13] was optimized, with the main differences being a higher coolant outlet temperature than the PBMR. On the other path, the feasibility of a single pebble-bed reactor module of 600 MWt was assessed, starting with the overall geometry of the GT-MHR.

III.A. NGNP Pebble-Bed Reactor Neutronics Designs[f]

The principal computational tool used in the pebble-bed reactor physics analyses was PEBBED [14]. PEBBED simultaneously solves the neutron diffusion equation and the equations for the concentrations of specified nuclides (the burnup equations) in a steady-state reactor with a flowing core using cross sections supplied by the MICROX [15] or INEEL’s COMBINE [16] codes. PEBBED provides an exact solution of the nuclide density over the specified mesh. The entry plane burnup is computed for arbitrary, user-defined recirculation patterns, and there are modules for estimating nominal and accident fuel temperatures. PEBBED also contains an automated optimization technique that allows hundreds of cases to be run in a few hours in an intelligent search for

[f] The NGNP Pebble Bed neutronics design and analyses were performed by Hans Gougar, Abderrafi Ougouag, and William Terry at the INEEL.

configurations that best meet a combination of design goals.

The analyses started with the PBMR and GT-MHR core dimensions.[13,5] The PBMR pebble inner reflector was replaced with a solid graphite inner reflector. The pebble designs were optimized as discussed below and then the size of the inner reflector and the fuel annulus was sampled.[17] The height of the core and the discharge burnup were also evaluated. The reactivity versus inner reflector radius of the 300 MWt pebble-bed NGNP calculated by PEBBED using two different cross section sets is plotted in Fig. 14. This is one of the results that show the need for better treatment of the cross sections. However, both calculations show that the performance of the reactor is sensitive to the inner reflector design.

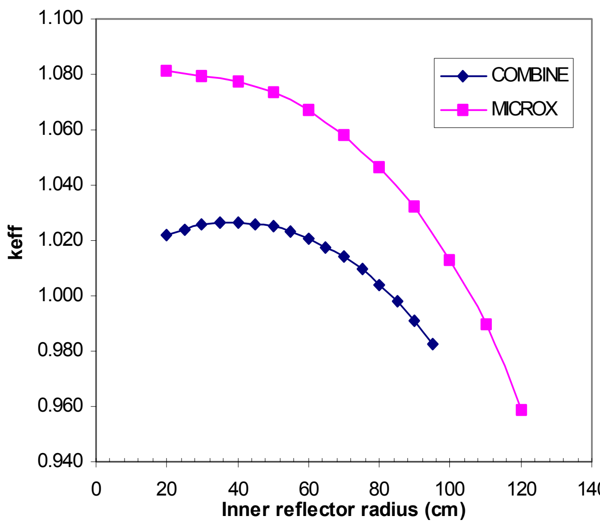

Fig. 14. Reactivity versus inner reflector radius of the 300 MWt pebble-bed NGNP.

Both the 300 MWt and 600 MWt versions of the pebble-bed NGNP utilize a pebble design tailored for the specific core configuration to give better fuel utilization and safer response to reactivity insertion events than in previous pebble-bed reactors. The pebble design feature that is tailored to the specific reactor is the moderator-to-fuel ratio, which is adjusted by properly selecting the radius of the fueled zone within the pebble as shown in Fig. 15. This optimized pebble at least partially mitigates water ingress accidents from the neutronics standpoint; similar optimal moderation is not possible with batch-loaded reactors because the moderator-to-fuel ratio changes continuously in a batch-loaded reactor as the fuel is burned. In contrast, the pebble-bed reactor reaches a steady state distribution of burnup and fuel composition because of its continuous refueling. The improved fuel utilization provided by the optimized pebble leads to lower fuel costs, and it also permits the core with optimized pebbles to be smaller than a core with standard pebbles, so that reactor capital cost is reduced.

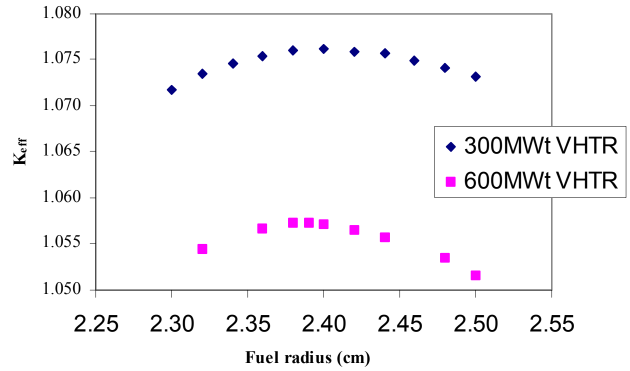

Fig. 15. Reactivity versus fuel radius of 6 cm balls in the 300 and 600 MWt pebble-bed NGNP.

