

## S-PRISM High Burnup Metal-Fuel Core Design

### For Session 3: Future Deployment Programs And Issues

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**Abstract** – *S-PRISM is an advanced Fast Reactor plant design that utilizes compact modular pool-type reactors sized to enable factory fabrication and an affordable prototype test of a single Nuclear Steam Supply System (NSSS) for design certification at minimum cost and risk. Based on the success of the previous DOE sponsored Advanced Liquid Metal Reactor (ALMR) program GE has continued to develop and assess the technical viability and economic potential of an uprated plant called SuperPRISM (S-PRISM) <sup>1-4</sup>.*

*S-PRISM retains all of the key ALMR design features including passive reactor shutdown, passive shutdown heat removal, and passive reactor cavity cooling that were developed under the earlier DOE program. Under JAPC sponsorship, investigation of the potential to further improve plant performance and economics has continued.*

*This paper assesses the performance potential for a metal-fuel core with increased burnup. The reference metal core for S-PRISM is limited to an average burnup of 100 GWd/MTHM to remain within the existing HT9 cladding irradiation database. The high burnup core studied here targets an average burnup of 150 GWd/MTHM and will exceed the existing fast fluence database. Analysis indicates metal-fueled cores of the S-PRISM size tend to be fluence limited by the irradiation database and a shortage of material performance data at high fluence. As a result, a cladding development and verification irradiation test program will be required to permit the burnup increase. For this study, it is assumed the material property correlations remain consistent when extrapolated to higher fluence.*

*Results indicate the high burnup core will have nuclear and thermal-hydraulic performance comparable to the reference metal core. Cladding and duct structural performance is slightly degraded by the lifetime increase, but remains acceptable, with expected high fuel reliability. The annual fuel reload requirement for the high burnup core is reduced significantly, and in isolation would indicate a significant fuel cycle cost reduction. However, in a realistic scenario of LMR commercialization, with new reactors being started at frequent intervals, the need for fuel cycle facilities, and thus fuel cycle infrastructure cost, is dominated by the need for full initial cores. As a result, the fuel cycle cost reduction is expected to be small, on the order of about 10%, or about 0.5 mill/kW-hr.*

## I. INTRODUCTION

The PRISM plant concept has been uprated to improve economics and reduce excessive conservatism. Each reactor (Figure 1) produces 1000 MWt, with two reactors combined into a power block that supplies steam to a single turbine-generator. Since a single reactor design is intended to support commercialization under differing national environments, metal and mixed oxide (MOX) fuel are accommodated within a common geometric envelope, and a breeding ratio from about 0.8 to 1.3 is supported.

The reference radially-heterogeneous metal fuel core is conservatively designed to remain within the existing fuel and cladding irradiation database. It has a refueling interval of 23 months and employs blanket shuffling to enhance breeding and improve internal temperature distributions. The fabricated core height is 1016 mm (40 in.), with 203 mm (8 in.) axial blankets. The resulting breeding ratio is 1.2. Average fuel burnup is 100 GWd/MTHM, with a peak burnup of 150 GWd/MTHM. Peak fast fluence is  $3.6 \times 10^{23}$  n/cm<sup>2</sup> in the fuel and  $3.8 \times 10^{23}$  n/cm<sup>2</sup> in the blankets.

This core design study examines the benefits of increasing fuel burnup beyond that of the reference core. It is a part of the research and development program for Fast Breeder Reactors under sponsorship of the nine Japanese Electric Power Companies, Electric Power Development Co. Ltd., and the Japan Atomic Power Company. Assuming a cladding irradiation program is successful, this higher burnup is supported within the metal fuel irradiation database. This high burnup core targets an average burnup 50% greater than the reference core, or 150 GWd/MTHM.

## II. CORE DESIGN

The high burnup core uses the same configuration (Figure 2) and blanket shuffle pattern (Figure 3) as the reference core. To increase fuel burnup without major impact on the blanket assemblies, the fuel assembly residence time is increased from 3 cycles to 4. The refueling interval is also increased from 23 months to 26. Blanket assemblies remain in a 4-cycle shuffle pattern. The fuel alloy is assumed to include 2 volume percent Nd-143 to represent use of reprocessed fuel with fission products carried back into fresh fuel. Such carry-over may be an integral part of a highly-proliferation resistant and lower cost reprocessing system.

The core height, 1016 mm (40 in.), is set to provide the desired average fuel linear power. The axial blankets then are sized to provide the desired breeding ratio. Figure 4 plots breeding ratio as a function of axial blanket height

for the reference core. With no axial blankets, the breeding ratio is 1.05. With tall axial blankets, it reaches above 1.3. The impact of increasing the number of fuel batches and enrichment to produce the target high burnup is shown in the figure. Replacing 2% of the fuel alloy with Nd-143 causes a further decrease in breeding ratio. To restore the breeding ratio to 1.2, the axial blanket length must be increased to 229 mm (9 in.).

## III. CORE NUCLEAR PERFORMANCE

Table I compares important steady-state nuclear performance parameters for the metal high burnup core with reference metal core.

Most of the core design and performance differences are the result of extending the fuel life and burnup by 50% over the reference core. The addition of 2% Nd-143 to the fuel has a much smaller impact. The small impact of Nd-143 is a consequence of the isotope having very little neutron cross-section in the fast spectrum. Most cross-section is in the thermal neutron spectrum. As a result, in a fast reactor, the addition of Nd-143 to the fuel alloy is only a dilution with a non-fissile, non-fertile material, like the Zr.

Cycle burnup swing in the reference core is about a 1\$ reactivity gain. Increasing the cycle length and fuel burnup causes poorer internal conversion and increases the cycle average burnup swing to a 0.35\$ reactivity loss. Both cycle swings are small and advantageous in core design and plant operation. A small burnup swing reduces the control rod worth requirement. It also reduces rod motion and rod stop system adjustments during the cycle.

Average fuel burnup in the reference core is 103 GWd/MTHM, yielding a peak burnup of 145 GWd/MTHM. This peak burnup is within a conservative limit based on the existing fuel irradiation database. Increasing the average burnup to 150 GWd/MTHM causes the peak burnup to reach 207 GWd/MTHM. The higher peak fuel burnup is only slightly greater than the existing metal fuel database and is expected to be acceptable following a continued fuels irradiation program.

Since enrichment is low in the metal fuel cores, the addition of 2% Nd-143 has only a small effect on enrichment. Enrichment and fuel inventory increase slightly, however the increase is due more to the extended burnup than to the Nd-143 addition. Fissile inventory increases by 84 kg, a 3% increase.

Increasing fuel burnup by 50% causes a significant decrease in fuel supplied per year and per unit of power generated. Fuel kg/yr is reduced from 364 in the reference core to 259 in the high burnup core. Fuel kg/GWDt is also

reduced by 29%, from 1.17 in the reference core to 0.83 in the high burnup core.

