THE OHIO STATE UNIVERSITY

Engineering Experiment Station

142 Hitchcock Hall
2070 Neil Avenue
Columbus, OH 43210-1275

Phone 614-292-4903

May 6, 1988

Mr. Theodore S. Michaels
Project Manager
Standardization and Non-Power Reactor
  Project Directorate
Division of Reactor Projects III, IV, V
  and Special Projects
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington DC 20555

<u>SUBJECT:</u> **Response to Questions Regarding Fuel Conversion**

Dear Mr. Michaels:

Enclosed are responses to your questions related to conversion of The Ohio State University Research Reactor (OSURR) to use low-enrichment uranium (LEU) fuel. In addition to the responses to your questions, modified Technical Specifications for the OSURR are included which reflect initial operation with LEU at the current 10 kilowatt steady-state thermal power level, with a maximum excess reactivity of 1.5% delta k/k. Additional comments follow.

Changes to our proposed Technical Specifications for the initial request for 500 kilowatt operation were made to reflect initial operation with LEU at 10 kilowatt, pending resolution of questions related to 500 kilowatt operation submitted under a separate cover. It continues to be our understanding that the Commission will act on the power change request in a [illegible]nner through the normal license amendment process, once we have su[illegible]ponses to your questions. The changes proposed herein are made in resp[illegible]itten comments from your office and in subsequent telephone communications between Commission representatives and OSURR staff on or about February 25, 1988.

An attempt has been made to maintain consistent pagination between this submission and the initial submission of the revised Technical Specifications and Safety Analysis Report. As a result of brief additions, two minor page changes have been made. Page number 255a was added at the end of Section 3, and page 274a was added to the end of Section 6. Changes are noted by a vertical line in the right-hand margin.

The table on page 248 showing the Reactor Safety System components required to be operable was modified to delete those requirements for 500 kilowatt operation. The table entries were renumbered to reflect these changes. Sections 3.3.1, 3.3.5, 3.6.1(2), 4.6.2, and 5.5.2, which relate to 500 kilowatt operation were left in place to maintain numbering continuity, but with the notation "This section deleted for 10KW operation." These sections will be applicable when

A020
1/1

Technical Specifications for 500 kilowatt operation are resubmitted. Also, since our current Technical Specifications for 10 kilowatt operation require the safety system to initiate a reactor trip at 15 kilowatts (150% full power) from the power level safety channels, this limit has been retained as a limiting safety system setting. Section 2.2 reflects this requirement.

The University Reactor Operations Committee approved these changes to the Technical Specifications in their meeting of March 10, 1988. Committee approval of the responses to the questions related to fuel conversion was obtained on April 27, 1988.

It is our understanding that the regulatory process from this point will involve review of these responses, which, if approved, will lead to issuance of an order to allow us to proceed with the fuel conversion phase, based on a schedule we would propose under a separate cover. This conversion schedule would take into account availability of LEU fuel elements for the OSURR (to be supplied by DOE under the Fuel Assistance Program) and the operating cycle of the the OSURR. We will develop this schedule in consultation with your office and DOE representatives.

As per established procedure, this office will continue to be the point of contact between the Commission and the University for official communications regarding the status of OSURR license changes. For questions or comments related to the technical content of the enclosed documents, please contact Mr. Joseph Talnagi of the OSURR staff at 614/292-6755. For matters related to overall project administration and policy, communications should be directed to Dr. Don W. Miller, Director of the OSURR, at 614/292-7979.

Thank you for your continued assistance and cooperation in the OSURR conversion effort.

Sincerely,

Robert F. Redmond

Robert F. Redmond
Director

RFR/s

Enclosures

c: D. W. Miller
J. Talnagi
R. Myser
K. Brown, DOE
L. K. Seymour, EG&G Idaho, Inc.

HEU-LEU Fuel Conversion

The Ohio State University Research Reactor

License No. R-75

Docket No. 50-150

Responses To Questions

April 19, 1988

1. **During the conversion from HEU to LEU is there a time period when the fuel inventory of both cores will be at the OSURR?**

   Yes. Current plans call for removal of the existing HEU fuel elements from the reactor pool and storing them in a safe, secure storage area to allow additional decay of the fission product activity inventory prior to shipment of the fuel to a DOE facility.

   a. **If so, have the fuel storage facilities at OSURR been analyzed for safe storage for criticality concerns?**

      Yes, a storage rack has been designed and analyzed for criticality safety. The design has been reviewed and approved by a subcommittee of the University's Reactor Operations Committee. Full committee approval is expected shortly. This analysis will be submitted to the NRC for approval.

      This plan has already been submitted to NRC for approval of changes in the physical security plan. Communications have been established between the OSURR staff and the Safeguards office of the NRC. The Safeguards office has been kept informed of our intentions. Except for very minor wording changes, the security plan has received final NRC approval.

   b. **Is the total fuel inventory of both cores allowed by the current OSURR license?**

      No. R-75 license limits are not high enough to allow the total U-235 expected for simultaneous possession. It is our understanding the NRC order to convert to LEU can incorporate a (temporary) provision to allow simultaneous possession of the two fuels up to the limit of their total U-235 content. Such a provision is desirable in the case of the OSURR because of the extremely serious economic penalties that might result from prior shipment of HEU and possible delay in procuring and using the LEU fuel. Thus, we request that the NRC incorporate in their conversion order a provision to allow simultaneous possession of both HEU and LEU, up to the limit obtained by the sum of the existing R-75 license limits and the total LEU fuel inventory as described in the revised Safety Analysis Report.

