DR-1150

NUS-1277

**PLUTONIUM-238 PRODUCTION IN COMMERCIAL
WATER REACTORS EXPERIMENTAL AND
ANALYTICAL PROGRAM**

Report for the Period July 1, 1973—September 30, 1974

September 1974

NUS Corporation
Dunedin, Florida

MASTER

MASTER

DISTRIBUTION OF THIS DOCUMENT UNLIMITED

UNITED STATES ATOMIC ENERGY COMMISSION
OFFICE OF INFORMATION SERVICES • TECHNICAL INFORMATION CENTER

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

This report has been reproduced directly from the best available copy.

Available from the National Technical Information Service, U. S. Department of Commerce, Springfield, Virginia 22161

Price: Paper Copy \$5.45
Microfiche \$2.25 (domestic)
\$3.75 (foreign)

PLUTONIUM-238 PRODUCTION
IN COMMERCIAL WATER REACTORS
EXPERIMENTAL AND ANALYTICAL PROGRAM

Report for the Period July 1, 1973 — September 30, 1974

Kenneth E. Roach
Stanley E. Turner
Walter Mitchell, III
Marva D. Gurley

September 1974

Prepared for the U.S. Atomic Energy Commission
Under Contract No. AT(11-1)-2322

Southern Nuclear Department
NUS Corporation
Dunedin, Florida

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights

DISTRIBUTION OF THIS DOCUMENT UNLIMITED

26

TABLE OF CONTENTS

	Page
I. INTRODUCTION	1
II. SIMPLIFIED THREE-DIMENSIONAL MODEL	3
III. CORRELATION OF EXPERIMENTAL DATA AND PREDICTIONS	8
A. General	8
B. Isotopic Composition	9
C. Dosimetry Measurements	12
D. Electron Microprobe Analyses	14
E. Chemical Composition	15
F. Revision in Reaction Cross Sections	19
IV. REPROCESSING AND TARGET DILUENTS	22
V. DEMONSTRATION IRRADIATION AND UTILITY SEMINAR	25
A. Demonstration Irradiation	25
B. Utility Seminar	29
VI. ECONOMIC EVALUATION	30
VII. PROGRAM REVIEW	47
VIII. CONTINUING PROGRAM	48

TABLE OF CONTENTS (Continued)

	Page
REFERENCES	51
APPENDIX A: DRAFT OF LETTER OF INVITATION TO UTILITIES TO ATTEND SEMINAR ON PU-238 PRODUCTION	A-1
APPENDIX B: QUESTIONNAIRE ON PU-238 PRODUCTION .	B-1

LIST OF ILLUSTRATIONS

Figure

- 1 Critical boron concentrations as a function of burnup calculated in the NUTRIX-1 model of the Connecticut Yankee reactor.
- 2 Assembly-wise burnup in the NUTRIX-1 model of Connecticut Yankee Core 3 at a core average burnup of 10192 Mwd/mtU.
- 3 Connecticut Yankee Core 3, NUTRIX-1 model — x, y power distribution, BOL.
- 4 Connecticut Yankee Core 3, NUTRIX-1 model — x, y power distribution, 849.3 Mwd/mtU.
- 5 Connecticut Yankee Core 3, NUTRIX-1 model — x, y power distribution, 1698.7 Mwd/mtU.
- 6 Connecticut Yankee Core 3, NUTRIX-1 model — x, y power distribution, 2548.0 Mwd/mtU.
- 7 Connecticut Yankee Core 3, NUTRIX-1 model — x, y power distribution, 3397.3 Mwd/mtU.
- 8 Connecticut Yankee Core 3, NUTRIX-1 model — x, y power distribution, 4246.6 Mwd/mtU.
- 9 Connecticut Yankee Core 3, NUTRIX-1 model — x, y power distribution, 5096 Mwd/mtU.
- 10 Connecticut Yankee Core 3, NUTRIX-1 model — x, y power distribution, 5945.3 Mwd/mtU.
- 11 Connecticut Yankee Core 3, NUTRIX-1 model — x, y power distribution, 6794.6 Mwd/mtU.
- 12 Connecticut Yankee Core 3, NUTRIX-1 model — x, y power distribution, 7644.0 Mwd/mtU.

