ADVANCED NEUTRON SOURCE ENRICHMENT STUDY*

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ABSTRACT

A study has been performed of the impact on performance of using low enriched uranium (20% ^{235}U) or medium enriched uranium (35% ^{235}U) as an alternative fuel for the Advanced Neutron Source, which is currently designed to use uranium enriched to 93% ^{235}U . Higher fuel densities and larger volume cores were evaluated at the lower enrichments in terms of impact on neutron flux, safety, safeguards, technical feasibility, and cost. The feasibility of fabricating uranium silicide fuel at increasing material density was specifically addressed by a panel of international experts on research reactor fuels. The most viable alternative designs for the reactor at lower enrichments were identified and discussed. Several sensitivity analyses were performed to gain an understanding of the performance of the reactor at parametric values of power, fuel density, core volume, and enrichment that were interpolations between the boundary values imposed on the study or extrapolations from known technology.

INTRODUCTION

The Advanced Neutron Source is a nuclear reactor that is being designed by Oak Ridge National Laboratory under the sponsorship of the U.S. Department of Energy. Its purpose is to produce intense quantities of neutrons for use in fundamental and applied research in physics, chemistry, biology, medicine, and materials technology. The performance goal is to build a machine with a neutron beam intensity, or flux, that is at least five times higher than existing facilities. The Advanced Neutron Source will be a more powerful research tool than existing facilities and will replace some facilities after their useful lifetime is reached.

As with all other research reactors that have been operated for the purpose of producing a very high flux of neutrons, the Advanced Neutron Source is designed to burn highly enriched uranium (HEU) fuel. This means that the fuel is comprised of 93% of the isotope ^{235}U and 7% of the isotope ^{238}U . With this isotopic mix, the design meets the performance goal, has acceptable safety characteristics, and is feasible to build within state-of-the-art engineering practices and cost envelopes.

Considerable effort has gone into the design of the reactor over the past several years and the design has evolved as new information became available or requirements were imposed. The budget guidance for fiscal 1994 for the Advanced Neutron Source included the directive that a study be

*This work was performed under the auspices of the U.S. Department of Energy.

conducted of the impact on performance of using medium enriched uranium (MEU) fuel or of using low enriched (LEU) fuel. LEU contains a mix of 20%/80% of $^{235}\text{U}/^{238}\text{U}$, and MEU contains ratios greater than that of LEU but less than that of HEU. For the purposes of this paper, MEU is defined as uranium containing 35% ^{235}U . The Department of Energy requested that Brookhaven National Laboratory lead this Enrichment Study.

Because of concerns about HEU fuel being diverted for non-peaceful purposes, both in the USA and elsewhere, this study was performed. The logic was that, if the USA forgoes the use of HEU in its plans for the Advanced Neutron Source, other countries might be persuaded to do likewise in their plans for new high-performance research reactors. Compared to LEU or MEU, HEU is much more attractive to those who would seek to divert uranium fuel for non-peaceful purposes. For perspective, a prompt critical system (consisting of an unmoderated, unreflected sphere of uranium metal) based on HEU involves approximately 50 kilograms, and one based on MEU and LEU involves one tonne and six tonnes, respectively. The total uranium content for a core of the existing HEU design of the Advanced Neutron Source is approximately 25 kilograms.

APPROACH

The Brookhaven Study involved the participation of three other national laboratories with special expertise in fuel enrichment studies of research reactors. These are Argonne National Laboratory, which has conducted extensive evaluations of HEU to LEU conversion of research reactors worldwide, Oak Ridge National Laboratory, which is responsible for the design of the Advanced Neutron Source, and Idaho National Engineering Laboratory, which contributes technically to the design and has much experience in the design of research and test reactors. The study was conducted in a collegial manner; the laboratory participants agreed upon a mode of technical inquiry and on work assignments for each laboratory. A set of calculations were agreed upon for various enrichments, core volumes, and fuel densities. Technical criteria for the acceptability of results were defined. In order to perform the analysis within the confines of the schedule, it was decided that Oak Ridge National Laboratory and Idaho National Engineering Laboratory run the computer cases that the four laboratories determined should be run. Argonne National Laboratory and Brookhaven National Laboratory provided quality assurance checks of the calculations by running selected cases at their own organizations and with their codes. Interim study results were evaluated jointly, and areas for further investigation were defined, and tentative conclusions were identified. All of the laboratories performed careful and critical reviews of key assumptions and results. Argonne and Brookhaven requested additional calculations which became available to all participants. Between meetings of the Study Group, the participants performed analysis at their respective institutions.