The extra axial blanket heights in the high burnup core cause a small decrease in average linear power.

Peak linear powers in the fuel and blanket assemblies are not significantly different from those of the reference core. The increase in burnup in the fuel pins causes the peak power to increase slightly at beginning of life to offset the extra burnup in power at end of life in the high burnup core.

Fuel enrichment, core volume and power distributions are not changed much by the increase in burnup or addition of Nd-143 so the peak neutron flux is also changed little.

Increasing the cycle length and fuel residence time to achieve a high fuel burnup causes a proportional increase in fast fluence in core structures. The existing database for HT9 extends to a fast fluence of  $4 \times 10^{23}$  n/cm<sup>2</sup>. The reference metal core remains within this database, over which no "breakaway" irradiation swelling has been observed. The high burnup core increases fast fluence in the fuel assembly to  $5.3 \times 10^{23}$ . This is well beyond the database and requires an irradiation program to determine whether the material will maintain acceptable strength and swelling characteristics up to this high fluence. Fast fluence in the blanket assemblies reaches  $4.2 \times 10^{23}$  and remains close to the database.

Table II provides details of fuel conversion and breeding. Most of the conversion and breeding come from the low enrichment fuel and the fresh blankets in the internal blanket positions. Axial and radial blankets contribute only about 1/3 of total breeding.

Table III provides additional detail in fuel mass flows. Increasing burnup causes a 1.7% increase in total heavy-metal inventory due to the lengthened axial blankets. Providing the extra reactivity for increased burnup causes the fissile inventory to increase by 3.4%.

While inventories are increased, the higher fuel burnup and longer cycle reduce the fissile fuel loaded and discharged each year by about 30%. The extra axial blanket mass causes the total mass loaded and discharged each year to be reduced by a lesser amount, about 20%.

#### IV. CORE THERMAL, HYDRAULIC AND STRUCTURAL PERFORMANCE

Table IV summarizes important parameters related to core orificing. The cores have the same inlet and outlet temperatures and the same flow rates. The bypass flow fraction, primary coolant not passing through the core assemblies, is 1.5 % for both cores. The low core outlet temperature is selected to provide thermal margins for RVACS operation and to limit thermal creep in core

components. The low core temperature rise, 250 F (139 C), limits high cycle thermal fatigue in near-core structures and also reduces the effects of temperature peaking factors, many of which are proportional to core temperature rise.

While homogeneous layout cores with only radial blankets derive very little power from blanket assemblies (typically about 5% or less), radially heterogeneous cores derive almost 25% of total power from blanket assemblies. Thus the blanket assemblies contribute to core performance in more ways than simply breeding and radial shielding. This effect is an important benefit of the radial heterogeneous configuration which greatly aids in economical performance in a small core.

Because the core-wide power distributions are similar and local peaking factors are similar, the two cores require similar flow orificing. The extra fuel residence cycle and greater blanket breeding in the high burnup core causes an increase in the number of required orifice groups. However, since each assembly seats into an independent orificing/inlet module in the high pressure coolant plenum, the extra orifice groups do not represent a cost difference between the cores.

For the reference core, flow is distributed to assemblies such that the peak cladding midwall temperatures of the peak pins in each assembly are equalized. For fuel assemblies, the peak in life (over 3 cycles) is equalized. For blanket assemblies with shuffling, the peak in each cycle is equalized. The goal of this global equalization (BOL fuel cladding temperatures and cyclic blanket cladding temperatures) is to roughly equalize the cladding damage accrual across all core assemblies.

This simple cladding temperature equalization orificing basis results in acceptable cladding creep rupture damage fraction (CDF) accrual in the reference core, as shown in Table V. The peak cladding CDFs in the most severely loaded fuel and blanket pins are small and of similar magnitude. The blanket pins show less CDF and strain than the fuel pin. Both pins show good margins to limits.

However, applying the simple cladding temperature equalization orificing basis to the high burnup core does not result in acceptable blanket pin performance. The added power produced in the blanket pins causes the peak pin to creep excessively. By observation, most of the excess creep occurs during the last 2 cycle of life, while the blankets are in radial positions. Thus, a small amount of coolant must be diverted from the fuel and internal blanket assemblies to reduce the temperature in the radial blanket positions. The cycle-3 radial blankets are assigned extra flow to reduce their peak cladding midwall temperatures by 18 F (10 C) below the original, globally-equalized temperature, and the cycle-4 radial blankets are

reduced by 45 F (25 C). The fuel and internal blanket assemblies increase in temperature by about 10 F (6 C) as a result of the diverted flow.

With this change to the high burnup core flow orificing distribution, the peak fuel and blanket pins show acceptable performance compared to design limits. However, the high fuel burnup and high blanket temperatures reduce performance margins compared to the reference core.

An ample margin to coolant boiling is shown for peak subchannels in both cores.

Fuel temperatures show margins to centerline melting and surface eutectic liquefaction at full power for both cores. However, the current design basis for the Reactor Protection System (RPS) allows power to reach 113% at scram, including RPS uncertainties. As a result of this design basis event, in which no fuel melting is permitted under the licensing basis chosen for the US ALMR Program, the fuel temperature limits should be reduced to consider the effects of power increased to 113%. The large margins to coolant boiling and fuel centerline melting accommodate this transient power increase. Figure 5 plots peak cladding midwall and fuel centerline temperatures as functions of time in life and elevation in the core for the peak fuel and blanket pins.

The small margins between the fuel surface temperature and the eutectic limit of 1300 F (704 C) indicate this limit does not accommodate the transient scram power increase. For the 1300 F (704 C) eutectic limit to be met at 113% power, the limit should be reduced to 1231 F (666 C) at 100% power.  $[(1300 - T_{inlet})/1.13 + T_{inlet}] = 1231$  Both cores fail to satisfy this reduced eutectic constraint and thus would allow a small amount of eutectic liquefaction during scram transients.

There are two potential approaches to allowing the increased fuel surface temperatures.

- Metal fuel experiments indicate short-term eutectic liquefaction degrades cladding very slowly and is thus acceptable. The licensing basis can be changed to allow slight liquefaction during design basis transients. This approach is believed to be acceptable, but increases licensing complexity and cost.
- The formation of the eutectic phase can be eliminated by the use of barrier cladding, a PRISM-patented design feature. In barrier cladding, a non-iron layer is placed between the fuel and cladding. This prevents interdiffusion of iron and Pu or U and eliminates the formation of the eutectic layer. With this design feature, the fuel surface temperature limit, under the current no-melting licensing basis is increased to the

fuel alloy melting temperature, i.e. the same temperature limit as applied for fuel centerline melting. The fuel surface temperature limit is then always less limiting than the fuel centerline temperature limit.

S-PRISM uses the second option, barrier cladding, and thus all pin limits are met with margin. This design feature requires irradiation experiments to verify existing laboratory results and to build the required licensing database.