LIST OF ILLUSTRATIONS (Continued)

Figure

- 13 Connecticut Yankee Core 3, NUTRIX-1 model – x, y power distribution, 8493.3 Mwd/mtU.
- 14 Connecticut Yankee Core 3, NUTRIX-1 model – x, y power distribution, 9342.6 Mwd/mtU.
- 15 Connecticut Yankee Core 3, NUTRIX-1 model – x, y power distribution, 10192 Mwd/mtU.
- 16 Critical boron concentrations as functions of burnup for Connecticut Yankee Core 3, with NUTRIX-1 G value as a parameter.
- 17 Connecticut Yankee Core 3 – relative power distribution from the 3-D PDQ-7 calculation compared with NUTRIX-1 calculations having different transport kernels – BOL.
- 18 Connecticut Yankee Core 3 – relative power distribution from the 3-D PDQ-7 calculation compared with NUTRIX-1 calculations having different transport kernels – 849.3 Mwd/mtU.
- 19 Connecticut Yankee Core 3 – relative power distribution from the 3-D PDQ-7 calculation compared with NUTRIX-1 calculations having different transport kernels – 2,548.0 Mwd/mtU.
- 20 Connecticut Yankee Core 3 – relative power distribution from the 3-D PDQ-7 calculation compared with NUTRIX-1 calculations having different transport kernels – 4,246.6 Mwd/mtU.
- 21 Connecticut Yankee Core 3 – relative power distribution from the 3-D PDQ-7 calculation compared with NUTRIX-1 calculations having different transport kernels – 5,945.3 Mwd/mtU.
- 22 Connecticut Yankee Core 3 – relative power distribution from the 3-D PDQ-7 calculation compared with NUTRIX-1 calculations having different transport kernels – 7,644.0 Mwd/mtU.

LIST OF ILLUSTRATIONS (Continued)

Figure

- 23 Relative power in location J-14 of Core 3 as a function of burnup.
- 24 Critical boron concentration as a function of burnup from PDQ-7 and NUTRIX-1 calculations for the reference reactor.
- 25 Reference PWR — NUTRIX-1 model assembly-wise burnup at an average core burnup of 17,000 Mwd/mtU.
- 26 Reference PWR — PDQ-7 two-dimensional model assembly-wise burnup at an average core burnup of 17,000 Mwd/mtU.
- 27 Reference PWR — NUTRIX-1 model, assembly-wise power distribution, BOL.
- 28 Reference PWR — NUTRIX-1 model, assembly-wise power distribution, 1,000 Mwd/mtU.
- 29 Reference PWR — NUTRIX-1 model, assembly-wise power distribution, 2,000 Mwd/mtU.
- 30 Reference PWR — NUTRIX-1 model, assembly-wise power distribution, 3,000 Mwd/mtU.
- 31 Reference PWR — NUTRIX-1 model, assembly-wise power distribution, 4,000 Mwd/mtU.
- 32 Reference PWR — NUTRIX-1 model, assembly-wise power distribution, 5,000 Mwd/mtU.
- 33 Reference PWR — NUTRIX-1 model, assembly-wise power distribution, 6,000 Mwd/mtU.
- 34 Reference PWR — NUTRIX-1 model, assembly-wise power distribution, 7,000 Mwd/mtU.

LIST OF ILLUSTRATIONS (Continued)

Figure

- 35 Reference PWR – NUTRIX-1 model, assembly-wise power distribution, 8,000 Mwd/mtU.
- 36 Reference PWR – NUTRIX-1 model, assembly-wise power distribution, 9,000 Mwd/mtU.
- 37 Reference PWR – NUTRIX-1 model, assembly-wise power distribution, 10,000 Mwd/mtU.
- 38 Reference PWR – NUTRIX-1 model, assembly-wise power distribution, 11,000 Mwd/mtU.
- 39 Reference PWR – NUTRIX-1 model, assembly-wise power distribution, 12,000 Mwd/mtU.
- 40 Reference PWR – NUTRIX-1 model, assembly-wise power distribution, 13,000 Mwd/mtU.
- 41 Reference PWR – NUTRIX-1 model, assembly-wise power distribution, 14,000 Mwd/mtU.
- 42 Reference PWR – NUTRIX-1 model, assembly-wise power distribution, 15,000 Mwd/mtU.
- 43 Reference PWR – NUTRIX-1 model, assembly-wise power distribution, 16,000 Mwd/mtU.
- 44 Reference PWR – NUTRIX-1 model, assembly-wise power distribution, 17,000 Mwd/mtU.
- 45 Comparison of measured and predicted self-shielding effect, Targets A and B.
- 46 Comparison of measured and predicted self-shielding effect, Target C.