2. **What facility hardware modifications are being made for the LEU core, i.e., core support, control rods, control rod drives, experiment facilities, or any other hardware changes necessary to accommodate the new core.**

   Very few changes are being made to system hardware that are specifically related to the fuel conversion. Indeed, it is our understanding that one of the policy goals of the NRC in issuing the rule to require LEU conversion was to minimize facility changes. In the case of the OSURR, a few relatively minor changes are envisioned. For example, the neutronic analysis of the LEU core indicates that to maintain reactivity limits the LEU core must be smaller than the HEU core (i.e., fewer fuel elements). Thus, the grid plate will have a few vacant positions that will require plugs to be fabricated and placed in the vacant positions to maintain desirable thermal-hydraulic characteristics (plugs are not for reactivity "shimming"). The row of 5 graphite isotope irradiation elements (GIIE) along the east edge of the OSURR core will also have to be removed and the positions filled with plugs.

3. **What changes are being made in the instrumentation system for the LEU core? Is there new instrumentation not previously discussed in the HEU SAR?**

   There are no instrumentation changes being made that are specifically related to the LEU conversion. The revised SAR describes additional instrumentation that will be required for power change, but these are <u>not</u> specific to the fuel conversion.

4. **Page 1**

   It is suggested that you refer to the fuel enrichment as "less than 20%, which qualifies it as LEU fuel, somewhere early in your discussion. It is further suggested that when you ref[illegible] the actual enrichment, you precede the enrichment value with the word nominal, i.e., "enriched to a nominal 19.5%."

   No action required. We recognize that enrichment values are nominal 19.5%, and enrichments less than 20% qualify fuel as LEU by existing conventions.

5. Page 25

The developer of the fuel is mentioned. Please give specific references to publications you have used to support your analysis throughout this SAR.

Development and testing of the U_3Si_2 fuel to be used in the OSURR, as well as other fuel types developed and tested for other uses, are comprehensively described in the following documents:

"Performance of Low-Enriched U_3Si_2-Aluminum Dispersion Fuel Elements in the Oak Ridge Research Reactor", G.L. Copeland, R.W. Hobbs, G.L. Hofman, and J.L. Snelgrove, Report No. ANL/RERTR/TM-10, Argonne National Laboratory, Argonne, IL, October 1987.

"The Use of U_3Si_2 Dispersed in Aluminum in Plate-Type Fuel Elements for Research and Test Reactors", J.L. Snelgrove, R.F. Domagala, G.L. Hofman, T.C. Wiencek, G.L. Copeland, R.W. Hobbs, and R.L. Senn, Report No. ANL/RERTR/TM-11, Argonne National Laboratory, Argonne, IL, October 1987.

Other references are listed at the conclusion of those chapters in the revised SAR that utilize references. In addition to references listed, guidance in methodology and techniques was obtained from other sources, primarily the SARs of other currently or previously licensed research reactors. These included the University of Missouri-Rolla Reactor (License No. R-79, Docket No. 50-123), The University of Lowell Reactor, The Ford Nuclear Reactor at the University of Michigan, and the Battelle Research Reactor (License No. R-4, Docket No. 50-6).

6. Page 31

Partial fuel elements are mentioned. Are these all fabricated as partial fuel elements or is it planned that plates are removable (or additive) by the Licensee? A picture or drawing of a fuel assembly is requested.

All partial fuel assemblies will be supplied by the fuel fabricator as partial elements. There are no plans to have fuel assemblies that will allow removal or addition of fuel plates by the Licensee.

Sketches of a standard fuel element and control rod fuel element are shown on pages 29 and 30, respectively, of the revised SAR. Attachment A of this document is a copy of the design drawing for the partial fuel elements.

7. Page 95

It is indicated that a single standard fuel element contains eight fueled plates. Please explain.

This is a typographical error. In section 4.2.2 on page 95, paragraph 3, line 10, the parenthetical notation for "8 fueled plates" should be changed to "16 fueled plates".

8. Page 105

Please furnish a reference for the VIM code.

The following references discuss the operation and development of the VIM code:

R.E. Prael and L.J. Milton, "A User's Manual for the Monte Carlo Code VIM", FRA-TM-84, Feb. 20, 1976.

E.M. Gelbard and R.E. Prael, "Monte Carlo Work at the Argonne National Laboratory", Proceedings of the NEACRP Meeting, Monte Carlo Study Group, ANL-75-2, Page 201, Argonne National Laboratory, Argonne, IL

9. Page 106, Section 4.5

a. In paragraph 2 you discuss the temperature coefficient of reactivity but do not distinguish between fuel-related and moderator-related coefficients. Please discuss.