Duct radial growth is also shown in Table V. Both cores exceed the very conservative "assembly unit cell" limit. The reference core meets the empirical limit derived from reactor experience for refueling. Thus the reference core is expected to show acceptable refueling forces. However, the high burnup core shows the most severely stressed blanket ducts to slightly exceed the empirical limit. Since the limit is "soft" (not a fixed numerical limit), it is expected that the assemblies will most likely also show acceptable refueling forces. In addition, the radial dilation can be easily reduced by increasing the duct wall thickness and/or by reducing the assembly pressure drop. Both changes will have a negligible impact on core neutronic performance.

The cladding and duct structural analyses assume the property correlations applicable up to the limit of the existing irradiation database extrapolate to the higher fluence predicted for this core. A test program will be required to validate this assumption.

## V. LMR COMMERCIALIZATION

The economics of commercialization of this LMR concept are studied under conditions that do not directly correspond to those assumed in the Feasibility Study ongoing in Japan. In both this study and in the Feasibility Study, it is expected that, after development of supporting technologies to allow increased fuel burnup, overall power cost will be reduced with increasing burnup. However, due to the assumed reactor commercialization scenario, in which a) one S-PRISM power block is built per year and b) startup fuel is provided by recycle of spent LWR fuel, the cost savings for increased burnup, i.e. reduced demand for reload fuel, is masked by the large LWR recycle mass needed for new reactors. The basic cause is the low enrichment of spent LWR fuel, about 1/10<sup>th</sup> that of spent LMR fuel – 10 times as much reprocessing capacity is required to get a kg of startup fissile fuel from LWR spent fuel as is needed to get a kg of reload fuel from spent LMR fuel. A related masking cause is the use of a radial heterogeneous core in S-PRISM, compared to homogeneous cores in other plants. A heterogeneous core concentrates fissile fuel into fewer assemblies and

produces higher burnups for the same power output. The net result of these differences is a reduced benefit from increased burnup.

Figures 6 and 7 summarize the characteristics of an assumed LMR commercialization scenario. S-PRISM power blocks (2 reactors) are assumed started at 1 per year for 50 years, starting in 2020. The scenario period covers years 2000 through 2100, inclusive. The period before the first LMR startup in 2020 is required to permit the LWR-SFRF startup and operation and thus supply the fuel required for the first reactor by reprocessing of spent LWR fuel. Pyro-process technology is assumed for reprocessing.

Figure 6 plots the number of SPRISM power blocks and fuel cycle facilities operating each year. The LMR power blocks increase linearly from 2020 through 2069, from 0 to 50 power blocks. A total of 100 cores are loaded. These reactors begin decommissioning in 2080, after a 60 year life. To maintain constant electric output, a second set of power block startups begin in 2080 and continue through the end of the study period, 2100.

Overall fuel cycle inventories are plotted in Figure 7. Note that the plot is semi-log so that the large inventories of spent LWR fuel reprocessed and the resulting uranium inventory do not render the much smaller fissile and waste inventories unreadable.

The first LWR-SFRF is completed in 2015 and reaches full production in 2017. This facility builds an inventory of fuel to start and reload the initial power blocks, however, the demand for fuel rapidly exceeds the supply and a second LWR-SFRF is required to be operating by 2021. Both facilities operate for 20 years and 9 months and are then shutdown.

The first LMR-SFRF is completed in 2028 and reaches full production in 2030. The second LMR-SFRF is completed in 2040 and reaches full production mid-year in 2042. Subsequent LMR-SFRF startups are spaced about 12 years apart. The startup of each facility is delayed until sufficient spent fuel inventory and production is available to keep it fully loaded and thus operating at 100% of capacity.

The combination of LWR-SFRFs and LMR-SFRFs provide fuel together until 2046, assuming LWR-sourced fuel is the preferred source. The transition interval is controlled by the choice of a preferred fuel source. If LMR-sourced fuel is the preferred source, the LWR-sourced fuel is slowly used as makeup and lasts until the LMR fuel system reaches fissile breakeven in 2065. The choice of using LWR-sourced fuel in preference to LMR-sourced fuel is made based on an assumed desire to reduce the inventory of LWR-sourced fissile material as quickly as possible for non-proliferation concerns. This choice a) ties up LWR fissiles in new LMR cores and makes them

unavailable for other uses at the earliest date, b) reduces the peak total fissile inventory in the LMR fuel system and c) contains the major fissile inventory at the LMR-SFRFs in a more diversion-resistant form.

The fissile inventory from LWR recycle initially increases to about 6.7 MT in 2021 during the early build up of excess fuel. This inventory is rapidly depleted and a second LWR-SFRF comes on line in time to prevent a fuel shortage. The two facilities then combine to build an inventory of fissile material. The fissile inventory peaks in 2034 at 25.4 MT. The LWR-SFRFs are shut down in 2038 and 2045 and the fissile inventory is used to make up the LMR-sourced shortfall until the LMR fuel cycle reaches fissile break-even in 2065.

The LMR-sourced fissile inventory grows initially as it is not fully used for resupply each year. The maximum processed fissile inventory at the SFRF reaches 88.9 MT in 2045, about the time the LWR-SFRFs are shutdown and the last of the LWR-sourced fuel is exhausted. The fissile inventory then drops to a low of 2.6 MT in 2066, at the time fissile breakeven is reached in the LMR fuel system. After that time, a recovery in fissile inventory is made possible and overall inventories can be controlled through changes in the core breeding ratio.

When compared to the matching reference core commercialization scenario, the fuel reload metal required for each core each year is reduced for the high burnup core compared to the reference core. However, the fuel cycle facility throughput requirement is dominated for many years by the production of new cores, not by reload of existing cores. Thus, the total fuel masses requiring reprocessing each year are about the same with the high burnup core as with the reference lower burnup core. While the fuel mass flow drops by the increase in burnup, the blanket mass flow is unchanged. The fuel column mass represents less than 1/3 of the total reload mass and thus total mass flow in the reload is not greatly different between the two cores. Since recycle facility throughput is defined and limited in terms of total heavy metal, the large increase in fuel burnup results in only a small change in facility throughput requirement. The net result is the LMR recycle facilities can be delayed only a few years in startup, but the same number of facilities are required. In addition, the initial high burnup cores require more heavy metal than the reference cores and thus increase the demands on the facilities reprocessing LWR spent fuel used to begin the LMR fuel cycle. Based on the required timing of recycle facility construction and in the number of fuel assemblies to be fabricated, it is estimated the high burnup core will show a small fuel cycle cost advantage compared to the reference core. The reference core fuel cycle cost is estimated as 5 mills/kW-hr assuming utility ownership of the facilities. The high burnup core is

estimated to provide about a 10% cost reduction from these values. The resulting busbar power cost is 29 mills/kW-hr.

