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Summary of Near-Term Options for Russian Plutonium Production Reactors

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Executive Summary

The Russian Federation desires to phase out the production of weapons-grade plutonium. To this end, ten graphite-moderated, water-cooled reactors have been shut down during the last several years. However, complete cessation of plutonium production is impeded because the three operating Russian reactors supply district heat and electricity to the Tomsk and Krasnoyarsk regions in addition to producing weapon-grade plutonium.

In August 1992 the Russian Federation Ministry of Atomic Energy (MINATOM) and the Russian Nuclear Regulatory Agency (GAN) requested U.S. assistance for achieving a cessation of weapons-grade plutonium production, placing the plutonium production reactors under safeguards, and conducting a program to evaluate and assist in the upgrade of plant safety.

As a result of that and subsequent communications, Secretary O'Leary and Minister Mikhailov have signed a protocol that expressed their desire to shut down the three remaining plutonium production reactors as soon as possible by replacing them with alternate energy sources. In the meantime, both MINATOM and the Department of Energy (DOE) are concerned about the safety of the plants as well as the difficulty in ceasing the production of plutonium as long as the plants continue to operate. A military subsidy has been provided for operation of the production reactor complex. Revenues received for providing district heat and electricity are insufficient to cover costs for the current natural uranium metal fuel cycle. A more economical fuel cycle is needed for civilian operations.

The Russians are considering the use of a cermet fuel, capable of high burnup, for operating their production reactors in a power-only mode. The cermet fuel consists of uranium oxide dispersed in aluminum. Such fuel is presently being fabricated in Russia and used in about 100 channels of each production reactor for equalizing the distribution of power. If the United States shared its experience with adding erbium as a burnable neutron absorber to fuel, the undesirable positive-coolant-void reactivity effect (inherent in these reactors) could be eliminated. As a result, the operational stability and other safety characteristics of the Russian production reactors would be significantly improved. Furthermore, this improved cermet fuel could be fabricated and loaded into the production reactors during refueling in the next two years. This improved cermet fuel would contain <20% enriched uranium. Its use would eliminate discharge of weapons-grade plutonium in spent fuel; reduce the quantity of spent fuel discharged by at least an order of magnitude; provide attractive options for long-term, spent fuel storage; and make reprocessing unattractive. In addition, the quantity of plutonium discharged annually in such cermet fuel would be about 20 times less than in the current natural uranium fuel. Finally, the spent cermet fuel would contain only plutonium that is highly burned (>30% ²⁴⁰Pu), and be more proliferation resistant than commercial spent fuel.

U.S. assistance to Russia would likely be required for any near-term change in the fuel cycle for the Russian production reactors. Implementation of a proliferation-resistant fuel type containing erbium would require U.S. technical support for changes in fuel fabrication, materials behavior,

reactivity and burnup analytical methods, and safety analysis. This expertise exists at Pacific Northwest Laboratory.^(a) This document provides the technical basis for further U.S. discussions with Russia concerning fuel cycle alternatives for the production reactors.

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1.0 Introduction

The Russian Federation desires to stop producing weapons-grade plutonium. During the last several years, ten graphite-moderated, water-cooled, production reactors have been shut down. However, complete cessation of weapons-grade plutonium production is impeded by the fact that the last three operating Russian plutonium production reactors supply electrical energy and district heat as well as produce plutonium. These reactors are major suppliers of heat in the Tomsk and Krasnoyarsk regions of Siberia.

1.1 Bilateral Cooperation

In August 1992 the Russian Federation's Ministry of Atomic Energy (MINATOM) and Nuclear Regulatory Agency (GAN) requested U.S. assistance for achieving a cessation of weapons-grade plutonium production, placing the plutonium production reactors under safeguards, and conducting a program to evaluate and assist in safety upgrades of the plant.

As a result of that and subsequent communications, Department of Energy (DOE) Secretary O'Leary and MINATOM Minister Mikhailov signed a protocol in March 1994 that expressed their desire to shut down the three remaining plutonium production reactors as soon as possible by replacing them with alternate energy sources. In the meantime, these reactors will continue to operate. Both the Russian Federation and the United States are concerned about the safety of the plants as well as the difficulty in stopping plutonium production.

1.2 Reactor Design and Operations

The three operating Russian plutonium production reactors are located in the Tomsk (2) and Krasnoyarsk (1) regions of Siberia (Figure 1). These reactors were designed and constructed in the 1960s. The design and operations were conducted in a closed environment necessitated by the security requirements of the nuclear weapons program. No U.S. citizen has ever visited these plants. Information available to the U.S. indicates that these reactors represent a very unique design; the only comparable plants are those located at the U.S. Hanford Site.

Figure 2 shows a simple block diagram of the reactors developed from the available design information. The reactor fuel is located in the process tubes, the active core region being in the center of the graphite region. Figures 3 and 4 are simple schematics of the reactor heat and electricity delivery systems for the Tomsk and Krasnoyarsk stations, respectively. These figures illustrate how the reactor coolant loop interfaces with the distribution of district heat to the local regions.

Figure 1. Map of Russia

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Figure 2. Russian Production Reactor Diagram

Figure 3. Tomsk Production Reactor Schematic

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Figure 4. Krasnoyarsk-26 Production Reactor Schematic

As a result of the closed environment in which the Russian production plants exist, there has been essentially no opportunity for the designers or operators of these plants to benefit from the considerable nuclear infrastructure that exists today for other plants. The development and use of analytic tools, the performance and analysis of experiments, and the construction and operation of the reactors was largely done in isolation.

A similar condition existed in the U.S. at the Hanford site for many years. Scientists and engineers at the Hanford Site designed, constructed and operated eight single-pass graphite-moderated, water cooled reactors for plutonium production, in a highly isolated technical environment. The ninth plant, N-Reactor, was a dual-purpose plant that produced weapons-grade plutonium and steam for generating electricity. The N-Reactor was also different in that some external safety reviews were conducted during its design, and extensive design and operational safety reviews and upgrades were performed during the operating life of the plant. Considerable effort was expended in order to apply modern safety criteria to a one-of-a-kind plant. When the U.S. requirements for weapons-grade plutonium diminished, the plant was operated in the power-only mode (with higher fuel burnup) for a period of about 10 years.

1.3 Near Term Prospects

MINATOM wants to produce more electrical energy and district heat from its operating reactors at Tomsk and Krasnoyarsk, (Mikerev 1994). Its intention is to change the fuel cycle to one that is economical without military subsidies. This method is the fastest and most reliable approach to stop production of weapons-grade plutonium. In addition, such a fuel cycle change provides the opportunity to enhance inherent safety characteristics of these reactors. Russian reactor design experts recognize that the addition of erbium to fuel should enhance reactor operational safety, but they do not have the operational experience or validated analytical methods to make reliable performance predictions for this material. Responsibility for review and assessment of safety for the production reactors is assigned to GAN, which will determine areas where improvements are required (Dmitriev 1994). In any event, operation of three production reactors by MINATOM for electricity and district heat is expected to continue, at least, until the turn of the century.