LIST OF ILLUSTRATIONS (Continued)

Figure

- 47 Comparison of measured and predicted self-shielding effect, Target D.
- 48 Error Plot of Pu and Np Composition determined by three techniques of chemical analysis.
- 49 Parametric study of cross-section effects on Pu production, Target Segment A.
- 50 Parametric study of cross-section effects on Pu production, Target Segment B.
- 51 Parametric study of cross-section effects on Pu production, Target Segment C.
- 52 Parametric study of cross-section effects on Pu production, Target Segment D.
- 53 Effect of annual production rates on the cost of Pu-238 production.

I. INTRODUCTION

Under the sponsorship of the U.S. Atomic Energy Commission, the Southern Nuclear Department (formerly Southern Nuclear Engineering, Inc.) of NUS Corporation has conducted an analytical and experimental program to determine the feasibility of producing plutonium-238 in light-water power reactors.

Plutonium-238 is formed by the irradiation of neptunium-237, and the production technique suggested initially by Southern Nuclear Engineering involves the replacement of some of the burnable poison rods in a conventional pressurized water reactor with rods containing neptunium-237.

Analytical work on the concept began in 1969, and a concurrent program to irradiate neptunium-loaded target rods in an operating power reactor commenced in 1970. Target rods were designed by Southern Nuclear Engineering, fabricated by Battelle Columbus Laboratories, and irradiated in Core 3 of the Connecticut Yankee reactor. The irradiation covered the period from approximately June, 1971 until June, 1972. Following irradiation, the target rods were removed from the reactor and shipped to Battelle Columbus Laboratories for analysis. Since that time, the direction of the program has been toward correlating experimental results with theoretical predictions, refining and simplifying calculational methods, investigating target reprocessing techniques, evaluating the economics

of producing plutonium-238 in power reactors, and structuring a large-scale demonstration irradiation.

This report describes NUS Corporation (SND) activities on the program from July 1, 1973 to September 30, 1974. Past reports in the Southern Nuclear series are listed as Refs. 1, 2, 3, and 4. For reasons of continuity, it has been necessary for SND to discuss work done by others, most specifically Battelle Columbus Laboratories; for more information on that work, the reader should consult the appropriate Battelle reports.

II. SIMPLIFIED THREE-DIMENSIONAL MODEL

Three-dimensional physics studies using 8- and 24-plane PDQ-7 calculations were reported in Refs. 3 and 4. During the report period covered here, physics work centered about the development of a simplified three-dimensional model.

Earlier calculations, despite their overall complexity, had utilized epithermal self-shielding factors determined by the use of a simple calculational model. It appeared prudent to verify those factors by recourse to a sophisticated resonance model. Thus, epithermal self-shielding factors as a function of the concentrations of both Pu-238 and Np-237 were calculated with a sophisticated model, using the latest ENDF/B cross section data. A comparison of values showed that the results from the sophisticated model were practically identical with those from the simple model; therefore, it was concluded that the earlier epithermal self-shielding factors were adequate, as was the simple model used to calculate them.

The simplified three-dimensional model developed uses the NUTRIX computer program.⁵ NUTRIX calculates region-dependent power density, fuel exposure, coolant density, and xenon for a three-dimensional (x,y,z) core geometry. The model is based on a modification to one-group transport theory, in which only infinite multiplication factor (k_{∞}) and neutron migration area (m^2) are considered, instead of group-dependent neutron cross sections.

The leakage of neutrons from each region is determined from the transport kernel calculated from the linear dimensions and m^2 for the region. Only mesh points within the active fuel regions are considered, since the reflector has been effectively replaced by albedos at the core-reflector interface.

This program is a revision of the FLARE program,⁶ with the major difference being the addition of an optional critical soluble poison search. NUTRIX optionally computes the critical control rod positions or searches the critical power distribution at the end of life under an optimum power distribution; it can simulate the complete history of an operating reactor including power oscillations due to xenon and refueling or fuel shuffling.

At each node, k_{∞} is calculated using the space dependent effects of control rod positions, moderator density, equilibrium or time-dependent xenon, Doppler coefficient of reactivity, and fuel exposure.

The I-135 and Xe-135 production and decay are calculated under the assumption of constant neutron flux for a given time step. The effects of reactivity loss due to fuel burnup are included by a simple fit of k_{∞} to exposure and exposure- and control-weighted void fractions. The fuel burnup is given in terms of megawatt-days per metric ton of uranium instead of explicit representation of nuclide production and depletion.

The program version used in the analysis of Connecticut Yankee Core 3 was NUTRIX-1, which has a maximum of 12 axial nodes and 14 nodes in the x and y directions. The model was described by 564 nodes (one for each

of the 47 fuel assemblies in the quarter core times 12 axial nodes).