## VI. CONCLUSIONS

Overall, the high burnup core shows nuclear performance comparable to the reference core.

The ability to optimize internal conversion, breeding and coolant flow distribution with blanket shuffling are evident in nuclear and thermal-hydraulic performance. Shuffling permits a large gain in nuclear performance for radial heterogeneous cores without significant penalties in temperatures or fuel reliability.

The total heavy metal mass and fissile mass are both increased in the high burnup core to support the increased burnup of fissile fuel and the extra axial blanket required to offset the breeding loss from the Nd-143 addition. The reduction in fuel reload requirement is expected to permit about a 10% reduction in fuel cycle cost compared to the reference core.

Core thermal-hydraulic and structural performances are comparable between the two cores. Both cores require HT9M as the structural material to limit thermal creep in the cladding. Both cores require barrier cladding to eliminate the eutectic liquefaction limit at the bounding design basis condition of 113% power at scram.

The cores fit the S-PRISM geometric envelope and show acceptable steady-state nuclear, thermal-hydraulic, structural and economic performance.

## REFERENCES

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**Table I Core Nuclear Performance Summary Comparison**

	<b>High Burnup Heterogeneous Metal Core ( With 2% Nd-143 )</b>	<b>Reference Heterogeneous Metal Core</b>
Average Breeding Ratio	1.20	1.22
CSDT (years)	45.1	41.88
Burnup Reactivity Loss (% $\Delta k/k$ )	0.12	-0.31
Average Fuel Burnup (MWd/kg)	151.0	102.71
Peak Burnup (MWd/kg)		
Fuel	207.1	145.14
Inner Blanket	33.5	29.3
Radial Blanket	68.5	59.51
Maximum Feed Enrichment (wt.%)		
Fissile Pu in U-TRU	18.0	16.39
Total Pu in U-TRU	23.3	21.29
Total TRU in U-TRU-Zr	21.6	19.68
Fissile Pu Inventory at BOC		
kg	2542.9	2458.8
kg/MWt	2.54	2.46
Supplied Fissile Pu		
kg/year	259.2	363.97
kg/GWDt	0.83	1.17
Fissile Pu Gain (kg/year)	61.06	69.91
TRU Consumption Rate (kg/year)	-73.75	-84.63
Average Linear Power (kW/m)		
BOC	18.88	19.25
EOC	16.85	17.39
Peak Linear Power (kW/m)		
Fuel	30.60	29.77
Internal Blanket	39.26	38.3
Radial Blanket	30.53	29.8
Spatial Power Peaking Factor		
BOC	1.35	1.40
EOC	1.41	1.43
Peak Neutron Flux ( $10^{15}$ n/cm <sup>2</sup> -s)		
Total Flux	3.46	3.49
Fast Flux	2.33	2.37
Peak Fast Fluence ( $10^{23}$ n/cm <sup>2</sup> )		
Fuel	5.31	3.61
Inner Blanket	2.51	2.26
Radial Blanket	4.22	3.79

**Table II Core Breeding Performance Comparison**

	High Burnup Heterogeneous Metal Core ( With 2% Nd-143 )		Reference Heterogeneous Metal Core	
	BOC	EOC	BOC	EOC
Conversion Ratio				
Fuel	0.572	0.621	0.603	0.645
Axial Blanket	5.848	3.716	8.573	4.717
Inner Blanket	7.904	2.953	8.485	3.184
Radial Blanket	2.079	1.798	2.228	1.918
Breeding Ratio				
Fuel	0.468	0.445	0.504	0.479
Axial Blanket (Note 1)	0.113	0.113	0.106	0.105
Inner Blanket	0.405	0.410	0.409	0.410
Radial Blanket	0.232	0.207	0.231	0.206
Total	1.217	1.174	1.249	1.199

Note 1 Includes axial blankets on fuel assemblies only.

Axial blankets in blanket assemblies are included with the blanket assemblies.

**Table III Core Mass Flow Comparison**

	High Burnup Heterogeneous Metal Core ( With 2% Nd-143 )		Reference Heterogeneous Metal Core	
Supplied Fuel (kg/yr)				
Fissile Pu	259.20		363.97	
Total TRU	345.88		485.69	
U + TRU	4450.43		5660.45	
Discharged Fuel (kg/yr)				
Fissile Pu	320.26		433.88	
Total TRU	419.63		570.31	
U + TRU	4135.38		5345.37	
Fuel Consumption (kg/year)				
Fissile Pu	-61.06		-69.91	
Total TRU	-73.75		-84.63	
U + TRU	315.05		315.08	
Fuel Inventory at BOC (kg)				
Fissile Pu	2542.90		2458.80	
Total TRU	3338.80		3195.90	
U + TRU	36882.70		36248.60	