The Pacific Northwest Laboratory (PNL) has been working cooperatively with experts at the Kurchatov Institute and the Nuclear Safety Institute in Moscow, Russia, to assess potential fuel cycle options for Russian production reactors. This report provides analytical comparisons of performance and safety characteristics of proliferation-resistant fuel cycle alternatives. Each of the fuel cycle options evaluated incorporates erbium as a burnable neutron absorber in the fuel. Russian production reactor performance and safety characteristics, with each of these fuel types, were predicted by PNL using state-of-the art nuclear fuel burnup models that have been validated for analysis of fuel containing erbium in U.S. commercial nuclear power plants, and for analysis of coolant voiding in N-Reactor fuel at Hanford (see Appendix B).

1.4 Report Outline

Section 2.0 of this report provides the near-term options for enhancing safety of the three operating reactors in the change from military to civilian use. The preferred fuel cycle option for operating the Russian Federation's plutonium production reactors is described in Section 3.0, and the study's conclusions are provided in Section 4.0. Section 5.0 lists the references cited in the report.

2.0 Near-Term Options

2.1 Current Situation

Russian technical experts provided sufficient information to PNL staff regarding the production reactors' configuration, fuel and control rod composition, reactor core loading, and operating parameters to initiate investigation of alternative fuel cycle comparisons. The Russian production reactors are currently fueled primarily with natural uranium metal fuel, as shown in Figure 5. Fuel channels and control channels are vertically oriented in graphite blocks on a 200 mm square pitch in a configuration illustrated in Figure 6. The reactor core contains about 300 MT of natural uranium, and half of it is discharged after every 30 effective-full-power-days (EFPD) of operation. Approximately 1200 MT of spent fuel is discharged annually from each reactor. In the central region of the core, a control rod is located in each 4 x 4 array of channels, as shown in Figure 7. This control rod system has a total reactivity worth of about 8%, at operating conditions. Reactivity of the natural uranium fuel changes less than 3% during irradiation, as shown in Figure 8. The principal safety issue regarding this fuel is that the reactivity increase with fresh fuel due to voiding the water coolant in fuel channels is about 0.5% (~70% of the delayed neutron fraction), as shown in Figure 9. Positive reactivity changes due to water density reductions to 80% of normal, by increases in coolant temperature, are also shown in Figure 9. The positive reactivity void defect increases during irradiation as indicated by the coolant temperature coefficient of reactivity, shown in Figure 10. Reactivity transients due to coolant voiding must be controlled by 25 fast-acting rods (worth about 1% of reactivity), which can be inserted in 2 seconds. Even though the natural uranium metal fuel has sufficient excess reactivity to achieve a 360 EFPD burnup rate (comparable to N-Reactor fuel), it would not be technically feasible to operate Russian production reactors with such large positive reactivity effects due to coolant voiding at those fuel exposures. About half a metric ton of weapons-grade plutonium is being discharged annually from each operating reactor. This fuel is being reprocessed within a year or two after discharge because of the concern about potential degradation of the aluminum cladding on the natural uranium metal fuel and consequences of exposed uranium dissolution during wet storage. Dry storage of such a large quantity of spent fuel would be too costly. As a result, the separation, recovery, and stockpiling of an additional 1.5 MT of weapons-grade plutonium continues each year with the current natural uranium fuel cycle.

2.2 Fuel Cycle Alternatives

A range of potential fuel cycle alternatives were evaluated against the following criteria:

- convert plutonium production reactors from military to civilian use
- meet needs for district heat & electricity
- eliminate discharge of weapons-grade plutonium in spent fuel

Figure 5. Russian Production Reactor Fuel

Legend:

- = Natural U Metal Fuel
- S = Spike Fuel in Cential Region
- R = Spike Fuel in Periphary
- C = Control Rod

Figure 6. Representative Core Map

25 Fast Acting Rods: 2 second
 (1.5 β Shutdown): Insertion Time
 100 Normal Control Rods: 6 second
 Insertion Time
 (>1% Cold Shutdown Margin)

Coolant Void Reactivity in Fuel $\sim 0.5\% \Delta K/K$ (0.7 β)

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Figure 7. Russian Production Reactor Control and Shutdown Rods

Figure 8. Natural U Reactor K-infinity Versus Burnup

Figure 9. Natural U Reactor Void Defect

Figure 10. Natural U Reactor Temperature Coefficients

- eliminate positive coolant void reactivity
- reduce quantity of spent fuel discharged
- reduce quantity of Pu discharged.
- eliminate need for reprocessing spent fuel and recovery of plutonium
- provide options for long-term spent fuel storage
- use current technology and have minimum impact on facilities
- meet near-term implementation requirements.

Four fuel types were evaluated as shown in Table 1.

Each of these fuels met the screening criteria. However, the medium-enriched uranium cermet fuel can be fabricated at the existing Russian fuel plant that produces the cermet fuel for the spiked fuel columns in these production reactors (see Figure 5). Additionally, this alternative does not involve weapons useable material in fresh fuel and contains only low concentrations of highly burned plutonium ($> 30\%$ $^{240}\text{Pu}/\text{Pu}$) in the spent fuel. As a result, the medium-enriched cermet fuel was chosen as the preferred fuel option.

Table 1. Four Fuel Types Containing Natural Erbium Burnable Absorber

Feed Material	Fresh Fuel Composition	Cladding Material	Spent Fuel Quantity/Reactor
Low-Enriched Uranium	1.2 % Enriched Uranium Metal 0.5 wt % Erbium (burnable absorber)	Zircaloy-2 (coextruded with fuel)	2/3 core/yr
Medium-Enriched Uranium (< 20 % ^{235}U)	33 wt % U_3O_8 (19.9 % $^{235}\text{U}/\text{U}_{\text{total}}$) 6 wt % Er_2O_3 (burnable absorber) 61 wt % Aluminum (CERMET)	Aluminum (coextruded with fuel)	1/4 core/yr
Highly-Enriched Uranium (> 20 % ^{235}U)	7 wt % U (93 % $^{235}\text{U}/\text{U}_{\text{total}}$) 6.9 wt % Erbium (burnable absorber) 86 wt % Aluminum (ALLOY)	Aluminum (coextruded with fuel)	1/3 core/yr
Weapons-Grade Plutonium	20 wt % Pu (6 % $^{240}\text{Pu}/\text{Pu}_{\text{total}}$) 38 wt % Erbium (burnable absorber) 42 wt % Aluminum (ALLOY)	Aluminum (coextruded with fuel)	1/3 core/yr

3.0 Preferred Fuel Option

As stated in Section 2.2 the medium-enriched, uranium cermet fuel, coextruded with aluminum cladding, is the preferred fuel option for the operating Russian plutonium production reactors.