Computer running time for NUTRIX is much shorter than that for point-wise, few-group, diffusion-theory programs such as PDQ-7.⁷

Results of NUTRIX calculations for the Connecticut Yankee reactor are given in Figs. 1 through 22. Calculated critical boron concentrations from an early NUTRIX run are compared with observed values in Fig. 1, while Fig. 2 shows assembly-wise burnup at a core average burnup of 10192 Mwd/mtU. Figures 3 through 15 give x,y power distributions from beginning-of-life to a core average burnup of 10192 Mwd/mtU. The critical boron concentrations and, to a greater degree, the power distributions from early NUTRIX runs show less than good agreement with observed boron concentrations and calculated power distributions from PDQ-7.

In NUTRIX, the diffusion kernel can be adjusted through changes in an arbitrary parameter designated "G." In an attempt to achieve a better match between NUTRIX results and those obtained by measurements and PDQ-7 calculations of the Connecticut Yankee reactor, NUTRIX calculations were run using G values that differed from the value used in the earlier calculations (zero). Critical boron concentrations as a function of burnup are shown in Fig. 16 for G values ranging from 0 to 2. As the figure illustrates, a G of about 0.75 in NUTRIX gives answers that are in good agreement with the observed values of critical boron concentration.

Although an adjustment in the diffusion kernel — through changes in G — gave improved results in the calculation of critical boron concentrations, the power distributions calculated with NUTRIX still showed poor agreement with PDQ-7 calculations. Figures 17 through 22 give the relative power distributions calculated by the two methods, with NUTRIX results shown for G values of 0.5, 1.0, and 2.0.

Measurements from Core 3 of the Connecticut Yankee reactor were received subsequent to the completion of the calculations described above. Power distributions obtained from measured values were compared with the distributions calculated by PDQ-7 and NUTRIX; the conclusion drawn from the comparisons is that there is not a consistent relationship between measured and calculated values. As an example, Fig. 23 shows the measured, PDQ-7, and NUTRIX relative powers in the assembly that contained the neptunium targets (Assembly E-08, Location J-14).

NUTRIX calculations in three dimensions were also performed for the reference design reactor. A comparison between NUTRIX and two-dimensional PDQ-7 calculated reactivity decreases with burnup in that reactor, as indicated by the critical soluble boron concentration, is shown in Fig. 24. Agreement throughout most of the burnup range between the two analytical methods (NUTRIX and PDQ-7) is good, with some deviation appearing toward the end of the cycle lifetime. NUTRIX calculations tend to indicate a slightly higher reactivity near the end of a cycle. An apparent

discrepancy at the beginning of the cycle irradiation is attributable to the fact that NUTRIX calculations include a fictitious initial xenon reactivity loss, whereas the zero-burnup PDQ-7 calculation does not. When the equilibrium xenon concentration is attained in the PDQ-7 calculation, the reactivities calculated by the two analytical techniques are essentially identical — at least to a burnup of 10,000 to 11,000 Mwd/mtU.

Figures 25 and 26 compare the end-of-cycle assembly burnups (core average of 17,000 Mwd/mtU) as calculated by the two techniques. Figures 27 through 44 show the assembly-wise power distribution calculated by NUTRIX for the reference reactor for each 1000 Mwd/mtU burnup step throughout the cycle lifetime. In general, the NUTRIX calculations tend to result in a somewhat "flatter" burnup distribution than the two-dimensional PDQ-7 calculation of the same core. The reasons for this discrepancy in calculated assembly-wise burnup distribution are not readily apparent, although the differences may, in part, be attributable to the axially-varying fuel burnup included in the NUTRIX calculations, but not in a two-dimensional PDQ-7 calculation. Unfortunately, there are no experimental data available on the reference reactor, nor any three-dimensional calculations for the reference reactor, that might permit resolving the apparent discrepancy or improving the reliability of the NUTRIX calculation. NUTRIX would appear to be a valuable tool for scoping-type calculations, but final results should be verified by more sophisticated analytical techniques.

III. CORRELATION OF EXPERIMENTAL DATA AND PREDICTIONS

A. General

Comparisons between experimental data and theoretical predictions are given in the following paragraphs. In these comparisons, experimental data obtained by the Battelle Columbus Laboratories (chemical analyses, electron microprobe analyses and dosimetry results), by the Los Alamos Scientific Laboratory (chemical and isotopic analyses), and by the Savannah River Laboratory (isotopic analysis and Pu-236 alpha counting) were employed in evaluating the precision of the theoretical predictions. Good agreement between measured values and the predictions was obtained for the isotopic composition of the plutonium produced (except for Pu-236), for the thermal and epithermal neutron flux values, and for the relative spatial distribution of plutonium production within the target rods (flux depression or self-shielding effects). However, the spread in results of chemical analyses by three different techniques precluded making a reliable evaluation of the predicted total plutonium yield. The most direct method of chemical analysis, and perhaps the most reliable, was the electron microprobe analyses; these agreed most closely with theoretical predictions. Results of the other two analytical techniques differed by as much as 50% from the predictions and by as much as 40% from the direct electron microprobe analysis, although they were in good agreement with each other — at least for total plutonium content.