**Table IV Core Orificing Comparison**

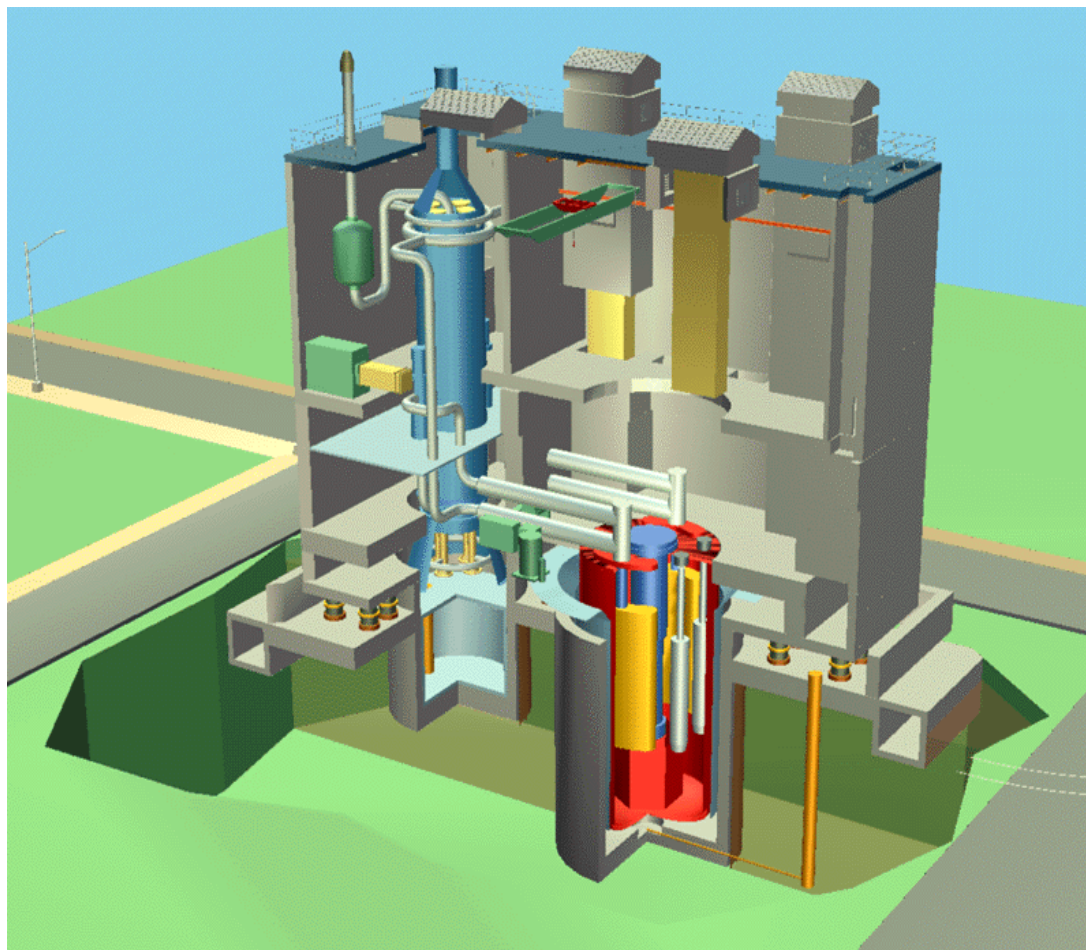
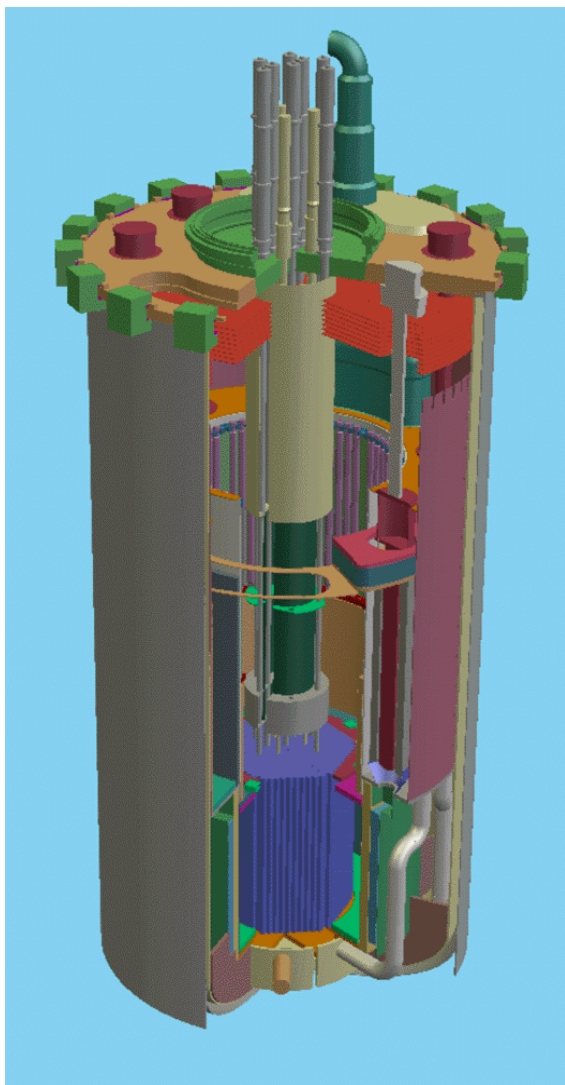
	<b>High Burnup Heterogeneous Metal Core ( With 2% Nd-143 )</b>		<b>Reference Heterogeneous Metal Core</b>	
Primary Loop Flow Rate (lbm/hr / kg/sec)	45014259 / 5671.8		45014259 / 5671.8	
Bypass Flow Fraction (%)	1.5		1.5	
Core Inlet Temperature (F / C)	700 / 371		700 / 371	
Core Outlet Temperature (F / C)	950 / 510		950 / 510	
Core Flow Orificing Basis	Equalized peak pin cladding midwall temperatures in each assembly across the core, with reductions for radial blankets: Peak in lifetime for fuel Peak in each cycle for cycle 1 & 2 (internal) blanket Peak in each cycle, less 18 F (10 C) for cycle 3 blanket Peak in each cycle, less 45 F (25 C) for cycle 4 blanket		Equalized peak pin cladding midwall temperatures in each assembly across the core: Peak in lifetime for fuel Peak in each cycle for blanket	
	<b>Fuel</b>	<b>Blanket</b>	<b>Fuel</b>	<b>Blanket</b>
Assembly Group Power Fraction At MOC *	0.749	0.243	0.760	0.231
Assembly Group Flow Fraction At MOC *	0.703	0.269	0.678	0.291
Orifice Groups	6	9	5	7
Peak Assembly Flow Rate (lbm/hr / kg/sec)	241596 / 30.44	163984 / 20.66	233585 / 29.43	169569 / 21.37
Peak Assembly Pressure Drop (psi / MPa)	52.4 / 0.362	58.9 / 0.406	49.0 / 0.338	63.0 / 0.434

\* MOC = Middle Of Cycle

**Table V Peak Assembly T/H And Structural Comparison**

	High Burnup Heterogeneous Metal Core ( With 2% Nd-143 )		Reference Heterogeneous Metal Core		Limit
	Fuel	Blanket	Fuel	Blanket	
Assembly Location (ring – row)	6 – 8	Path 2	6 – 8	Path 1	
Cladding Material	HT9M	HT9M	HT9M	HT9M	
In-Core Residence Time (EFPD)	2690	2690	1785	2380	
Peak Fast Fluence (n/cm <sup>2</sup> )	5.27 E+23	4.22 E+23	3.59 E+23	3.79 E+23	
EOL Fission Gas Release (%)	100	100	100	100	
End Of Life Plenum Pressure At Full Power (psi / MPa)	1388 / 9.57	524 / 3.61	958 / 6.61	429 / 2.96	
Peak Creep Rupture Damage Fraction (Fractional)	1.56 E-4	3.08 E-3	5.76 E-5	1.45 E-5	0.20
Peak Total Creep (%)	0.882	1.260	0.451	0.228	
Peak Thermal Creep (%)	0.124	0.739	0.076	0.050	1.0
Peak Irradiation Creep (%)	0.783	0.824	0.378	0.208	
Peak Irradiation Swelling (%)	0.050	0.049	0.049	0.049	
Peak Total Diametrical Growth (%)	0.928	1.302	0.491	0.271	2.0
Peak Duct Radial Dilation (in / mm)	0.096 / 2.45	0.138 / 3.51	0.086 / 2.19	0.094 / 2.40	0.085 / 2.16 (Conservative Limit) 0.135 / 3.46 (Approximate Test-based Limit)
Peak Fuel Centerline Temperature (F / C) (See Note 1 below)	1517 / 825	1375 / 746	1492 / 811	1341 / 727	Fuel: 1630 F / 888 C Blanket: 1745 F / 952 C
Peak Fuel Surface Temperature (F / C) (See Note 1 below)	1254 / 679	1233 / 667	1242 / 672	1195 / 646	Standard cladding: 1231 F / 666 C Barrier cladding: Not Applicable
Peak Cladding Midwall Temperature (F / C)	1228 / 664	1241 / 672	1216 / 658	1224 / 662	
Peak Subchannel Coolant Temperature (F / C) (See Note 1 below)	1194 / 646	1212 / 656	1184 / 640	1200 / 649	1806 F / 986 C (Boiling At Full Flow)

Note 1. Limits after reductions to compare 100% performance values with limits applied at 113% power.