3.1 Fuel Description

The cermet fuel is in the form of a large rod, 34 mm in diameter, composed of 33 wt% U_3O_8 , 6 wt% Er_2O_3 , and 61 wt% aluminum metal, which is coextruded to final size with 1 mm thick aluminum cladding. These dimensions are the same as those for the current natural uranium metal fuel, shown in Figure 5. The uranium is enriched to only 19.9% $^{235}U/U_{total}$ to satisfy concerns regarding proliferation resistance of higher enrichments. Three percent of the fuel element length is aluminum end caps. Each of the 2600 fuel channels currently being fueled with natural uranium would be reloaded with the cermet fuel capable of high burnup (up to 1000 EFPD). With a four-batch core, the reactivity change from beginning to end of each annual cycle is estimated, from analyses shown in Figure 11, to be less than 4%. The changes due to an assumed 1 mm increase in fuel diameter and a 0.5 mm increase in water coolant thickness are analyzed in Appendix A to determine the sensitivity to these parameters, and show that they result in small improvements.

3.2 Fuel Exposure Limitations

Cermet fuel containing 33 wt% U_3O_8 (19.9% $^{235}U/U_{total}$) has about twice the ^{235}U concentration of natural uranium metal fuel. The 6 wt% erbium oxide (Er_2O_3) burnable absorber concentration is about the same as that approved by the Nuclear Regulatory Commission (NRC) and currently used in commercial light water reactor (LWR) fuel. With this composition of uranium and erbium oxides in the fuel, the reactivity changes with burnup are sufficiently small such that a 4-year, in-reactor residence time (up to 1000 EFPD) can be sustained, as shown in Figure 11. Nuclear fuel elements containing U_3O_8 dispersed in an aluminum matrix have been used in research and test reactors for about 30 years. Cermet fuel with coextruded aluminum cladding containing up to 50 wt% uranium has excellent irradiation behavior. It has been successfully irradiated to fission densities of about 2×10^{21} fissions/cm³ of fuel (Peacock 1989). This is an order-of-magnitude-higher fission density than would be achieved in 1000 EFPD of irradiation of cermet fuel proposed for use in Russian production reactors. The addition of Er_2O_3 to uranium oxide fuel in commercial LWRs has been demonstrated to have no significant effect on heat transfer, fission gas retention, or dimensional stability during irradiation to high exposures. (ABB/Combustion Engineering 1990; ABB/Combustion Engineering 1992.)

Figure 11. K-Infinity Versus Burnup

3.3 Safety Related Parameters

The use of erbium as a burnable absorber for graphite-moderated, water-cooled Russian production reactors is an innovative design application which provides the basis for eliminating the positive coolant void reactivity effect that is inherent in this type of reactor. Erbium is a rare-earth element, similar in chemical and metallurgical properties to gadolinium. Like gadolinium, erbium is comprised of several naturally occurring isotopes. The depletion chain is shown below. The natural abundancies and nuclear cross-section data are shown in Table 2. The primary neutron absorber is ^{167}Er .

The energy-dependent neutron absorption properties of ^{167}Er include a large, double resonance in the vicinity of 0.5 eV, as shown in Figure 12. This property enhances neutron absorption in the upper end of the thermal neutron spectrum and provides the additional characteristic of making the coolant temperature and void coefficients of reactivity negative, as shown in Figures 13 and 14. Negative coolant temperature and void coefficients improve safety by enhancing power stability in the reactor and making it more controllable. The Doppler coefficient of reactivity due to fuel temperature increases ranges from $-1.6 \times 10^{-5}/\text{K}$ to $-1.0 \times 10^{-5}/\text{K}$ from beginning to end of life. These values compare very favorably with the Doppler coefficient for the current natural uranium fuel, which is about $-1.2 \times 10^{-5}/\text{K}$ (Figure 10).

In contrast to gadolinium, erbium has a slower depletion characteristic as a burnable absorber, releasing reactivity gradually over a longer period of fuel burnup. This is due, in part, to the replenishment of ^{167}Er that has burned out with fresh material from ^{166}Er neutron absorption, as illustrated in Figure 15.

The residual reactivity worth of erbium at end-of-life of the fuel is very small because the ^{168}Er that builds up has low cross-sections for neutron absorption, as shown in Table 2. If the $\text{U}_3\text{O}_8/\text{Er}_2\text{O}_3/\text{aluminum}$ cermet fuel is irradiated in four batches in the core (1/4 of the fuel columns reloaded annually), the existing control rod system should have sufficient reactivity worth to provide adequate shutdown margin ($> 1\%$) while compensating for changes in reactivity during irradiation.

3.4 Fuel Burnup

As the cermet fuel is irradiated, more than half the ^{235}U initially loaded will be burned out, as shown in Figure 16. About 20% of the neutron absorptions in ^{235}U result in the formation of ^{236}U .

Figure 13. Temperature Coefficients Versus Burnup

Figure 14. Void Defect Versus Burnup

Figure 15. Erbium Isotopes Versus Burnup

Figure 16. Uranium Isotopics Versus Burnup

Table 2. Erbium Characteristics

	¹⁶⁶ Er	¹⁶⁷ Er	¹⁶⁸ Er	¹⁷⁰ Er
Isotopic Abundance	33.6%	22.95%	26.8%	14.9%
σ_a (Barns)	15	670	2.7	5.8
RI (Barns)	100	3000	37	2.2

The ²³⁸U content remains nearly constant because only small quantities of plutonium are formed. More than 99% of the actinide content in the spent fuel is uranium; only about 0.5% is plutonium. The concentration of each of the plutonium isotopes, as they buildup during irradiation, is shown in Figure 17. More than 1% of the plutonium is ²³⁸Pu. The principal non-fissile isotope, ²⁴⁰Pu, builds up to more than 30% of the total plutonium, as shown in Figure 18. Less than 30 kg of plutonium would be discharged from each reactor in spent fuel annually. This is nearly 20 times less than the current natural uranium fuel cycle. Due to the low concentration and higher isotopic content of the plutonium in this high-burnup cermet spent fuel, reprocessing would be unattractive.

3.5 Fuel Fabrication

About 100 fuel columns in each of the Russian plutonium production reactors contain a highly-enriched UO₂/aluminum cermet fuel that is clad with aluminum and coextruded to the same dimensions as the natural uranium metal fuel. As shown in Figure 5, this cermet fuel contains 8.5 wt% UO₂ (90% ²³⁵U/U_{total}) dispersed in aluminum. This fuel has the same ²³⁵U concentration as natural uranium metal. It is fabricated in the Novosibersk Fuel Plant. This plant also fabricates 20% enriched UO₂-aluminum cermet fuel for naval reactors. Up to 60 wt% UO₂ dispersed in aluminum has been fabricated and irradiated with excellent fuel performance. This is the preferred fuel form that the Russians are considering for conversion of the plutonium production reactors to civilian use (production of only electricity and district heat).