The final paragraphs of this section discuss possible modifications to the cross-sections used in the theoretical model, on the basis of the available experimental data, recent publications, and more sophisticated thermal group calculations. Some of these revisions to the basic data improve the comparison in Pu-236/Pu-238 ratio between measured and calculated values.

B. Isotopic Composition

Samples from each of the target segments were analyzed for isotopic distribution by mass-spectrometric techniques at the Savannah River and Los Alamos Laboratories. Results of these analyses, given in Table I, are in very good agreement with theoretical predictions for all cases, except for the Pu-236/Pu-238 ratio. The good agreement observed confirms the analytical model and reaction cross-sections used in the plutonium isotopic chain.

Initial predictions of the Pu-236/Pu-238 ratio were low by a factor of about 2. These predictions had been made using a branching ratio of 0.33 for the path leading to Pu-236 from the $n, 2n$ reaction in Np-237. Subsequent information indicates that a better value for the branching ratio is 0.57. Paulson and Hennelly⁸ obtained good results on calculated Pu-236/Pu-238 using a branching ratio of 0.5, but including a small amount of γ, n reaction yielding additional Pu-236. The γ, n production path for Pu-236 was not included in the calculations reported here.

Table I ISOTOPIC COMPOSITION OF TARGET SAMPLES

		Atom percent of total Pu				
Sample No.		Pu-238	Pu-239	Pu-240	Pu-241	Pu-242
D-1	EXPTL*	92.57	6.54	.665	.210	.014
	EXPTL**	92.91	6.27	.611	.199	.013
	PRED	92.70	6.42	.676	.201	.012
D-2	EXPTL*	91.07	7.65	.913	.345	.027
	EXPTL**	91.39	7.38	.861	.338	.027
	PRED	90.46	8.00	1.075	.428	.033
D-4	EXPTL*	90.65	7.94	.984	.390	.032
	EXPTL**	90.71	7.85	.983	.417	.035
	PRED	89.30	8.74	1.31	.602	.057
C-7	EXPTL*	92.41	6.62	.691	.264	.019
	EXPTL**	91.61	7.20	.839	.334	.025
	PRED	90.23	7.91	1.22	.586	.058
B-9	EXPTL*	93.81	5.52	.485	.176	.010
	EXPTL**	93.18	6.01	.579	.221	.013
	PRED	92.29	6.28	.954	.454	.027
A-11	EXPTL*	94.96	4.57	.340	.115	.012
	EXPTL**	94.95	4.58	.346	.120	<.01
	PRED	94.68	4.34	.643	.30	.028

*Los Alamos results

**Savannah River results

Table II summarizes the experimental values* of the Pu-236/Pu-238 ratio, and shows the earlier calculated values as well as the later revision for branching ratio. With a branching ratio of 0.57, and using revised cross-sections (see Subsection F below), the experimental and calculated ratios are in acceptable agreement, particularly for the targets of lower neptunium loading (D series) which are of greater interest in the present program.

Table II Pu-236/Pu-238 RATIOS

<u>Sample</u>	<u>Pu-236/Pu-238 Ratio, ppm</u>			
	<u>Experimental</u>	<u>Earlier prediction</u>	<u>Revised calculation</u>	
			(1)	(2)
A	20.2 $\pm$.2	8.01	13.57	17.60
B	13.9 $\pm$.2	5.61	9.53	12.59
C	10.6 $\pm$.1	4.66	7.91	10.60
D-4	8.8 $\pm$.3	4.02	6.72	9.17
D-2	8.9 $\pm$.3	4.10	6.88	9.35
D-1	8.9 $\pm$.3	4.40	7.05	9.54

(1) Branching ratio increased to 0.57.

(2) Increase in branching ratio to 0.57 and revised cross-sections for Np-237.

* Independent analyses of the Pu-236/Pu-238 ratios at Mound Laboratories confirmed the Battelle results.

In view of the uncertainty in total plutonium yield, discussed in Subsections E and F below, adjustment of the $n, 2n$ cross section of Np-237 to realize a further improvement in the predicted values of the Pu-236/Pu-238 cannot be reliably made.