During the course of the study, a special expert panel on fuels was convened to assess the feasibility of developing and manufacturing a postulated fuel form that would be needed in the reactor at lower enrichments. The panel was comprised of international experts in fuel design, manufacture, and performance. The conclusions of this panel are also included in this study.

The scope of the Enrichment Study, as defined by the Department of Energy, was to work within the existing design of the reactor, not produce a much higher power reactor that would greatly

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increase the capital costs, and to investigate the implications of using a hypothetical fuel with material density approximately five times greater than the fuel specified for the existing design.

The impact of using either low or medium enrichment fuel in the ANS was measured by considering the change in the following four parameters.

1. Neutron Flux - As a representative parameter the maximum unperturbed thermal neutron flux in the reflector was used as a measure of performance. The value of this flux was compared to the baseline design goal, which in turn has been fixed at 7.0×10^{19} n/m²-s.
2. Cost - The cost of the ANS is composed primarily of two components: construction and operating costs. Changes in the construction costs are a function of the reactor power and significant deviations from the currently chosen operating power (330 MW) have an impact on the plant cost. Changes in the operating cost are dominated by the number of fuel elements fabricated and burned per year.
3. Safety - The safety impact of using lower enrichment uranium is composed of several elements and some of these are given here. First, the higher fertile content of the fuel enhances the Doppler coefficient, reducing the demand on the control system. Second, the higher fuel density required with lower enrichments reduces the fuel thermal conductivity, and hence increase the fuel centerline temperature and reduce the safety margin. Lower enrichment cores may require larger volumes with radially longer fuel plates of lower curvature, which would be mechanically less stable at high coolant velocities. The power density in alternative core configurations must be kept within acceptable safety margins. There would be increased plutonium build-up with irradiation and this would impact the cleanup following a severe accident.
4. Safeguards - The safeguards dimensions are measured in terms of the requirements of implementing safeguards programs in the U. S. as a function of enrichment, the potential for diversion of fuel elements, the production of plutonium, and the implications for international policy. While this study was motivated by international policy concerns, its objective was to focus on the technical impacts of using alternative enrichments for the fuel of the Advanced Neutron Source. The study does determine the implications of potential alternative designs on the DOE domestic safeguards program, on the number of cores that would be required for non-peaceful purposes, and on the amount of plutonium produced. These parameters provide a measure of the significance of designs with various enrichments and this may be useful in determining the implications for international policy.

The Study Group determined that two parameters should independently be varied to assess the impact of using either MEU or LEU in the reactor. One parameter is the uranium fuel density which would be increased to compensate for decreased ²³⁵U content in the lower-enriched fuels. The other parameter is the reactor core volume, which would be increased to compensate for reactivity losses that would result from lowering the enrichment. Thus the fuel density was varied from the existing design value of 1.7 gU/cc to values in excess (in response to the directive of the Department of Energy) of 6 gU/cc. Three core volumes were studied: the existing core volume of 67.6ℓ, and two larger cores of 82.6ℓ and 108ℓ. The existing core is comprised of two cylindrical shell fuel elements,

and the two larger cores each contain three cylindrical shell fuel elements. The 108ℓ core is a hypothetical example that was constructed to study the physics behavior of a large core. In practice, this core design may suffer from large and unsafe deflections of the fuel plates due to forces acting on the wide relatively flat, and, therefore, flexible plate span. Thus, an additional research and development program would be required to investigate the mechanical fluid dynamic, heat transfer, and safety implications of the 108ℓ core, were it to be selected for the ANS.

RESULTS

Many potential configurations of fuel density, enrichment, and core volume were analyzed and only those that met criteria for sufficient initial reactivity, acceptably safe power density and fuel temperature, and the capability of sustaining an acceptable (at least 17 days) core life were retained for further consideration. A wider range of configurations was considered than are feasible or desirable in order to enhance intuition with regard to the impact of parametric variations. Table 1 is a summary of nineteen cases that were considered.

Case 1 was the reference design of the ANS when this study was performed. Based on the many configurations evaluated, the following main conclusions are drawn.