In the United States the UO₂ and U₃O₈ dispersions in aluminum were developed to increase the uranium density beyond that fabricable with the U-Al alloy, so that low enriched uranium (<20% ²³⁵U) could be used in research and test reactors throughout the world. This use reduces the risk of diversion and proliferation of highly enriched uranium. A core of 65 wt% U₃O₈ in aluminum at 20% enrichment was used successfully 30 years ago in the Puerto Rico Nuclear Center reactor (Richt et al. 1963). The irradiation performance of both U₃O₈ and UAl₃ dispersions was satisfactory. The cermet fuels made by powder metallurgical methods can accommodate up to about 50 wt% uranium in the core of fuel elements clad in aluminum and coextruded. This fabrication process is illustrated in Figure 19. The physical, mechanical, and chemical properties of cermet fuels developed in the United States are well documented (Peacock 1989).

Figure 17. Plutonium Isotopics Versus Burnup

Figure 18. Pu Percentages Versus Burnup

Figure 19. Powder Metallurgy Process

Addition of an erbium oxide burnable absorber to the U_3O_8 prior to blending with aluminum powder would be the only change to the fabrication process. Erbium has been extensively used in TRIGA fuel manufactured in the United States to provide a more negative fuel temperature coefficient for high enrichment uranium fuel (General Atomics). ABB/Combustion Engineering (ABB-CE) has more recently developed the application of erbium as a burnable absorber for pressurized water reactors (PWRs) in the form of Er_2O_3 admixed with enriched UO_2 . This fuel, containing up to 2.5 wt% Er_2O_3 , has been generically approved by the NRC and is in commercial operation in several ABB-CE-fueled PWRs. For extended fuel cycle lengths and high burnups, erbium burnable absorbers provide major advantages (ABB/CE, 1990 and 1992).

The fabrication of 33 wt% U_3O_8 /6 wt% Er_2O_3 /Aluminum cermet fuel, containing 19.9% ^{235}U enrichment, will result in an order of magnitude lower U concentration in fresh fuel than in natural uranium metal. The initial core loading of cermet fuel will contain about 4MT ^{235}U per reactor. Annual reloads will contain about 1MT ^{235}U per reactor.

3.6 Spent Fuel

The cermet fuel could be burned for up to 1000 EFPD, whereas the natural uranium metal fuel is currently being used for only 60 EFPD. Therefore, the quantity of spent fuel generated annually would be reduced by more than a factor of 16. Wet storage pools at the Russian production reactor sites are designed to hold 3-years worth of natural uranium-metal fuel discharges (12 cores per reactor). Because only 1/4 core of cermet fuel would be discharged annually, the storage pools could easily accommodate the reactor lifetime discharges of cermet fuel.

The decay heat of the cermet fuel would be about a factor of 10 higher than the natural uranium metal fuel. However, the total heat load in the storage pool would not approach the current operating level until cermet fuel had been discharged for 5 years. If the reactors have to continue operating beyond that time, the spent fuel pool cooling equipment (chillers) may need to be upgraded to accommodate the higher decay heat loads.

The spent fuel pools at Russian production reactor sites are not lined with stainless steel, nor is the water deionized. Aluminum clad (30 mils thickness) containing natural uranium metal fuel cannot be stored in these pools for more than 2 to 3 years because of the concern for aluminum clad failure due to pitting corrosion, as illustrated in Figure 20. However, the coextruded aluminum clad cermet fuel should be very durable in wet storage because the metallic aluminum in which the uranium oxide is dispersed would be self protecting. Pitting corrosion of the aluminum cladding would not cause a significant release of radioactive contamination to the pool.

Figure 20. Schematic of Al and Zircaloy Corrosion

3.7 Fuel Cycle Costs and Schedule

The process of changing to medium-enriched uranium oxide cermet fuel in Russian production reactors is expected to require about \$30M over a 2- to 3-year period. The developmental program would consist of optimizing the fuel design; verifying fuel models and codes; conducting in-reactor tests, demonstrations, and post-irradiation examinations; and verifying fuel performance and safety requirements. This developmental program would result in regulatory approval for full-scale implementation of fabrication and utilization of <20% enriched U_3O_8/Er_2O_3 /aluminum cermet fuel in the operating production reactors at Tomsk and Krasnoyarsk. The cost of fabricating this cermet fuel is not expected to be significantly different than the highly enriched uranium (HEU) oxide dispersed in aluminum fuel currently being used in about 100 spike channels of these reactors. The annual cost of fuel feed material for an entire reactor core would be \$72M based on the cost of HEU (\$18/gm ^{235}U), which could otherwise be blended with depleted U and sold commercially for use in LWRs. Annual reloads would cost an additional \$18M per reactor. Other fuel fabrication costs are expected to be about 10% of the costs of these feed materials. These fuel costs are about \$10 M/yr less than for the current natural uranium metal fuel. Each reactor is capable of producing about 200 MW of electricity (at 18% thermal efficiency) and 700 MWth of district heat, when operating at 2000 MWth. Since

estimated revenues from the sale of district heat and electricity per reactor could offset the estimated cost of operation and maintenance (\$4M/yr) and fuel (\$20M/yr), this should be a financially attractive option. A feasible schedule for implementing this cermet fuel is illustrated in Table 3.

Since the cermet fuel refueling would be reduced an order of magnitude, plant availability would be substantially increased. Refueling and maintenance outages could be scheduled annually at times when there is low demand for district heat, or electrical power, produced by the Russian plutonium production reactors.

Table 3. Conversion Schedule for Production Reactors to Use <20% Enriched UO₂/Erbium/Aluminum Fuel

Milestone	Time from Start in Months
• Joint Russian/U.S. Study of Fuel Alternatives/Safety Upgrades	3
• Modification of UO ₂ -Al Fuel Fabrication Facility	6
• Delivery of Test Fuel to Critical Facility at Kurchatov	8
• Completion of Critical Physics Verification Experiments	16
• Insertion of Lead Test Fuel Channels in Production Reactor	18
• Completion of Safety Analysis for Fuel Changes/Upgrades	20
• Delivery of First Reload of UO ₂ -Al Fuel to Production Reactor	22
• Begin Irradiation of UO ₂ -Al Fuel (Half Core)	24
• Last Discharge of Weapons-Grade Pu in Spent Fuel	26
• Complete Core Loading of UO ₂ -Al Fuel	26

4.0 Conclusions

Although the U.S. and Russia have a formal agreement on shut down of the Russian plutonium production reactors, it is not expected to be implemented until after the turn of the century. MINATOM will continue to operate these reactors to provide electricity and district heat in the meantime.