Figures 4.2-4.4 of the Safety Analysis Report show some of the core configurations analyzed for the OSURR. The analyses included estimates of both the fuel-related and moderator-related reactivity effects for each core configuration. As expected, he fuel-related Doppler coefficients showed little variation, while the moderator-related temperature coefficients showed a slight change with core configuration. The following estimates were obtained:

Average Moderator Temperature Coefficient for 23 to 60 Degrees C. in $\%\Delta k/k/^{o}C$.:

$-4.9x10^{-3}$ (minimum)
$-5.9x10^{-3}$ (maximum)

Average Fuel-Related Doppler Coefficient for 23 to 300 Degrees C. in $\%\Delta k/k/^{o}C$.:

$-1.1x10^{-3}$ (minimum)
$-1.3x10^{-3}$ (maximum)

Total Temperature Feedback at 40 Degrees C. Average Coolant Channel Temperature and 80 Degrees C. Average Fuel Temperature in $\%\Delta k/k$:

-0.15 (minimum)
-0.18 (maximum)

The estimate shown on page 106, section 4.5, is the total average temperature coefficient of reactivity.

b. In paragraph 3 you mention the THOR reactor, without providing a description, or even a reference. Please discuss the relevance of the THOR reactor to the OSURR.

The THOR reactor is an open pool, light water moderated research reactor with a power level of 1 megawatt, operated by the Tsing Hua National University in the Republic of China (Taiwan). The full citation is as follows:

"One and a Half Year Experience with the Tsing Hua Open-Pool Reactor (THOR) and Its Utilization", staff of the Reactor and Radioisotope Department, Institute of Nuclear Science, National Tsing Hua Un versity, Republic of China, Preliminary Draft 1962, NP14832

10. Page 156

In Section 6.4 please refer explicitly to your arrangement with DOE to accept irradiated U_3Si_2-Al fuel elements. What are the details of this agreement?

Historically, DOE has retained title to reactor fuel used by university-owned research reactors. DOE enters a fuel assistance contract with the university to provide the reactor fuel at no charge to the user (no costs for enrichment or burnup are applied). The contract also allows for return of the material to DOE when utilization is ended, either by burnup, decommissioning, or fuel conversion. The OSURR falls under this arrangement and this was confirmed by personal communication with Mr. Keith Brown of EG&G Idaho, Inc., managers of the DOE fuel assistance program, on 4/4/88. It is our understanding at this time that DOE will accept irradiated LEU U_3Si_2-Al fuel elements provided that the appropriate information on the fuel (fuel characterization) is provided, and that the university, as the shipper of record, be responsible for arranging the fuel shipment.

11. What are your calculated maximum, axial, and radial thermal flux peaking factors? What are your calculated maximum, axial, and radial power peaking factors?

For the core discussed in Attachment B to this document, the radial power distribution was estimated based on element-averaged power per fuel plate divided by the core-averaged power per fuel plate. On this basis, the radial power peaking factor is estimated to be 1.285, and the axial power peaking factor is estimated to be 1.275. The total peaking factor is thus 1.64. It is expected that these power peaking factors will be approximately the same as the thermal flux peaking factors. Two-dimensional representations of the neutron flux profiles for various neutron energy groups are shown in Attachment B of this document for the core described therein.

12. For the Reactivity Insertion Accident:

a. **Is the 0.7% $\Delta k/k$ step reactivity insertion the same as analyzed for the HEU core?**

No, the accident analysis for the HEU-fueled OSURR, which is contained in the approved Hazards Summary Report (HSR) for the OSURR on file with the NRC, analyzes a reactivity insertion of 1.5% $\Delta k/k$. The revised SAR also analyzes a step reactivity insertion of 0.93% $\Delta k/k$ (about \$1.20). Since the existing (and revised) Technical Specifications limit the maximum experiment worth installed in or near the core at any one time to 0.7% $\Delta k/k$ (total of secured and unsecured experiments), it seems reasonable to define the maximum credible reactivity insertion accident as a 0.7% $\Delta k/k$ step insertion.

Attachment B to this document describes an LEU core and an analysis related to a 1.5% $\Delta k/k$ reactivity insertion.

b. **Please compare the consequence parameters for the HEU analysis and LEU analysis, i.e., peak power, energy release, maximum fuel, clad, and moderate (sic) temperature, and, if applicable, any fission product release.**

The safety analysis for the HEU-fueled OSURR considered a reactivity insertion of 1.5% $\Delta k/k$. A prompt-critical reactor period of 9.4 milliseconds was estimated. The resulting power transient is self-terminating by a combination of negative temperature and void coefficients of reactivity after reaching a peak power of about 1350 megawatts, with a total energy release of 30 megawatt-seconds. The maximum cladding temperature (hot channel) is estimated to be 636 degrees F. The fuel meat temperature will be slightly higher than this cladding temperature, but not significantly so, since the heat transfer from the fuel to the cladding is relatively efficient for this fuel type (aluminide HEU). Obviously, moderator temperature near the fuel plates will be high enough to induce steam void formation, which is the dominant shutdown mechanism for the postulated power transient. The maximum cladding surface temperature will be well below the melting point of 1100 aluminum, which will allow no fission product release.