**Figure 1 S-PRISM Reactor Module And NSSS**

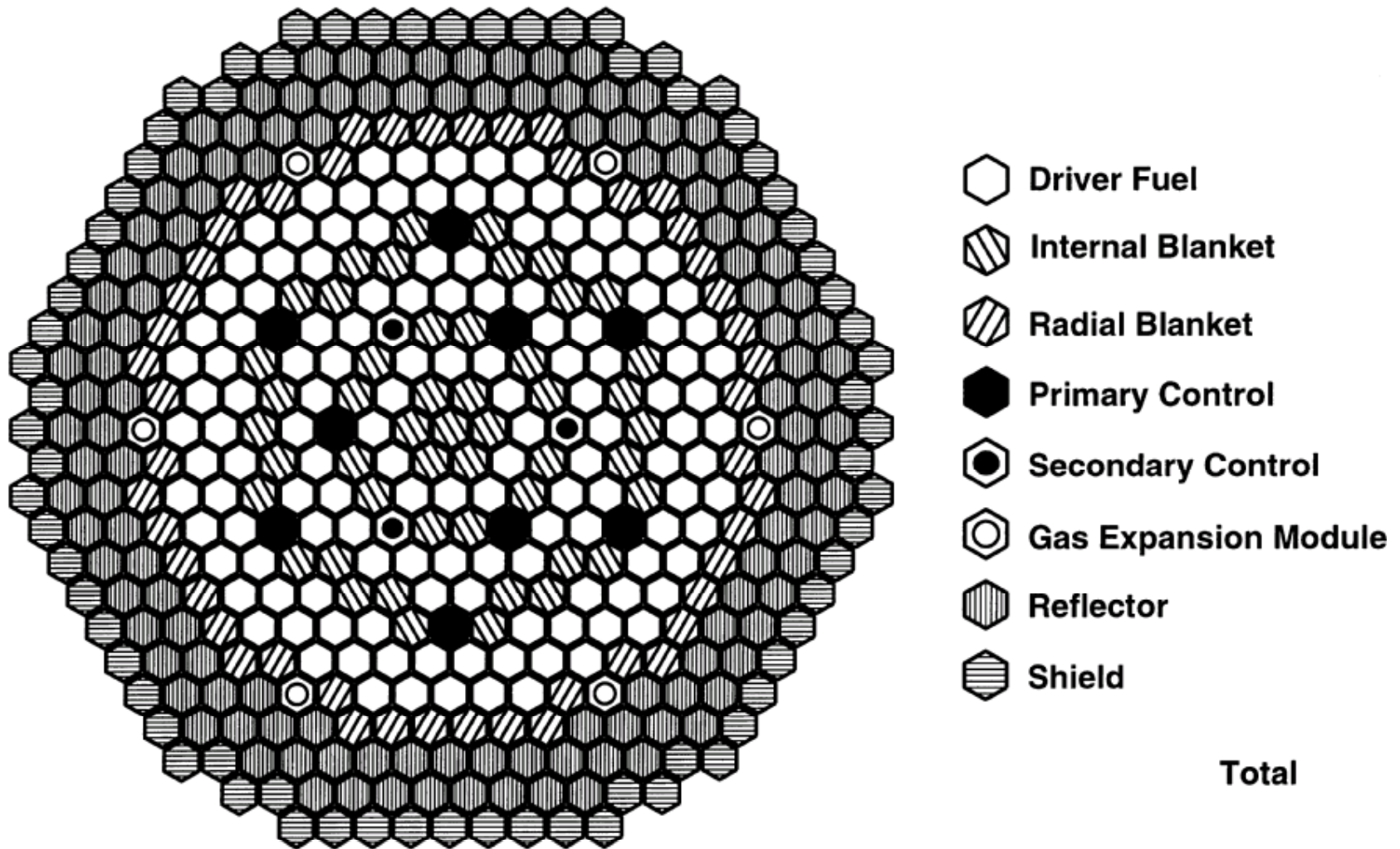


Figure 2 Metal Heterogeneous Core Configuration

**Blanket Shuffle Pattern**

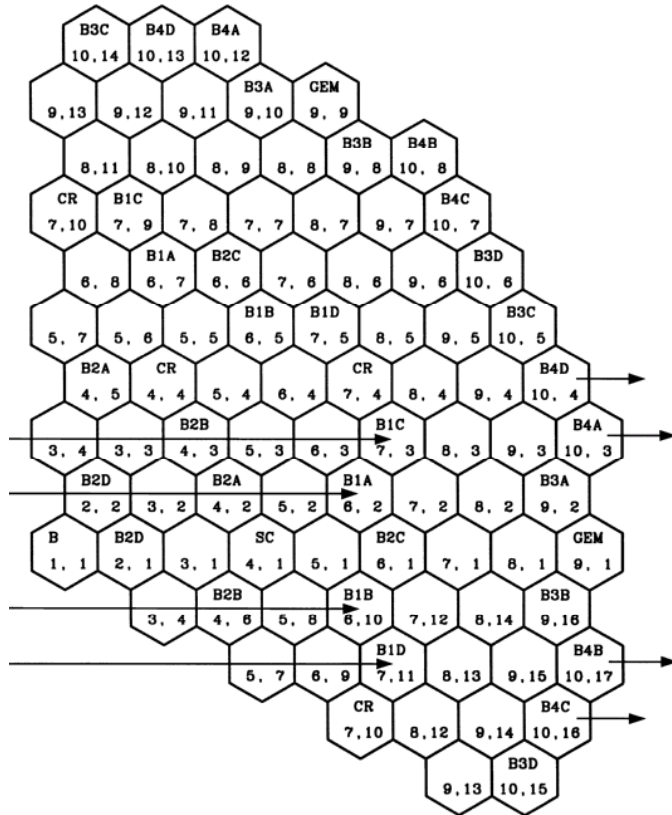
B1A → B2A → B3A → B4A → out

B1B → B2B → B3B → B4B → out