The earliest opportunity for stopping production of weapons-grade plutonium is to change the fuel cycle within the next 2 to 3 years. Such a fuel cycle change provides the opportunity to enhance the inherent safety characteristics of these reactors by adding erbium to the fuel, which can eliminate the undesirable positive coolant-void-reactivity effect.

The most proliferation-resistant fuel cycle that could be implemented quickly and reliably contains 33 wt% uranium oxide (19.9% $^{235}\text{U}/\text{U}$) and 6 wt% erbium oxide in aluminum, as a cermet, coextruded with aluminum cladding. This fuel cycle would be economical, minimize fuel fabrication and spent fuel storage requirements, enhance reactor operational safety, and discharge minimal quantities of reactor-grade plutonium containing $>30\%$ ^{240}Pu . The spent fuel would be suitable for long-term storage in existing pools.

U.S. assistance to Russia would likely be required for any near-term change in the fuel cycle for the Russian production reactors. Implementation of a proliferation-resistant fuel type containing erbium would require U.S. technical support for changes in fuel fabrication, materials behavior, reactivity and burnup analytical methods, and safety analysis. This expertise exists at PNL. This document provides the technical basis for further U.S. discussions with Russia concerning fuel cycle alternatives for the production reactors.

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Appendix A

Comparison of Results for Revised Fuel and Coolant Dimensions

Appendix A

Comparison of Results for Revised Fuel and Coolant Dimensions

The fuel rod and coolant channel dimensions were revised to evaluate its sensitivity on the results of this study. The original dimensions used in the calculations throughout the body of the report and the other appendices and the revised dimensions are compared below. A WIMS-E calculation was performed using a cermet fuel rod and coolant with the revised dimensions for comparison. The revised fuel rod is 1 mm larger in diameter, and the coolant channel is 0.5 mm wider. The increase in the volume of fuel results in a more negative Doppler coefficient and coolant temperature coefficient as shown in Figures A.1 and A.2 respectively. The void defect is also slightly more negative for the fuel cell with the revised dimensions as illustrated in Figure A.3. As expected, the larger fuel and coolant volume also results in a higher value of k-infinity at high burnups as shown in Figure A.4. However, the results given in the report are not very sensitive to this small change in fuel and coolant geometry.

	Original Dimensions	Revised Dimensions
Fuel Diameter	34 mm	35 mm
Al Clad Diameter	36 mm	37 mm
Water Coolant Thickness	2 mm	2.5 mm
Flow Tube Inner Diameter	40 mm	42 mm
Al Flow Tube Thickness	1.65 mm	1.65 mm
Square Lattice Pitch	200 mm	200 mm

Russian Production Reactor Cermet Fuel Cycle

Figure A.1. Doppler Coefficient Change with Burnup

Russian Production Reactor

Cermet Fuel Cycle

Figure A-2. Coolant Temperature Coefficient Change with Burnup

Russian Production Reactor

Cermet Fuel Cycle

Figure A.3. Void Defect Change with Burnup

Figure A.4. Reactivity Change with Burnup

Appendix B

Model Validation

Appendix B

Model Validation

The WIMS-E code model used for the calculations in this report was a single cell model using the Winfrith nuclear data library (1986 version). To ensure that the code model and nuclear data used are giving reasonable results for this particular application, several benchmark calculations were run to validate the results.

The model was validated using two methods. The first is a comparison to N-Reactor lattice experiment data reported in the 1960's. The comparison to the N-Reactor lattice serves to validate the model for metal fueled, graphite moderated, water cooled systems. While the N-Reactor and Russian production reactor fuels do not have exactly the same geometries, good comparison to this data provides a level of assurance in the calculation of graphite moderated-water cooled systems which would not otherwise exist.

The second method used is code comparisons. Data relating to the fuel assemblies analyzed for this report does not currently exist. Therefore, the WIMS-E calculations have been compared to other codes using differing calculational methodologies and/or cross section sets to ensure consistency in calculational results across a wide range of reactor configurations. The codes to which WIMS-E was compared are MCNP and DIT.

B.1 N-Reactor Lattice Comparison

N-Reactor was a uranium metal fueled, graphite moderated, water cooled reactor located on the Hanford Site in Washington. Its graphite moderation and water cooling make it similar in design to the Russian production reactors analyzed in this report. Data taken from N-Reactor lattice measurements therefore serve as a reasonable benchmark for WIMS-E for this type of reactor.

Lattice measurements for N-Reactor lattices were performed in the Physical Constants Test Reactor (PCTR) on the Hanford site before N-Reactor was operated (Johnson 1963). One of the lattice parameters determined from the measurements was the neutron multiplication factor (k -infinite) of the lattice for various coolant densities.

A WIMS-E unit-cell model of the N-Reactor lattice was developed and k -infinite calculations were performed for various water densities. A comparison of the measured and calculated results is shown

in Table B.1. The maximum difference between the calculated results and the nominal measured results is about 7 mk. These results are in good agreement and give confidence that the code model will produce adequate results for graphite moderated lattices.

B.2 MCNP Comparisons

The available N-Reactor data is limited to naturally or slightly enriched uranium metal lattices. For this report, it was desired to use WIMS-E to calculate a larger range of lattice designs, including the possibilities of oxide fuels (as cermet), and possibly non-fertile fuel loadings. Additionally, the inclusion of burnable absorbers in the fuel was desired.

MCNP is a generalized geometry Monte Carlo reactor physics code which can be used to calculate reactor parameters such as k-infinite, nuclide reaction rates, and other general reactor parameters (Los Alamos National Laboratory 1990). MCNP is used widely throughout the world for nuclear design, criticality, and benchmarking calculations.

Prior work was performed at PNL benchmarking WIMS-E to MCNP for a light water assembly lattice using 10% enriched uranium fuel (Reid 1992). This comparison showed that WIMS-E and MCNP were in agreement to within 6 mk for the lattice. Additionally, the fission rates for each of the pins in the lattice were compared. The maximum difference in fission rate for any pin was 3%, with most pins showing differences less than 1.5%.

Additional work at PNL was performed comparing WIMS-E and MCNP for mixed oxide and non-fertile plutonium oxide fuels containing erbium oxide burnable absorbers. The fissile oxide fuel pins were light water moderated, and calculations were made using a range of erbium oxide loadings in the pins. Figure B.1 shows the results of these calculations. The plot shows that the MCNP and WIMS-E calculations are in good agreement over the range of erbium concentrations.