Section 8.4.3, starting on page 199 of the revised SAR discusses a reactivity insertion accident for the LEU-fueled OSURR, operated at a power level of 600 kilowatts, with core configuration I as shown in Chapter 4. The worst-case accident analysis assumes a \$1.20 step reactivity insertion, uses the most pessimistic estimate for void coefficient, and takes no credit for a scram. The peak power is estimated to be 49 megawatts, and the peak cladding temperature is 146 degrees C. (hot channel). For total energy release, the area under the curve for case 4 shown on page 208 (Figure 8.4) indicates the total energy release during this transient. This is estimated to be approximately 9 megawatt-seconds or less. The maximum fuel cladding temperature is much less than the melting point of 6061 aluminum, so no fission product release will occur.

Attachment B to this document provides a comparison between the accident analyses for the HEU-fueled OSURR operated at 10 KW and a similar LEU-fueled core assuming a 1.5% $\Delta k/k$ reactivity insertion. For the LEU core, the accident consequences are less severe than for the HEU core, for the reasons discussed in the attachment.

c. **On page 108 you discuss the measured reactivity effect of different beam port configurations for the HEU core. What are these effects for the LEU core?**

Based on the fuel conversion experience of the Ford Nuclear Reactor (FNR) at the University of Michigan, we expect that the reactivity effects of the various beam port configurations will not be significantly different for the LEU core compared with those measured for the HEU core. This reasoning is logical in that at these locations the primary reactivity effects are a result of neutron reflection, which is dominated by the material nature of the reflector and its geometry. These will be unchanged with the conversion to LEU fuel. If any effects are noticed at all, we would expect an insignificant increase in the reactivity worth of neutron-reflecting materials placed in the beam port positions, since with the use of LEU fuel there will be slightly more fast neutrons available to leak out of the core, and be thermalized and/or reflected back into the core.

d. **On page 200 you discuss the reactivity effects of flooding either of the two beam ports or the rabbit (0.5% $\Delta k/k$ each) for the HEU core. What do you estimate this effect is for the LEU core?**

For these effects, an argument similar to that presented in response to question 12c above applies. Flooding of these experimental facilities (assuming they are completely voided and filled with air at the time of flooding) places a material (light water) near the core that is at once a neutron reflector, absorber, and thermalizer. In these locations, we expect the major reactivity effects to result from neutron reflection, which again should be almost identical to those estimated for the HEU core. If any effect is noticed at all, it might be expected to insignificantly increase the reactivity effects for the same reason as discussed in the response to question 12c.

e. **What is the estimated reactivity effect of voiding the Central Irradiation Position?**

First, it should be noted that the Central Irradiation Position (CIF) is normally voided. It is occupied by a hollow aluminum tube within which materials are placed for irradiation experiments. The CIF normally contains mostly air. Thus, flooding the CIF with light water causes a negative reactivity effect.

For the core described in Attachment B of this document, the local void coefficient of the normal CIF irradiation position is positive. The reactivity insertion caused by voiding the CIF position is estimated to be +0.45% $\Delta k/k$. As noted above, however, the CIF is normally voided, so any credible failure of the CIF will cause it to be flooded with light water, adding negative reactivity to the core.

13. **Describe those administrative steps to be taken to inform OSURR operational personnel of the changes resulting from the use of LEU fuel.**

The OSURR has approved procedures in place for informing personnel of changes related to the reactor system. These procedures will be followed to assure that appropriate information is disseminated to the staff.

Based on the experience of the FNR fuel conversion, we expect very little, if any, measurable changes in the operational characteristics of the OSURR. Thus, the fuel change, for the most part, should be "transparent" from the viewpoint of reactor operation. Training sessions are planned with licensed personnel to provide information on the characteristics of the LEU fuel elements, and measured effects. Information on measured changes in core physics parameters will be provided to appropriate operational personnel. For example, if the effective delayed neutron fraction is modified as a result of the fuel change, personnel will be so informed. Similarly, if control rod worths change slightly, the measured changes will be documented and information related to operational procedures will be updated. Again, we expect that any changes that may occur will be minor, and will have little effect on OSURR operational procedures. Thus, we feel that briefing sessions with licensed personnel, and written internal communications will appropriately inform personnel of any changes. Extensive operator retraining and/or relicensing or requalification will not be required as a result of the fuel change. To provide written documentation of measured changes in core physics parameters and related reactor characteristics, OSURR personnel will be provided a copy of the Executive Summary of the final report submitted to the Department of Energy as a requirement for the grant received by the University to assist in the fuel conversion effort. This summary will contain information on measured changes in reactor characteristics.

14. **Describe and analyze an alternative LEU-fueled OSURR core for operation at 10 kilowatts of steady-state thermal power with an excess reactivity of 1.5% $\Delta k/k$.**

This core is described and analyzed in Attachment B to this document.

Attachment A

Partial Fuel Element Design Drawing

Attachment B

Description and Analysis of An LEU Core

Providing 1.5% ΔK/K Excess Reactivity

B.1 Introduction and Background

The revised SAR discussed and analyzed an optimized LEU core (denoted core I in Chapter 4) assuming 500 kilowatt operation of the OSURR. Higher power operation requires additional excess reactivity to account for temperature feedback and xenon poisoning. Accordingly, the revised Technical Specifications show an excess reactivity limit of 2.6% $\Delta k/k$, which is the minimum estimated to allow for all feedback effects, plus allowance for experimental worth (0.7% $\Delta k/k$) and generation of reasonable reactor periods (0.2% $\Delta k/k$).