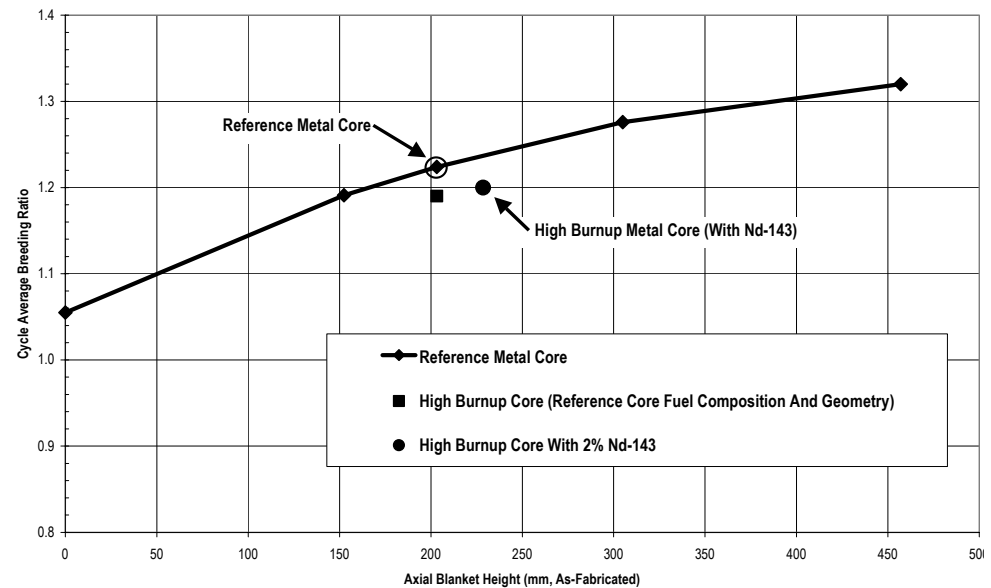
B1C → B2C → B3C → B4C → out

B1D → B2D → B3D → B4D → out

(Note: 60-degree rotational symmetry applies)



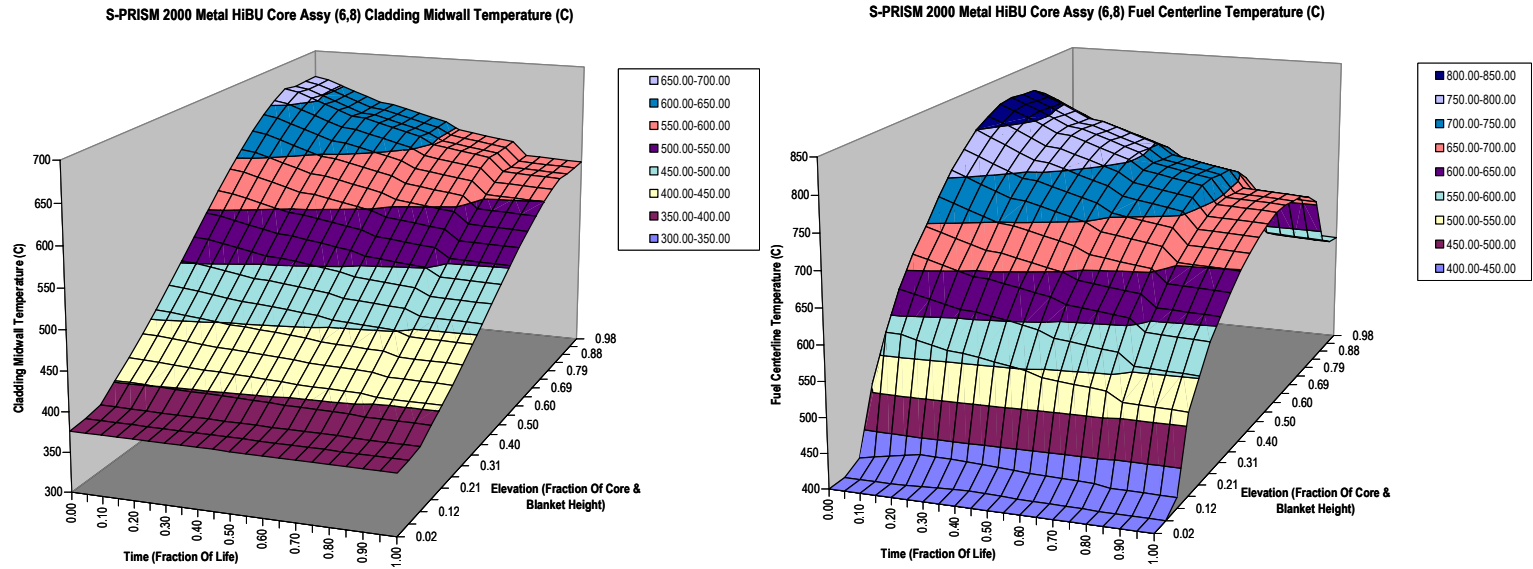
**Figure 3 Metal Heterogeneous Core  
Blanket Shuffle Pattern**