Table B.1. Comparison of WIMS-E to N-Reactor Lattice Measurements

<u>Water Density (g/cm³)</u>	<u>Measured K-inf</u>	<u>WIMS-E K-inf</u>
0.000	1.018 ± 0.003	1.0238
0.404	1.046 ± 0.003	1.0515
0.759	1.069 ± 0.003	1.0673
1.000	1.079 ± 0.003	1.0721

Figure B.1. WIMS-E/MCNP WNP-1 Pincell Comparison

B.3 Comparison to DIT

The DIT code is a two dimensional transport code used for reactor analysis by ABB/CE (CE Proprietary Topical Report 1993). The DIT code package is accepted by the USNRC for use in the design of nuclear reactor fuel. The DIT nuclear data library was recently updated to include ENDF/B-VI cross section evaluations. Code calculations using this new library have been compared extensively to critical data and tests and has been found to produce acceptable results (U.S. Department of Energy 1994).

A comparison of WIMS-E and DIT calculations for mixed oxide fuel containing erbium oxide in a light water lattice has been made (U.S. Department of Energy 1994). This comparison included burnup calculations and isotopic reaction rates at up to 42 GWD/MTHM burnup. The reactivity, isotopic reaction rates, and isotopic concentrations throughout the burnup range compare well between the two codes. All differences were considered to be acceptable.

B.4 Conclusions

While benchmarks for the exact types of fuel being analyzed in this report are limited or non-existent, the benchmark calculations performed provide a level of assurance that WIMS-E is calculating reasonable results for the Russian production reactors. The comparison to the N-Reactor lattice experiments indicates that WIMS-E performs adequately on water-cooled graphite-moderated lattices similar to the Russian production reactors. The benchmarks to MCNP indicate that WIMS-E is calculating reasonable results for moderately enriched uranium assemblies and for oxide fuels containing erbium. The comparisons to DIT indicate that the burnup results from WIMS-E are within acceptable ranges which give confidence in the depletion results for cells containing high plutonium and erbium concentrations.

While the validation performed by PNL is by no means complete, all indications are that the WIMS-E results for the Russian production reactors should be acceptable for the scoping analysis performed to date.

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Appendix C

Additional Options for Russian Production Reactor Operation in a Civilian Power Mode

Appendix C

Additional Options for Russian Production Reactor Operation in a Civilian Power Mode

The main body of this report discusses the preferred fuel option for operating the Russian production reactors in a civilian power mode. This preferred option was determined through the comparison of proliferation resistance, relative safety parameters, and costs. Several additional options have been investigated. This appendix serves to document the results of the options analyzed in addition to the one preferred.

C.1 Fuel Cycle Options

Four additional fuel cycle options were analyzed in support of this report as summarized in Table C.1. The options are:

1. long term dry storage
2. conversion to a zircaloy-clad LEU metal fuel cycle
3. conversion to a zircaloy-clad non-fertile HEU fuel cycle
4. conversion to a zircaloy-clad non-fertile WG plutonium fuel cycle.

C.2 General Information

The current fuel cycle for the operating Russian production reactors is described in the body of the report. As a result of continuing operations, about 1000 MT of spent fuel is discharged from each of the production reactors annually. The reactors will continue operation and produce plutonium with the aluminum-clad uranium-metal fuel currently being irradiated if viable civilian fuel cycles are not implemented.

The storability of aluminum-clad spent uranium-metal fuel in water is very sensitive to water quality. Experience (e.g., Argentina, Hungary, Italy) with aluminum-clad spent nuclear fuel has shown acceptable storage in deionized water for up to 24 years. The evidence of integrity is primarily based on measurement of radionuclides released to the pool. Pitting corrosion in deionized water

Table C.1. Fuel Option Comparisons

	Natural U	LEU	HEU	Pu	Cermet
Fuel Development Cost	NA	\$20M	\$30M	\$50M	\$30M
In Reactor Residence Time	60 EFPD	360 EFPD	750 EFPD	750 EFPD	1000 EFPD
Net Fabrication Cost Reduction Factor	-	3	6	2	8
U Feed Material Cost Reduction Factor	-	2.5	2	NA	1.5
Current Storage Pool Capacity	3 years	18 years	lifetime	lifetime	lifetime
Front End Proliferation Concerns	Low	Low	High	High	Low
Back End Proliferation Concerns	High	Moderate	Moderate	Moderate	Low
Other Costs	NA	Update spent fuel cooling	Update spent fuel cooling	Update spent fuel cooling	Update spent fuel cooling

was projected to penetrate aluminum cladding after approximately 30 years of pool storage (IAEA 1992). Al-clad fuel has failed in impure water in 10 to 15 years.

The spent fuel pools at Russian production reactors are not lined with stainless steel nor is the water deionized. Spent fuel is stored in these pools for no more than two to three years because of the concern for Al-clad failure. The consequences of cladding failure during wet storage are release of radioactive species to the pool and eventual degradation of the cladding and exposed fuel. Radioactive particulates can eventually be released into the pool.

Some of the fuel cycle options analyzed consider the use of zircaloy-clad fuel instead of aluminum-clad. Zircaloy-clad fuel has shown no perceptible degradation in deionized water for periods up to approximately 30 years. Zircaloy-clad fuel has not shown a sensitivity to the purity of the pool water. See the main report for additional discussion of clad corrosion.

The U.S. NRC has issued rules permitting wet storage of zircaloy-clad spent fuel for up to 100 years. It is expected that zircaloy-clad fuel for Russian production reactors could be maintained in wet storage for a similar period of time without difficulty.

To implement any of the alternative fuel cycle options (except dry storage) for the production reactors would be expected to require a 3 to 5 year development program (specific lengths for specific options are listed below) to optimize fuel design, verify fuel models and codes, conduct in-reactor tests, demonstrations, and post-irradiation examinations, and to verify fuel performance and safety requirements. The development program would result in regulatory approval for full scale implementation of fabrication and utilization of the fuel option in Russian production reactors. The cost of fabricating the uranium fuel options is not expected to be more than a factor of two higher than that for natural uranium metal fuel. Plutonium fuel fabrication costs would be somewhat higher (as explained below). However, the amount of fuel fabricated in all cases is reduced in comparison to natural uranium fuel, so net fuel fabrication cost reductions result.

C.3 Option Analysis

C.3.1 Dry Storage Option

The spent fuel from the Russian production reactors could be stored instead of being reprocessed to recover the plutonium. Wet storage of spent fuel in pools would be used for the first two to three years after discharge to reduce the decay heat rate to less than 300 W/MT. The spent fuel would be transferred into large metal dry storage casks such as the GNS-CASTOR5 and inerted for long-term monitored storage.

Spent Fuel Pool Storage Limitations

Based on three years of pool storage, about 3000 MT of spent fuel would be stored underwater at each Russian production reactor. This is consistent with current practice and does not impose any new limitations.

Dry Storage Requirements

Instead of reprocessing three-year cooled spent fuel from pool storage, dry storage casks would be used to store spent fuel in an inert atmosphere. Inerted dry storage has been licensed in the U.S. and other countries for commercial spent fuel and used for storage of aluminum-clad test reactor fuel in Japan.