The NRC review requested that the LEU-fueled OSURR be analyzed for initial operation at the existing 10 kilowatt level, and for the currently-licensed excess reactivity limit of 1.5% $\Delta k/k$. In response, an alternate LEU core geometry was analyzed which has an estimated excess reactivity of about 1.5% $\Delta k/k$. The analysis presented in the revised SAR chapters 4 and 8 remains valid and many of the parameters estimated therein are applicable to the alternate core geometry discussed in this document. The discussion and analysis to follow presents additional justification and detail on the characteristics of this core under normal and postulated transient conditions.

B.2 Core Description and Characteristics

Reduction of the excess reactivity is attained by reducing the fuel loading in the core. Essentially, standard fuel elements are removed to reduce reactivity, while still maintaining adequate control rod worths to meet shutdown reactivity margin requirements. Any vacant grid plate positions are assumed to be filled by an aluminum plug, as discussed in the revised SAR.

Figure B-1 shows a sketch of the proposed LEU core having a 1.5% $\Delta k/k$ excess reactivity. This core has a total of 16 standard fuel elements and 4 control rod fuel elements. The estimated U-235 loading is a nominal 3700 grams.

B.3 Core Neutronics

Figures B-2 and B-3 show the calculated neutron flux profiles in the north-south and west-east directions, respectively for four different neutron energy groups. As expected, the thermal neutron flux (Group 4) peaks in the Central Irradiation Facility (CIF) position, and also shows reflector peaking in the graphite thermal columns extensions to the west and south of the core. The fast and epithermal neutron fluxes tend to dip in the CIF position since there is no fuel at this position and neutrons are either absorbed or scattered (thereby losing energy) in the water that partially fills the CIF position.

Table B-1 summarizes some of the significant core neutronics parameters. The 1.5% $\Delta k/k$ excess reactivity requirement is satisfied, as well as restrictions noted in the Technical Specifications related to minimum shutdown margin and maximum regulating rod worth.

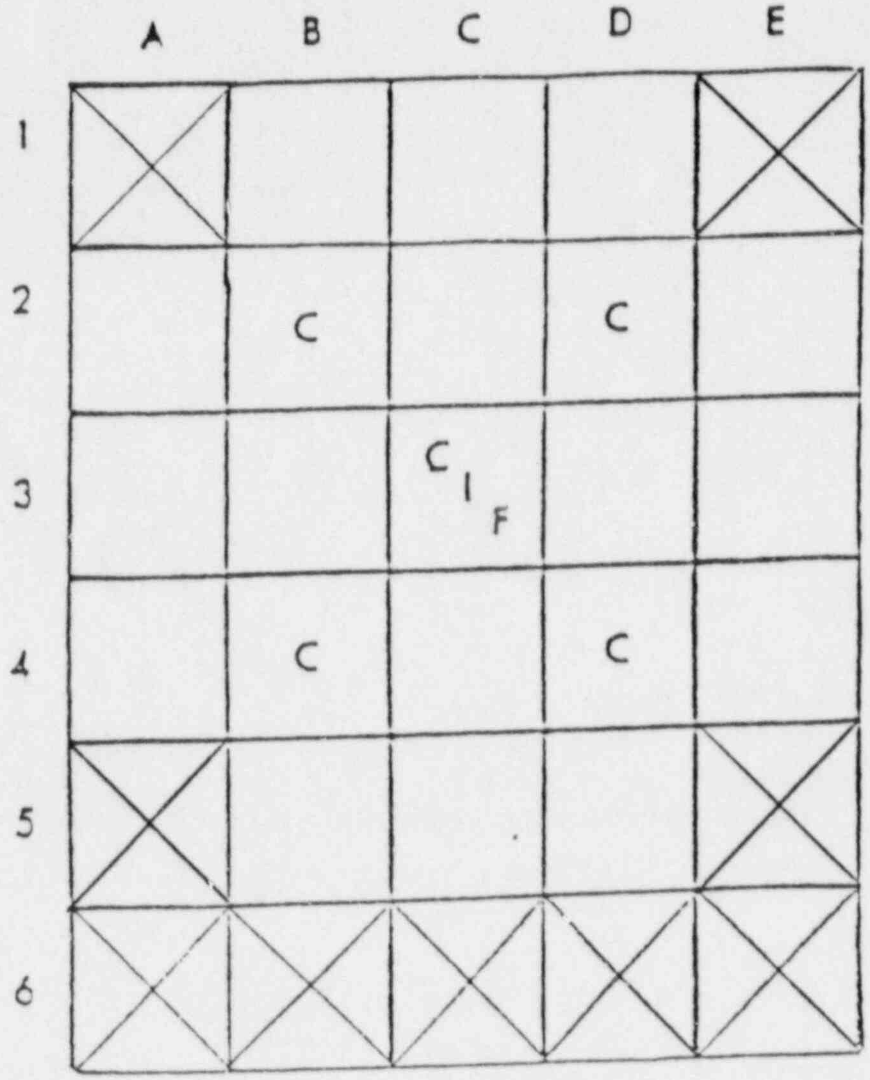

Figure B-1

Proposed LEU Core For The OSURR
With An Estimated Excess Reactivity of 1.5% $\Delta K/K$