1. HEU is better than lower enriched uranium fuels for the flux performance of the Advanced Neutron Source. In particular, configurations with enrichments of 35% or less consistently led to flux performance that is inferior to the HEU design.
2. If the enrichment were to be reduced to 35% (which we now define as reference MEU), then a reactor configuration was identified (case 6) which meets the above criteria. In this reactor, the core volume would be increased to 82.6ℓ, the fuel density would be increased to 3 gU/cc, the power would remain at 330 megawatts, but the resulting neutron flux would be approximately 20% below the reference design. The additional cost of the project, above the current cost of the existing design, would be approximately \$0.4 billion. This cost increase is mostly due to an increase in operating costs over the lifetime of the plant. Only \$5M additional would be needed for total project costs including an increased fuel development program. The uranium mass of the core would be approximately 60 kg, and from a safeguards perspective, 17 full cores would be required to achieve a prompt critical system. For the reference case, two full cores would be required for a prompt critical system.
3. If the enrichment were to be reduced to 20%, then a reactor (case 9) could be designed within the confines of the technology which would be utilized for the reference plant. The core volume would be increased to 82.6ℓ, the fuel density would be increased to 3.5 gU/cc (the practical upper limit), the power would have to be decreased to 125 megawatts, and the resulting neutron flux would be approximately 70% below the reference design. The additional cost of the project would be approximately \$70 million. This cost increase results from a \$160M increase in operating costs over the plant's lifetime relative to the reference plant and approximately \$90M decrease in total project costs because this reactor would operate at a much lower power. At this enrichment, more than 88 full cores would be needed to achieve a prompt critical system.

Table 1 - Performance Comparisons for Various Cores

Case	Element Number	Power (MW)	Enrichment (%)	Fuel Density (gU/cc)	Relative Flux	TPC* Penalty	OC* Penalty	Pu Prod.* (kg/yr)	DOE Safeguards Category ⁺
1	2	330	93	1.7	1.0	0	0	0.95	I
2	2	330	80	2.1	0.99	0	0	1.92	I
3	3	330	50	2.0	0.80	+5	+400	3.22	II/III
4	3	400	50	2.2	0.96	+29	+480	3.95	I/III
5	3	405	35	3.5	0.93	+20	+480	7.24	III
6	3	330	35	3.0	0.78	+5	+400	5.53	III
7	3L**	530	20	7.2	0.97	>202	>520	23.74	III/IV
8	3L	449	20	6.0	0.82	+202	+520	18.10	III/IV
9	3	125	20	3.5	0.3	-86	+160	3.6	III/IV
10	3L	330	20	4.8	0.65	+135	+360	12.07	III/IV
11	2	330	35	6.5	0.91	+132	0	9.21	III
12	3	340	50	2.0	0.82	+8	+360	3.28	II/III
13	3	370	35	3.3	0.85	+19	+440	6.62	III
14	3L	430	20	5.8	0.82	+171	+480	17.18	III/IV
15	3	330	93	1.0	0.83	+5	+400	0.53	II
16	2	200	45	3.5	0.59	-54	-200	4.06	III
17	3	250	45	1.8	0.64	-27	+280	2.66	III
18	3	200	35	2.1	0.52	-48	+240	2.80	III
19	3L	260	35	1.8	0.54	-21	+280	3.64	III

* \$M, TPC = Total Plant Cost, and OC = Operating Cost Over 40 Years

14 cycles per year are assumed

**3L = 108l 3 element core; all other 3 element cores are 82.6l

+Determined by DOE Order 5633.3A

These main conclusions are given for design configurations that would have the best technical chance of succeeding for the stated enrichments and with the derived flux and cost penalties. For MEU fuel, it is possible to design other reactors (case 5) for which the flux penalty relative to the existing design would be approximately 10%. This can be achieved by increasing the power to 405 megawatts, the maximum permissible based on heat removal considerations, and increasing the fuel density to 3.5 gU/cc. The additional cost of this design, relative to the reference design, is \$0.5B. A high flux can also be achieved with MEU by increasing (case 11) the fuel density to 6.5 gU/cc in the reference core volume with power at 330 megawatts. However, it was the conclusion of the Fuel Experts Panel that a program to develop fuel with the required heat transfer properties and dimensional tolerances at a uranium density 6.5 gU/cc has more than 90% chance of failure.

For LEU fuel, it is also possible to design reactors with a less severe flux penalty than described in item 3. A reactor can be designed (case 10) with LEU that has a 35 % flux penalty relative to the reference design, would operate at 330 megawatts in the largest core volume, 108ℓ, and requires a fuel density of 4.8 gU/cc. Again, the Fuel Experts Panel have judged that development of fuel for the Advanced Neutron Source at this density has a significant chance of failure. Higher fluxes can be achieved (case 8) by increasing the power in this core to 449 megawatts and increasing the fuel density to 6.0 gU/cc. This implies a flux penalty of 18%. In addition, a research and development program would be needed to assess the technical feasibility of the 108ℓ core.