Large metal dry storage casks, such as the CASTOR-5, being used in the U.S. and Europe would be used for long-term monitored storage of the spent fuel. Each of these casks has a mass of about 100 MT empty with an overall size of about eight feet in diameter and 16 feet in height.

Since the aluminum-clad uranium-metal fuel has about twice the density of commercial LWR fuel, about 20 MT of production reactor spent fuel could be loaded into each cask. The loaded weight of the cask (120 MT) would be a practical limit considering typical crane capacity available at reactor sites. The heat dissipation rate from the cask (<6000 watts) would keep the metal fuel from rapidly oxidizing if the inert atmosphere in the cask were lost.

The cost of these dry storage casks (in large quantities) is about \$1M apiece. The expected cost for obtaining dry storage casks at each reactor would be about \$50M/year. About three acres of land per year would be added to the space occupied by dry storage casks at each reactor.

Dry Storage Option Costs

In addition to the \$50M per reactor-year for obtaining dry storage casks, a reinforced concrete pad about half a meter thick would be required for the dry storage array (\$2M/reactor-year). Monitoring costs are expected to be about \$10M/reactor-year. Cask loading and handling costs are expected to be about \$3M/reactor-year. Thus, the total cost for the dry storage of spent fuel at each reactor is expected to be about \$65M/year.

C.3.2 Zircaloy-Clad LEU Metal Option

The production reactors could continue operation with a zircaloy-clad fuel containing about 0.5 wt% erbium alloyed with 1.2 wt% ²³⁵U enriched uranium metal. This fuel could be fabricated by a hot co-extrusion process similar to that used for N-Reactor. The spent fuel would be maintained in wet storage and would not be reprocessed for up to 100 years.

Fuel Exposure Limitations

Fuel containing 1.2 wt% ^{235}U in uranium metal with zircaloy clad has been irradiated in N-Reactor to exposures up to 4400 MWd/MT. To enhance the negative coolant temperature coefficient and void coefficient during irradiation, 0.5 wt% erbium would be alloyed with the 1.2% enriched uranium metal fuel (Figures C.1 and C.2). The erbium also acts as a burnable absorber which minimizes the reactivity change of the fuel during burnup (Figure C.3). Allowing for an axial power peaking factor of 1.2, the channel-average discharge exposure of fuel would be about 3600 MWd/MT, six-times the current natural uranium metal fuel exposure. As a result, only about 160 MT of spent fuel would be discharged from each of the Russian production reactors annually. The spent fuel would contain about 0.7 wt% of fuel-grade plutonium ($> 17\%$ ^{240}Pu). The plutonium and erbium concentrations as a function of burnup are shown in Figures C.4 and C.5, respectively.

Spent Fuel Pool Storage Limitations

The decay heat of LEU/erbium metal fuel elements would be about a factor of 5 higher than that of natural uranium-metal spent fuel. Therefore, the spent fuel pool cooling equipment would need to be upgraded to accommodate the higher decay heat loads associated with the more highly burned fuel.

Spent Fuel Storage Requirements

Since the zircaloy-clad LEU/erbium metal fuel would be burned for 360 EFPD, whereas the natural uranium-metal fuel is currently being burned for only 60 EFPD, the quantity of spent fuel generated annually would be reduced by a factor of 6. Wet storage pools designed to hold three years' worth of natural uranium-metal fuel discharges could accommodate 18 annual discharges of the more highly burned zircaloy-clad LEU/erbium metal fuel.

LEU Metal Fuel Option Costs

To implement the LEU metal fuel burning option in Russian production reactors is expected to require a \$20M development program over the next two years.

The amount of LEU metal fuel fabricated is reduced by a factor of 6 in comparison to natural uranium fuel, so the net fuel fabrication cost is reduced by about a factor of 3.

The cost for LEU which could otherwise be sold commercially for use in LWRs is estimated to be about \$6.20/gm ^{235}U . This corresponds to about \$75K/MT of 1.2% enriched U metal. The LEU feed requirements for each production reactor operating on LEU fuel would be about 160 MT/yr. The LEU feed costs amount to about \$12M/yr for each reactor. This amounts to only 40% of the annual cost for natural uranium feed material (1000 MT/yr) per reactor for the current fuel cycle.

C.6

Fuel Composition:
1.2 wt% U235
0.5 wt% Er

■ Doppler Coefficient ▲ Coolant Temperature Coefficient

Figure C.1. LEU Temperature Coefficient 0.5% Er

Figure C.2. LEU Void Defect 0.5% Er

Figure C.3. LEU K-Infinity 0.5% Er

Figure C.4. LEU Pu Buildup

Figure C.5. LEU Er Isotopic Burnup

HEU Disposition Rate

Since the LEU/erbium metal fuel would reside in the Russian production reactors for 18 months, two-thirds of the core would be discharged annually from each reactor. The total quantity of LEU metal fuel that would have to be fabricated and loaded into all three of the production reactors annually to sustain their operation could contain 5.1 MT of HEU blended with 475 MT of depleted uranium.

C.3.3 Zircaloy-Clad Non-fertile HEU Option

The production reactors could continue operation with a non-fertile fuel containing highly-enriched uranium (HEU) and erbium in an aluminum matrix clad with zircaloy. This fuel would not produce significant quantities of plutonium, and most of it would be non-fissile ^{238}Pu . The spent fuel would be maintained in wet storage, and would not be reprocessed for up to 100 years.

Fuel Exposure Limitations

Fuel containing 7 wt% HEU and 6.9 wt% erbium in an aluminum matrix can sustain a three-year in-reactor residence time (750 effective-full-power days). As shown in Figure C.6, the reactivity change with burnup is about 12% over the fuel lifetime. After equilibrium xenon has built up, the reactivity increases for the first 480 EFPD, reaches a maximum value of $K_{\infty} = 1.12$, and then decreases. As shown in Figure C.7, more than 60% of the uranium is annihilated during the fuel lifetime. The discharged uranium contains less than 60% ^{235}U , and more than 27% ^{236}U .

Safety Related Parameters

Non-fertile HEU/Er/Al fuel, with the same clad dimensions (18 mm radius) as the natural uranium-metal fuel currently being used, has a Doppler coefficient that changes linearly from $-9 \times 10^{-6}/\text{K}$ at beginning-of-life to $-2 \times 10^{-6}/\text{K}$ at end-of-life. These values are somewhat less negative than those for natural uranium ($-1.1 \times 10^{-5}/\text{K}$).

The coolant temperature coefficient also becomes less negative with burnup, changing from $-1.2 \times 10^{-4}/\text{K}$ at BOL to $-5 \times 10^{-5}/\text{K}$ at EOL. These values are much more negative than the value for natural uranium-metal ($-6 \times 10^{-6}/\text{K}$ at BOL). The temperature coefficients are shown in Figure C.8.