(X's Denote Plugged Grid Plate Positions)

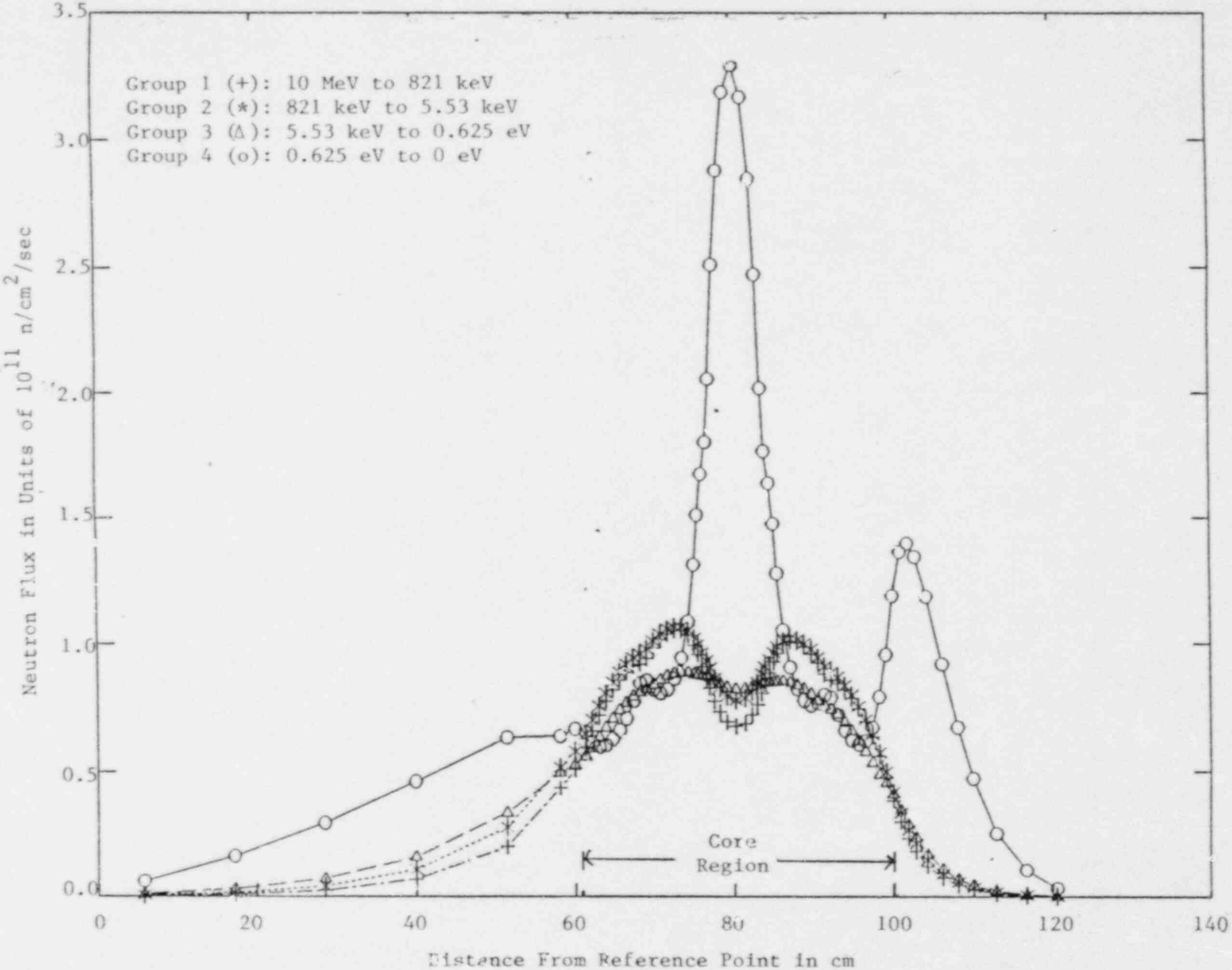

Figure B-2
Four-Group Neutron Flux Distribution
In The North-South Direction

Figure B-3

Four-Group Neutron Flux Distribution
In The West-East Direction

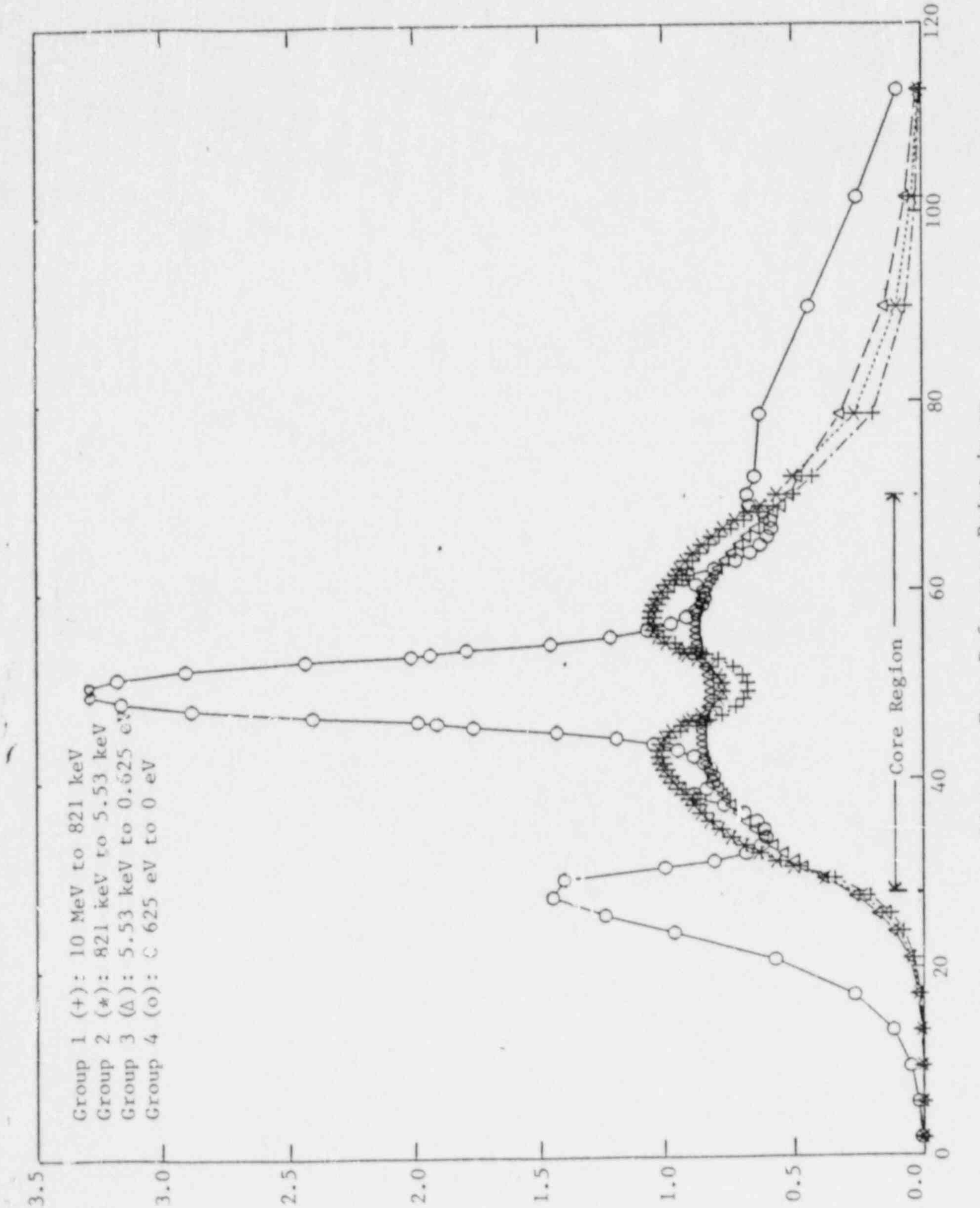

Table B-1

LEU Core Parameters

Excess Reactivity: 1.5% $\Delta k/k$

Radial Peaking Factor: 1.285 (maximum)

Axial Peaking Factor: 1.275 (maximum)

Graphite Reflector Element Reactivity Worth:

Position A1: 0.56% $\Delta k/k$
Position A5: 0.25% $\Delta k/k$

Shim Safety Rod Reactivity Worths:

Position B2: 2.70% $\Delta k/k$
Position D2: 2.47% $\Delta k/k$
Position D4: 2.16% $\Delta k/k$

CIF Position Void Coefficient Reactivity Worth: +0.45% $\Delta k/k$

Minimum[1] Shutdown Margin: 3.09% $\Delta k/k$

Total Core Loading: 3700 grams U-235 (nominal)

Core Geometry: Symmetric Array of 16 standard elements, 4 control elements

Notes:

(1) The shutdown margin estimate assumes that the most reactive shim safety rod and the regulating rod are fully withdrawn and remain so while the other two shim safety rods are fully inserted.

B.4 Thermal Hydraulics Analysis

In the revised SAR, a thermal analysis of the OSURR operated at 500 kilowatts, with natural convective cooling through the core, is presented in Chapter 4, beginning with section 4.8. A similar analysis can be done for the OSURR operated at 10 kilowatts, but it is obvious that at lower operating powers the various parameters estimated in chapter 4 will be lower, e.g., fuel plate surface temperature, coolant temperature, and coolant flow rate. Thus, we can assume that the analysis presented in Chapter 4 provides an "envelope" within which the OSURR operated at lower power levels will operate safely.