C. Dosimetry Measurements

Measurements of the Mn-54 activity — from the $\text{Fe}^{54} (n, p) \text{Mn}^{54}$ reaction — and the Co-60 activity — from the $\text{Co}^{59} (n, \gamma) \text{Co}^{60}$ reaction — were made on samples of the target cladding in order to derive values for the neutron flux in the fast and thermal groups. Observed counting rates (normalized to absolute activities by comparison with NBS standards) were evaluated for the same neutron energy group structure and thermal neutron spectrum used in the theoretical model. The flux values derived from the observed activities are compared in Table III with corresponding fluxes (time-weighted average) in the test assembly derived from the PDQ-7 calculations for the Connecticut Yankee reactor. In the fast group, the observed Mn-54 activities were interpreted using a cross-section of 29 millibarns calculated from ENDF/B data averaged over a typical water-moderated reactor spectrum above 0.625 ev. In the thermal group, the flux was calculated for a thermal group cross-section (18.8 barns) averaged over the thermal group neutron spectrum, and corrected for fast group activation of Co-59.* In addition, the reported thermal group flux was based on initial Co-59 content of 0.04%.

* Using a resonance integral for Co-59 of 70 barns.

Table III COMPARISON OF PREDICTED
AND OBSERVED NEUTRON FLUXES

Sample No.	Position*	Thermal group		Fast group	
		Observed	Predicted	Observed	Predicted
2D-1	3.5 - 4	2.18×10^{13}	2.58×10^{13}	1.68×10^{14}	1.52×10^{14}
2D-2	14.5 - 15	3.19×10^{13}	3.50×10^{13}	2.24×10^{14}	2.04×10^{14}
2D-4	39.5 - 40	3.53×10^{13}	3.95×10^{13}	2.43×10^{14}	2.34×10^{14}
2C-7	6.1 - 6.7	2.71×10^{13}	3.64×10^{13}	2.33×10^{14}	2.34×10^{14}
2B-8	3.5 - 4	3.12×10^{13}	3.20×10^{13}	2.37×10^{14}	2.32×10^{14}
2A-11	5 - 5.5	2.76×10^{13}	2.88×10^{13}	2.36×10^{14}	2.21×10^{14}

* Inches from the top of target segment.

Some uncertainty exists in the initial Co-59 content of the clad tubing and consequently in the experimental thermal neutron flux derived from the Co-60 activity. The tubing manufacturer reported 0.028% Co and BCL reported 0.07-0.075%. To resolve this discrepancy, a sample of tubing was submitted to Pittsburgh Testing Laboratory for analysis. The results, for duplicate analyses including comparison with an NBS standard run at the same time, indicated the clad tubing contains $0.04 \pm 0.01\%$ Co. The "observed" thermal flux values shown in Table III are based upon an initial Co content of 0.04% and should be correct to within 25%.

Within the accuracy of the measurements, the experimental and predicted neutron fluxes are in excellent agreement, both in absolute magnitude and in

relative axial distribution. The predicted fast group fluxes are some 10% lower than those derived from Mn-54 activation, which is within the expected uncertainty of the measurements and of the accuracy with which the $\text{Fe}^{54}(\text{n,p})$ cross-section is known. The relative axial fluxes observed suggest that the axial flux distribution may be slightly "flatter" than the predicted flux shape, although the magnitude of the flattening is within experimental error.

The observed thermal flux, derived from Co-60 activity measurements, is also in good agreement with the predicted fluxes, with the single exception of Sample C. In the γ -scan prior to destructive analysis, there was no indication of a flux depression or spacer at the 2C-7 sampling position. Consequently, it is likely that the low flux value from Sample C results either from a local Co-59 anomaly or to some unrecognized error. Excluding the C sample, the predicted thermal group fluxes are some 10% higher than the experimentally derived fluxes and are well within the uncertainty of the measurements. Relative flux values in the axial direction are also in good agreement and suggest slightly less "flattening" in the measured axial flux distribution than predicted.