**Figure 4 Metal Heterogeneous Core  
Breeding Performance**



PEAK FUEL PIN



PEAK BLANKET PIN

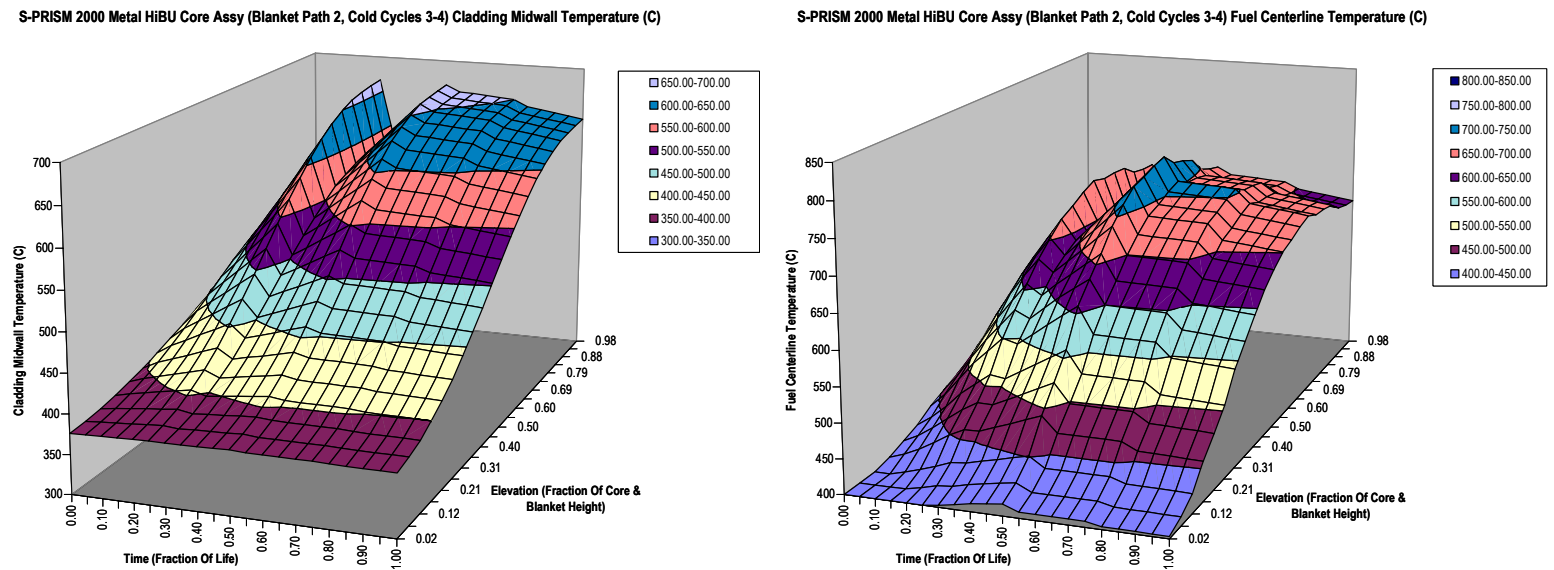
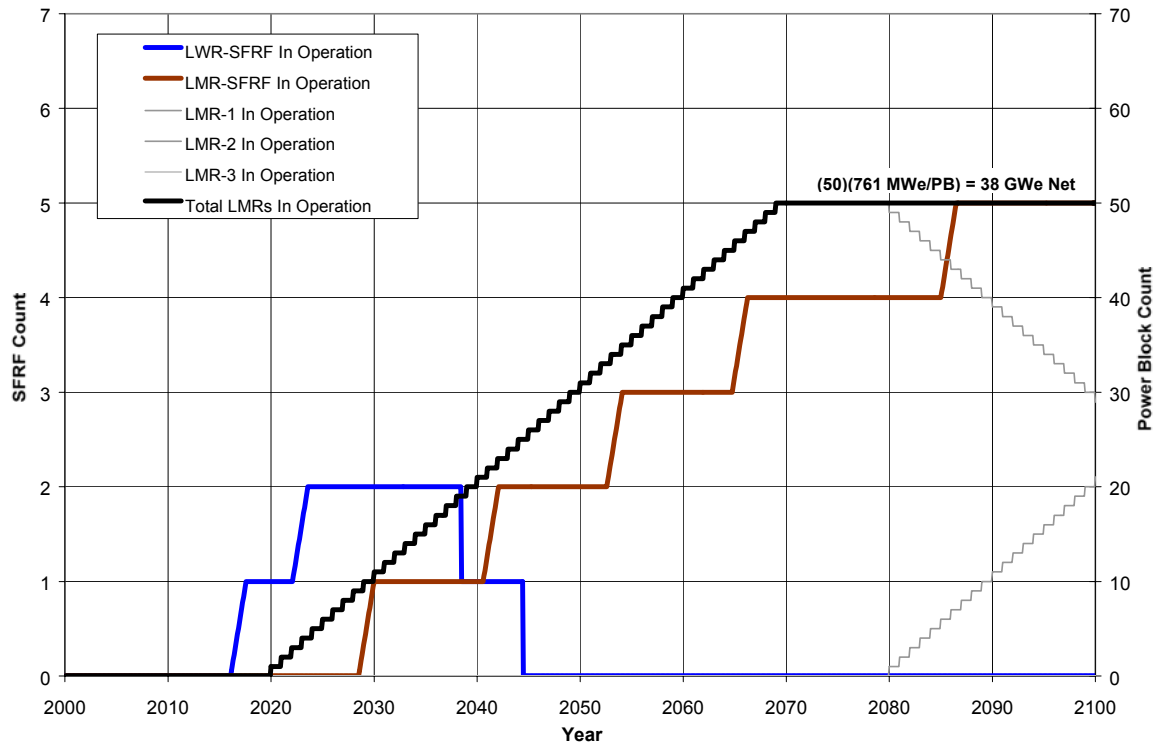
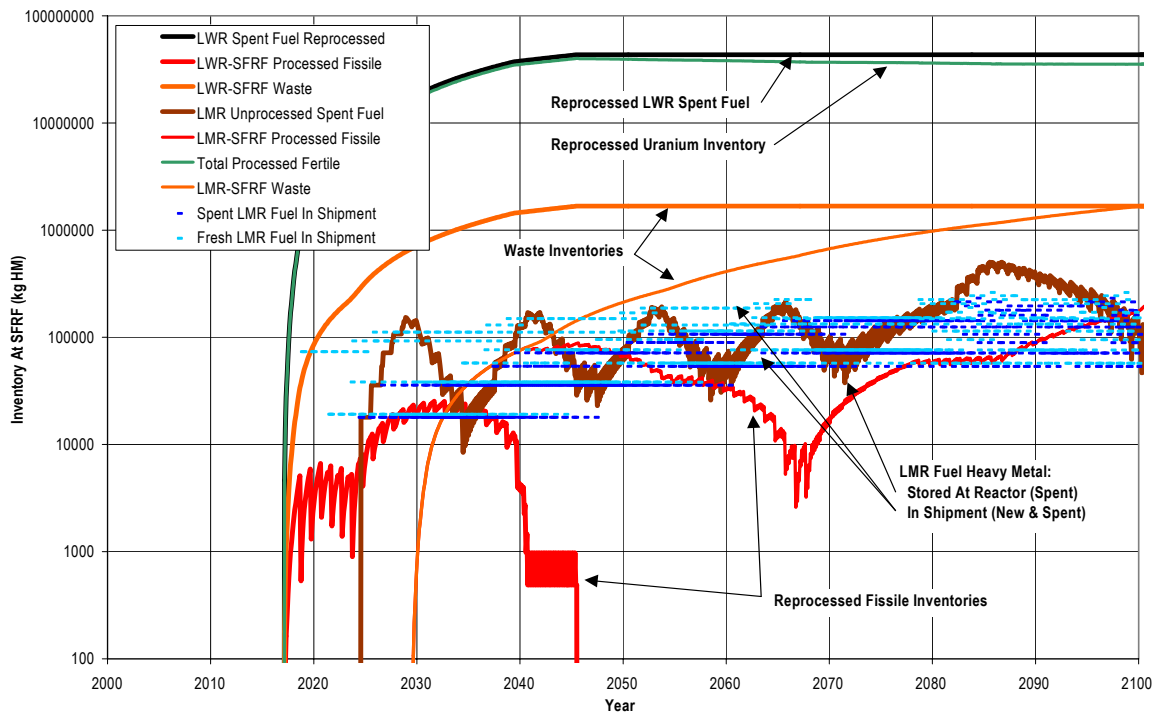


Figure 5 Metal High Burnup Core Peak Pin Performance



**Figure 6 Metal High Burnup Core With Nd-143 - Fuel Cycle Facilities In Operation**



**Figure 7 Metal High Burnup Core With Nd-143 - Fuel Cycle Inventories**