For example, in section 4.8.2 beginning on page 110 an estimate is made of the fuel plate surface temperature for the OSURR operating at 500 kilowatts. A similar analysis can be done for 10 kilowatt operation using the same constants provided for the 500 kilowatt power level. At 10 kilowatts, using a core made from 16 standard fuel elements and 4 control rod fuel elements, a power density of 1.74 watts/cc is obtained in the active fuel portion of the core. Inserting this value into the appropriate equations leads to a temperature difference of about 0.2 degrees between the fuel plate surface and the coolant temperature (compared with 8.4 degrees estimated for the 500 kilowatt case). Since the OSURR Technical Specifications for 10 kilowatt operation apply a maximum core inlet water temperature of less than 95 degrees F., the surface temperature of the fuel plates will be much less than that required to initiate the onset of nucleate boiling, and very much less than the melting point of aluminum.

B.5 Accident Analysis

In the approved Hazards Summary Report for the HEU-fueled OSURR (on file with the NRC), the maximum credible accident was defined as a 1.5% $\Delta k/k$ reactivity insertion. The analysis shows, by comparison with similar experiments in the BORAX program, that the OSURR achieves inherent shutdown through the negative void and temperature coefficients of reactivity feedback mechanisms, with peak fuel cladding temperatures less than those needed to damage the fuel plates to the point of causing release of fission products. For the LEU core discussed in the revised SAR, the accident analysis for 500 kilowatt operation of the OSURR is presented in Chapter 8. For the LEU core described in this document, an accident analysis was performed which compares the LEU core to the HEU core analyzed in the approved HSR for 10 kilowatt operation with HEU. The following paragraphs will discuss this analysis, and show that the accident consequences for an LEU core are less severe than for HEU fuel.

The accident analysis begins by comparing various core parameters. Table B-2 shows the important core physics parameters for both HEU and LEU-fueled cores. Using these data a reactor period can be computed assuming a 1.5% $\Delta k/k$ excess reactivity insertion from the following:

$$\tau = \ell^* / [k_{ex}(1-\beta)-\beta]$$

$$= 9.2 \text{ milliseconds.}$$

Table B-2

Comparison of HEU and LEU Core Characteristics

Core Parameter	Value for HEU Core	Value for LEU Core
Temperature Coefficient in % $\Delta k/k/^{o}C$.	-0.0021	-0.0067[a]
Void Coefficient in % $\Delta k/k/^{o}C$.	-0.28	-0.32[b]
Prompt Neutron Lifetime in Seconds	7.0×10^{-5}	6.6×10^{-5}
Effective Delayed Neutron Fraction	7.4×10^{-3}	7.7×10^{-3}

Notes:

(a) This is the total coefficient and accounts for -0.0055% $\Delta k/k$ for moderator temperature feedback and -0.0012% $\Delta k/k$ for Doppler feedback.

(b) This is the average void coefficient estimated for all LEU cores analyzed.

This compares well with the HEU-fueled core of 9.3 milliseconds. Now, since the LEU-fueled core has a stronger total temperature feedback coefficient of reactivity (-0.0067% $\Delta k/k/^{0}C$. vs. -0.0021% $\Delta k/k/^{0}C$.), and the LEU fuel plates contain more fuel (12.5 grams U-235/0.195 = 64 grams U) than the HEU fuel plates (14 grams U-235/0.93 = 15.1 grams U) and thus a higher heat capacity than HEU fuel plates, and the power peaking factor for the LEU core is much lower (1.64) than that assumed for the HEU core (2.68), the expected maximum temperature increase in the LEU core is lower than that in the HEU core for a reactivity insertion similar to that analyzed in the HSR for the HEU-fueled OSURR. If, conservatively, no credit is taken for energy transfer to the coolant during the transient (since the reactor has a very short stable period), Doppler feedback from fuel heating will be the most important mechanism for negative reactivity insertion. For the LEU core, this effect will be much stronger than it is for the HEU core. The IAEA Guidebook on the safety assessment of reactor core conversions (7th Draft, available from Argonne National Laboratory) shows that the ratio of the Doppler feedback in the LEU fuel plates (identical to those proposed for use in the OSURR) to that in the HEU plates is (-0.00247/-0.00006) = 41 (page 39) and is thus much higher than the corresponding ratio of the total feedback coefficients indicated by the data shown in Table B-2 (i.e., [-0.0067/-0.0021] = 3.2). Thus, one would expect an even more rapid mitigation of the power transient for the LEU-fueled core than for HEU. The consequences of a 1.5% $\Delta k/k$ reactivity insertion for an LEU core are less severe than for an HEU core, since the LEU core is more "tolerant" of a reactivity transient than is a similar HEU core. These conclusions are confirmed by Woodruff [1] using the PARET code in a comparison study of reactivity transients in both HEU and LEU cores.

Thus, it can be concluded that for the same type of transient analyzed for the HEU core (i.e., 1.5% $\Delta k/k$ reactivity insertion) the LEU core will respond in a way that results in less severe conditions being generated within it, and therefore will produce less severe consequences. No fission products will be released. Radiation levels at the top of the reactor pool will be similar to or lower than those predicted for the HEU core.

B.6 References for Attachment B

[1] William L. Woodruff, "A Kinetics and Thermal-Hydraulics Capability for the Analysis of Research Reactors", Nuclear Technology, Vol. 64, pp. 196-206 (reprint) Feb. 1984.