The core designs obtained from the search methods are listed in Table 2. Both pebble bed versions of the NGNP designs use an optimized pebble as discussed above.

Table 2. Summary of features of PBMR and optimal pebble bed versions of the NGNP.

	PBMR	NGNP-300	NGNP-600
Power (MW)	268	300	600
Inlet temperature (°C)	503	600	600
Outlet temperature (°C)	908	1000	100
Coolant flow rate (kg/s)	126	144	288
Active core volume (m^3)	81.8	79.8	119.4
Inner reflector radius (cm)	~ 87	40	150
Core radius (cm)	175	175	250
Outer reflector thickness (cm)	75	76	76
Active core height (m)	8.4	8.75	9.75
Pressure vessel outside diameter (m)	6.02	6.02	7.52
Mean pebble temperature (°C)	806	862	863
Peak pebble temperature (°C)	1041	1027	1028
Peak ΔT across pebble (°C)	59	67	76
Peak pebble power (W)	1379	1301	1791
Mean core power density(W/cm^3)	3.3	3.8	5.03
Peak core power density (W/cm^3)	6.8	7.9	9.3
60 year fast fluence near RPV (n/cm^2)	4.5×10^{19}	2.1×10^{19}	2.6×10^{19}
Reactivity ($) for steam ingress of 0.001/cm^3	0.30	0.42	0.09
Required pumping power (MW)	3	5.9	26
Peak LPCC temperature from PEBBED (°C)	1370	1598	1584

The 300 MWt pebble-bed NGNP differs little from the PBMR, being only slightly scaled up in power and delivering hotter outlet coolant to meet the requirements for hydrogen production. The 600 MWt pebble-bed NGNP required considerable adjustment to meet the requirements for passive safety. The key to achieving this objective was to find a balance between a short thermal conduction path to the heat sink, overall peak power density, and a reasonable vessel size. The core of the 300 MWt design is essentially the same as the PBMR while the 600 MWt design is comparable in diameter and length to the GT-MHR. The high neutron economy of the 300 MWt core allowed the fuel to be burned to a higher degree than the PBMR, the 600 MWt pebble bed, or the prismatic designs. This reduces the fresh pebble injection rate for this design and results in considerable fuel savings.

Table 3 contains the fuel utilization data for the three aforementioned cases. The discharge burnups of the NGNP models were adjusted to yield the same core multiplication factor as that computed for the PBMR. Because the pumping power required in a pebble bed can be significant, the fuel utilization (mass of initial heavy metal per MWd) is based on the net power output (thermal power minus pumping power). The 300 MWt pebble-bed NGNP clearly exhibits superior neutronics performance. As a result, this design uses 14% fewer fresh fuel particles per MWd than the PBMR. Even with the higher required pumping power, the fuel utilization of the 600 MW pebble-bed VHTR is comparable to that of the PBMR.

Table 3. Fuel utilization of PBMR and optimal pebble bed versions of the NGNP.

	PBMR	**NGNP-300**	**NGNP-600**
K_{eff}	1.073	1.073	1.073
Discharge Burnup (MWd/kg_{hm})	80.1	94.3	82.6
Enrichment	8%	8%	8%
HM loading (g)	9.086	7.96	7.96
Number of particles per pebble	15,000	13,271	13,106
Pebble Injection Rate (peb/day)	372	400	923
Number of passes per pebble	10	12	9
Residence Time (days)	875	1082	701
HM Mass Daily Throughput (g/day)	3,3344	3,183	7260
HM Mass Daily Throughput per MWd	12.5	10.6	12.1
Particles/MWd	21,024	18,047	21084

III.B. NGNP Pebble Bed Reactor Safety Studies[g]

Passive safety is the result of adequate heat removal in accident scenarios. The pebble-bed NGNP is shown to possess ample thermal reactivity feedback to shut the reactor down with only a 100 °C temperature increase during any reactivity insertion event, including water ingress accidents. Therefore, the only heat removal required is the post-shutdown decay heat. The peak temperatures during both HPCC and LPCC events were computed in PEBBED using a one-dimensional radial conduction-radiation model. This allows a rapid and but somewhat conservative assessment of passive safety to be generated during the design search.