The Study Group also evaluated the impact of reducing the enrichment of the fuel to 80%, 50%, and to 45%. For 80% enrichment (case 2), no significant differences were found in performance, cost, technical feasibility, or safety. Three full cores (6 elements) would be needed to achieve a prompt critical system. For the case of 50% enrichment, the flux penalty could be limited to 10% provided that the power is increased to 400 megawatts in the 82.6ℓ core. The fuel density would be increased to the reasonably achievable value of 2.2 gU/cc, but the additional operating costs would be \$0.5 billion over the life of the facility. On the order of eight cores would be needed to achieve a prompt critical system. The 45% enrichment cases were lower power density studies in the 67.6ℓ and 82.6ℓ core volumes. Both led to approximately 40% flux penalties with fuel densities that do not exceed 3.5 gU/cc.

Safeguards requirements are determined by Category as shown in Table 1. Requirements for Categories I and II include material control and accountability planning and management, threat considerations, performance criteria, accounting systems, physical inventories, measurement control, control limits, loss detection elements, training, access controls, containment, surveillance, etc. For Categories III and IV, requirements are determined by the local DOE Field Office and are less stringent. The differences between Categories I and II are small (with respect to the effort and cost of following requirements). The requirements for Categories III and IV are significantly less obtrusive, but the cost of following the requirements may not be significantly less than meeting the requirement for Categories I or II.

FUEL EXPERTS PANEL EVALUATION

The ANS-LEU Fuel Panel was convened as part of this project to assess the feasibility of achieving higher density LEU fuels for the ANS. This Panel consisted of five international experts on aluminum-based fuels for research and test reactors (see Acknowledgement Section).

They noted that there is no fuel in commercial production that can be compared with the ANS proposed designs: the gradients of the meat (lateral and longitudinal), power densities, heat flux, temperatures, dimensional tolerances, and percent burn-up each require extrapolations of known technology in fabrication, inspection, and irradiation performance, even for the HEU 1.7 gU/cc fuel currently being considered.

The Panel attempted collectively to quantify its conclusions, based on past experience with other fuel types and on the intuitive judgment of each individual member, with the results given in Reference 1. Heat transfer properties, which are essential in a high performance reactor, degrade

significantly at densities greater than 3.5 gU/cc. Panelist Yves Fanjas noted that, with techniques that are still proprietary, CERCA has successfully rolled flat plates, with no gradient or high heat transfer requirements, at meat concentrations of 4.8 and 6.0 gU/cc. However, 6.0 gU/cc must be counted as the maximum density achievable using conventional plate fabrication technology. Densities greater than 6.0 will certainly require development of new fuel plate fabrication technologies.

The Panel concluded that increasing the fuel density up to 3.5 gU/cc, using the current fabrication technology and U_3Si_2 fuel particles, will stretch current technology, but probably will not add greatly to the costs or decrease significantly the probability of success. Going to fuel loadings of 4.8 gU/cc and higher will introduce larger costs and uncertainties and require considerable development effort. Fuel loadings greater than 6.0 gU/cc will require a major development program, including development and testing of new fabrication technologies, which has a low likelihood of success.

QUALITY ASSURANCE

The quality assurance process forms an integral aspect of the analysis work performed in the project. A two pronged approach was taken in this effort. First, the numerical techniques were validated against all relevant critical experiments. The Monte Carlo physics methods are judged by their ability to reproduce the measured results of the FOEHN experiments (see Reference 1). All other physics methods, diffusion theory, and deterministic transport theory are judged by their ability to reproduce the Monte Carlo results. Second, independent calculational efforts were carried out by ANL and BNL staff, to check selected core analyses. In the case of the Monte Carlo calculations this consists primarily of checking the input parameters for consistency, and executing a selection of problems. In the case of the burn-up calculations, the ANL team started with a description of the core and created an independent input file which was executed on their software package for representative range in core volumes, enrichments and fuel densities (see Reference 2 for details). In this manner both the Monte Carlo and burn-up steps of the analysis are checked.

Quality assurance discussions from ANL and BNL are included in Reference 1. Monte Carlo calculations carried out at BNL agreed very closely with those carried out at INEL using different versions of the MCNP code.

In general, ANL results for the 93% enrichment core agree with ORNL results. There is also good agreement on fluxes and initial reactivities for the reduced enrichment cores, but the lifetimes calculated by ANL for the cores with 35% and 20% enrichment are significantly shorter than those reported by ORNL. However, there are sufficient differences in the methods employed by ANL and ORNL to account for the differences observed in the calculated results. Additionally, the results are, in general, consistent with previously reported comparisons of deterministic and stochastic methods used on the ANS design project [3]. Nevertheless, an additional set of calculations are currently being done to identify and resolve the causes of these differences. Finally, it should be pointed out, that regardless of the outcome of these calculations and the resolution of the differences, all study participants agree that the main results of this paper (and Reference 1) do not change.

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