D. Electron Microprobe Analyses

Using the electron microprobe technique, and observing the plutonium and neptunium M_{α} emission lines (as well as the Zr L_{α} line), diametral traverses of samples from each of the target segments were made, measuring

directly the spatial distribution of the Pu/Np ratios in the various target samples. These results are shown in Figs. 45, 46, and 47, with a $\pm 10\%$ error bound indicated. The curves drawn through the data in Figs. 45, 46, and 47 are the predicted curves from the self-shielding calculations. Overall agreement between the predicted and observed values is quite good, particularly for the targets of lower neptunium loading. The increasing deviation in shape that may be seen for targets of higher neptunium loading is qualitatively that which would be expected, since the calculational technique uses homogenized resonance cross-sections rather than individual values for each resonance peak. Average reaction rates, however, should be correct. Considerable variation in the resonance integral or in the calculated resonance self-shielding could be tolerated without seriously affecting the agreement between calculated and observed diametral traverses of the Pu/Np ratio. Consequently, these data, although generally confirming calculations of self-shielding effects, cannot be used to derive precise values for the neptunium resonance integral or for the self-shielding factors of individual resonance absorption peaks.

E. Chemical Composition

Chemical analyses of samples from each of the target segments were performed for total plutonium and neptunium content by three different instrumental techniques as follows:

- Electron microprobe analysis. The diametral microprobe data across pellets from each of the target segments were volume weighted to obtain average values for the Pu/Np ratio in the sample. From this ratio data, the plutonium and neptunium content can be inferred analytically, using the theoretical model.
- Battelle Columbus Laboratories. Analyzed for plutonium and neptunium content by dissolving samples of the targets and analyzing an aliquot of the solution by the controlled-potential coulometry technique.
- Los Alamos Scientific Laboratory. Analyzed for plutonium and neptunium content by a sealed tube reflux-leaching operation to prepare solutions of the samples. Plutonium content was then determined mass-spectrometrically using the isotope dilution method, and neptunium was determined by alpha counting after ion-exchange separation.

Results of the chemical analyses by the three techniques are shown in Table IV, together with predicted values for comparison. Since the EMP technique measured the Pu/Np ratio directly, these data are shown in Table V, together with corresponding values from the other analytical techniques.

Table IV RESULTS OF CHEMICAL ANALYSES OF Np TARGET SAMPLES

Sample No.	Location ⁽¹⁾	Wt % Pu				Wt % Np			
		BMI	LASL	Microprobe ⁽²⁾	Pred. ⁽³⁾	BMI	LASL	Microprobe ⁽²⁾	Pred.
D-1	3.5 - 4.0	2.13	2.11		2.54-2.60	10.56	10.8		10.41
D-2	14.5 -15	2.42	2.59		3.19-3.32	9.44	10.5		9.5
D-3	19.75-20.25			3.22	3.35-3.46			10.02	9.69
D-4	39.5 -40	2.90	2.71		3.57-3.69	9.40	10.2		9.17
C-6	3.75- 4.25			5.12				19.25	
C-7	6.1 - 6.7	3.65	3.63	5.42 ⁽⁴⁾	5.41-5.89	15.60	21.4	18.95 ⁽⁴⁾	16.85
B-8,9	4.5 - 6	5.23	4.88	7.85 ⁽⁵⁾	7.71-8.45	32.06	25.0	33.12 ⁽⁵⁾	31.10
A-11	5 - 5.5	5.75	6.16	8.00	8.72-9.37	50.96	58.6	59.67	54.11

(1) Inches from top of target segment.

(2) Inferred from volume weighted average of values obtained during diametral electron microprobe traverse.

(3) Range due to uncertainty in initial Np loading.

(4) Value at C-7 location derived from observed value at C-6 by correcting for effect of spacer — correction factor equals 1.076.

(5) Value measured by EMP is probably high for Pu and low for Np. Only a partial pellet available for EMP analysis.

Table V PLUTONIUM/NEPTUNIUM RATIOS

<u>Target</u>	<u>Pu/Np ratio</u>			
	<u>BMI</u>	<u>LASL</u>	<u>EMP</u>	<u>Predicted</u>
D-1	.202	.195		.244
D-2	.257	.247		.336
D-3			.321	.357
D-4	.307	.266		.389
C-6			.266	
C-7	.234	.170	(.286)	.321
B	.163	.195	.337	.248
A	.113	.105	.134	.161

From the data in Table IV, it is evident that considerable uncertainty exists in the chemical analyses of the irradiated neptunium target material. An error plot, shown in Fig. 48, illustrates graphically the extent of the discrepancies which exist, in a plot intended to reveal trends or the existence of a consistent bias. Chemical analyses for neptunium content fall within a discrepancy band of $\pm 25\%$, with no consistent bias evident, except for the microprobe data. The Battelle analytical results lie within a smaller discrepancy band, as does the neptunium content derived from the microprobe data. The microprobe data appear to have a nearly constant bias, with indicated neptunium content being slightly greater than predicted.