A more sophisticated model in the MELCOR computer code was also used to compute the peak temperatures during the LPCC event. During an accident when the flow in the core decreases to near zero, the heat generated by the pebbles is removed by conduction and radiation through the pebbles to the graphite reflector. The pebbles in the core are modeled as spherical heat structures, one heat structure per control volume. The heat being transferred from this single structure is then multiplied by the number of pebbles in the control volume to obtain the overall heat transfer from all the pebbles in the volume. A user subroutine is applied to model the conduction heat transfer between heat structures according to the following equation:

$$q = \frac{2\pi hk\left(T_2 - T_1\right)}{\ln\left(\frac{r_2}{r_1}\right)}$$

where k is the effective thermal conductivity of the pebble bed, h is the height of the area normal to the direction of heat flow, and q is the heat transfer rate between structures. The effective thermal conductivity of the pebble bed used in this model is the same as reported in Reference 18.

The thermal analysis of the incrementally uprated 300 MWt design showed that conduction and radiation heat transfer are adequate to remove the decay heat in a loss-of-coolant accident without exceeding prescribed temperature limits (1600 °C). To achieve the same passive safety in the 600 MWt design, the annular core was made somewhat larger (both in diameter and height). The power density increased from 3.8 to 5.0 W/cm^3.

A number of important plant licensing issues were also addressed in the pebble-bed NGNP design project. Previous work analyzed the effects of changes in pebble

[g] The NGNP Pebble Bed safety studies were performed by Richard Moore, Hans Gougar, Abderrafi Ougouag, and William Terry at the INEEL.

packing, as might be caused by earthquakes. It was shown that thermal feedback effects can be expected to overcome the reactivity insertions from such changes in pebble packing. [19] In the present work, an analysis was performed of the potential for hot spots to develop from random collections of high-power pebbles in regions of high thermal neutron flux. It was found that such hot spots would lead to maximum peak temperatures unlikely to cause fuel damage even during a loss-of-coolant accident. The likelihood for such hot spots to form randomly is extremely low (with the worst cases having infinitesimally small probability of occurrence). As noted above, the optimized pebble design mitigates the potential for reactivity insertions caused by water ingress. Previous studies are cited to argue that air ingress will not lead to fuel damage. Nuclear-weapons proliferation issues have also been assessed in previous work; it was shown that the pebble-bed reactor is a very poor choice for proliferation. [20]

IV. SUMMARY

This paper provides analytical evaluations of two possible versions of the Next Generation Nuclear Plant (NGNP), a prismatic fuel type helium gas-cooled reactor and a pebble-bed fuel helium gas reactor. Both designs will meet the three basic requirements that have been set for the NGNP: a coolant outlet temperature of 1000 °C, passive safety, and a total power output consistent with that expected for commercial high-temperature gas-cooled reactors. Two major modifications of the GT-MHR design were needed to obtain a prismatic block design with a 1000 °C outlet temperature: reducing the bypass flow and better controlling the inlet coolant flow distribution to the core. Sensitivity calculations were performed to determine the power that could be obtained for different core heights without exceeding a peak transient fuel temperature of 1600 °C. With a coolant inlet temperature of 490 °C and 10% nominal core bypass flow, it is estimated that the peak power for a 10-block-high core is 686 MWt, for a 12-block-high core is 786 MWt, and for a 14-block core is about 889 MWt. The core neutronics calculations showed that the NGNP will exhibit strongly negative Doppler and isothermal temperature coefficients of reactivity over the burnup cycle. In the event of rapid loss of the helium gas, there is negligible core reactivity change. However, water or steam ingress into the core coolant channels can produce a relatively large reactivity effect.

Two versions of an annular pebble-bed NGNP have been developed, a 300 and a 600 MWt module. From this work we learned how to design passively safe pebble-bed reactors that produce 600 MWt. We also found a way to improve both the fuel utilization and safety by modifying the pebble design (by adjusting the fuel zone radius in the pebble to optimize the fuel-to-moderator ratio). We also learned how to perform design optimization calculations automatically. We can now identify design parameters that optimize selected performance measures by rapidly and automatically performing hundreds of design calculations that evaluate a sequence of design parameter sets. And finally, we learned how to calculate cross sections more accurately for pebble-bed reactors, and identified research needs for the further refinement of the cross section calculations.

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