The reactivity change with burnup will require about 4% more control rod worth than for the natural uranium-metal fuel. A comparison of the 7 wt% HEU/Er non-fertile fuel burnup reactivity change with that of natural uranium during the first year of irradiation shows similarities, as seen in Figure C.9. The burnout of erbium (Figure C.10) slightly overcompensates for the reactivity loss due to ^{235}U burnout in the non-fertile HEU/Er fuel.

Figure C.6. HEU K-Infinity 6.9% Er

Figure C.7. Uranium Isotopic Burnup for HEU 6.9% Er Fuel

C.14

Figure C.8. HEU Temperature Coefficient 6.9% Er

Figure C.9. Comparison of Reactivity Changes with Burnup for Natural Uranium Metal and HEU/Er/Al Alloy Fuels

C.16

Figure C.10. Erbium Isotopic Burnup for HEU Er 6.9% Fuel

Spent Fuel Pool Storage Limitations

The decay heat of the non-fertile HEU fuel elements would be about a factor of 10 higher than that of the natural uranium-metal spent fuel. Therefore, the spent fuel pool cooling equipment would need to be upgraded to accommodate the higher decay heat loads associated with more highly burned fuel.

Spent Fuel Storage Requirements

Since the non-fertile HEU/Er/Al fuel would be burned for 750 EFPD, whereas the natural uranium-metal fuel is currently being used for only 60 EFPD, the quantity of spent fuel generated annually would be reduced by more than a factor of 12. Wet storage pools designed to hold 3-years worth of natural uranium-metal fuel discharges could accommodate the reactor lifetime discharges of non-fertile HEU fuel.

Non-fertile HEU Fuel Option Costs

To implement the HEU burning option in Russian production reactors is expected to require a \$30M development program over the next three years.

The amount of fuel fabricated is reduced by a factor of 12 in comparison to natural uranium fuel, so the net fuel fabrication cost is reduced by about a factor of 6.

The cost for HEU which could otherwise be blended with depleted U and sold commercially for use in LWRs is estimated to be about \$18/gm ²³⁵U. This corresponds to about \$17M/MT of HEU. The HEU feed requirements for each production reactor operating on non-fertile fuel would be about 0.9 MT/yr. The HEU feed costs amount to about \$15M/yr for each reactor. This amounts to only half the annual cost for natural uranium feed material (1000 MT/yr) per reactor for the current fuel cycle.

HEU Disposition Rate

Since the non-fertile HEU/Er/Al fuel would reside in the Russian production reactors for three years, one-third of the core would be discharged annually from each reactor. The total quantity of non-fertile fuel that would have to be fabricated and loaded into all three of the production reactors annually to sustain their operation would contain 2.6 MT of HEU. About 1 MT of reactor-grade uranium would be discharged annually in spent fuel.

C.3.4 Zircaloy-clad Non-fertile Weapons Grade Plutonium Option

The Russian production reactors could be reloaded with a non-fertile fuel containing weapons-grade plutonium and erbium in an aluminum matrix clad in zircaloy. The spent fuel would be maintained in wet storage, and would not be reprocessed to recover the remaining reactor-grade plutonium for up to 100 years.

Fuel Exposure Limitations

Fuel containing 20 wt% weapons-grade plutonium and 38 wt% erbium in aluminum can sustain a 3-year in-reactor residence time (750 effective full-power days). As shown in Figure C.11, the reactivity loss with burnup is only 7% in 750 EFPD for this fuel. As shown in Figure C.12, plutonium in the spent fuel contains about 80% of the initial amount of plutonium loaded, but the ^{240}Pu content is increased to $> 17\%$ (about the same as that in VVER spent fuel).

Safety Related Parameters

Non-fertile fuel, with the same clad dimensions (18mm radius) as the uranium-metal fuel currently being used, has a Doppler coefficient that remains fairly constant at a value of $-1.2 \times 10^{-5}/\text{K}$, as shown in Figure C.13. The water coolant temperature coefficient changes linearly from $-1.6 \times 10^{-5}/\text{K}$ at beginning-of-life to $-1.0 \times 10^{-5}/\text{K}$ at end-of-life. The fractional destruction of both ^{239}Pu and ^{167}Er are closely matched as the fuel is burned, as shown by comparisons of Figures C.12 and C.14. Burnout of about one-fourth of these isotopes essentially compensates for their reactivity effects over the fuel lifetime. Lower initial concentrations of plutonium and erbium result in less closely matched destruction rates and shorter lifetimes before the coolant temperature coefficient becomes positive.

Spent Fuel Pool Storage Limitations

The decay heat of non-fertile fuel elements would be about a factor of 10 higher than that of natural uranium-metal spent fuel. Therefore, the spent fuel pool cooling equipment would need to be upgraded to accommodate the higher decay heat loads associated with longer burned fuel.

Spent Fuel Storage Requirements

Since the non-fertile Pu/Er/Al fuel would be burned for 750 EFPD, whereas the natural uranium-metal fuel is currently being used for only 60 EFPD, the quantity of spent fuel generated annually would be reduced by more than a factor of 12. Wet storage pools designed to hold 3-years worth of natural uranium-metal fuel discharges could accommodate the reactor lifetime discharges of non-fertile fuel.

Figure C.11. K-Infinity Versus Burnup for Pu/Er/Al Fuel

Figure C.12. Plutonium Isotopic Burnup for Pu/Er/Al Fuel

C.21

Fuel Composition:
42 wt% Al
20 wt% Pu
38 wt% Er

■ Doppler Coefficient

▲ Coolant Temperature Coefficient

Figure C.13. Temperature Coefficient for Pu/Er/Al Fuel

Figure C.14. Erbium Isotopic Burnup for Pu/Er/Al Fuel

Non-fertile Weapons Grade Plutonium Option Costs

Implementation of the plutonium burning option in Russian production reactors is expected to require a \$50M development program over the next five years.

The cost of fabricating plutonium-bearing fuels for LWRs has been four to five times higher than for uranium fuels because the processes must be conducted in glove boxes to contain the potential spread of contamination. Since the quantity of plutonium-bearing fuel required is a factor of 12 lower than that for natural uranium-metal fuel, the net annual fuel fabrication cost should be less than half that currently being incurred.

Plutonium Disposition Rate

Since the non-fertile Pu/Er/Al fuel would reside in the Russian production reactors for three years, one-third of the core would be discharged annually from each reactor. The total quantity of non-fertile fuel that would have to be fabricated and loaded into all three of the production reactors annually to sustain their operation would contain 8 MT of weapons-grade plutonium. About 6 MT of reactor-grade plutonium would be discharged annually in spent fuel.

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