For the plutonium analyses, the Battelle and LASL experimental results are considerably lower than the predicted values of plutonium content: $\sim 25\%$ lower for the samples from the D target, and $\sim 50\%$ lower for samples from the A, B, and C target segments. There is no obvious explanation either for the magnitude of the discrepancy or for the difference in apparent bias between the D series and the other three target segments. Microprobe data for plutonium content show significantly better and more consistent agreement with predicted values, although in all cases the microprobe results consistently show a lower plutonium content than predicted. As indicated in Subsection F below, revised cross-section values decrease the calculated plutonium content, reducing the values below those derived from the microprobe data. However, additional chemical analyses of the target samples are needed to resolve the discrepancy in results from the three analytical techniques and to provide an unambiguous experimental basis for adjusting cross-sections used in the theoretical model.

F. Revision in Reaction Cross Sections

On the basis of the good agreement between measured and calculated isotopic distribution of the plutonium isotopes, the inference can be made that the plutonium chain and cross-section values used in the theoretical model are acceptable. Similarly, the calculational model for neutron flux values and self-shielding factors within a target rod appear to be satisfactory.

The weight of the experimental evidence, however, implies that the theoretical model overpredicted the plutonium yield. To adjust the calculated net yield of plutonium and the Pu-236/Pu-238 ratio, some revision of the Np-237 cross-sections appears necessary. Chemical analysis data indicate that a reduction in calculated plutonium production is needed, although the precision of the analyses is not sufficient to define quantitatively the magnitude of the reduction. The observed Pu-236/Pu-238 ratio also indicates that the net yield of plutonium should be reduced.

The original calculations used a resonance integral for Np-237 of 943 barns including the $1/v$ contribution. Subsequent evaluations^{9,10} indicate that a resonance integral (>0.625 eV) between 807 and 620 barns, including the $1/v$ contribution, would be more appropriate. A resonance integral of 620 barns is obtained by calculation with the reported resonance parameters. Similarly, the original calculations in the thermal group yielded a spectrum weighted effective thermal cross-section of 162 barns (<0.625 eV) from the TEMPEST calculation. More recent calculations, employing a sophisticated 30-group thermal spectrum model and the ENDF/B cross-section library, resulted in a reduction of the effective thermal group microscopic cross-section of Np-237 to 136 barns. Recalculation of the Pu-236/Pu-238 ratio using a resonance integral of 620 barns and an effective thermal cross-section of 130 barns gave better agreement with the observed values as indicated in Table II shown earlier.

Parametric calculations of the plutonium yield were made for several values of the resonance integral (620 and 472 barns) and thermal group microscopic cross-sections (130 and 97 barns). Results of these calculations are shown in Figs. 49, 50, 51, and 52 for the A, B, C, and D target segments, respectively. Also shown on the figures are the experimentally determined plutonium content obtained by all three chemical analysis techniques.

Inspection of Figs. 49, 50 and 51 reveals that no reasonable or consistent adjustment in cross-section values could bring the calculated plutonium content in targets A, B, and C into agreement with the value determined by the Battelle or LASL chemical analysis. For the D target (see Fig. 52), however, such agreement could be arbitrarily forced. However, no other justification for such modified cross-sections exist.

A resonance integral between 620 and 807 barns, together with an effective thermal group cross-section of 136 barns would give acceptable agreement with the total plutonium yield derived from the microprobe data. These values are justified on the basis either of other measurements^{9,10} or by sophisticated thermal spectrum calculations using the best data currently available (ENDF/B). For further calculations, these are the recommended values.

IV. REPROCESSING AND TARGET DILUENTS

It was reported earlier⁴ that the reprocessing of $\text{NpO}_2\text{-ZrO}_2$ target-rod pellets requires an extrapolation of the techniques used for reprocessing reactor fuel elements. Stated simply, $\text{NpO}_2\text{-ZrO}_2$ pellets are difficult to dissolve, with their dissolution requiring processes other than the nitric acid one typically used in reprocessing facilities.

During this report period, the reprocessing problem was discussed in meetings at Savannah River and Oak Ridge, as well as with representatives of Allied-Gulf and General Electric. The early consensus was that it is probably possible to develop a process using hydrofluoric acid for the dissolution of $\text{NpO}_2\text{-ZrO}_2$ targets, but that existing equipment would have to be modified to handle the hydrofluoric acid.

Representatives from AEC, BCL, DuPont, and SND met at Savannah River to determine how the reprocessing of targets similar to those irradiated in the Connecticut Yankee reactor might be accomplished in the existing reprocessing facility at SRL. The conclusion was that the existing facility can accommodate only nitric acid solutions having low concentrations of HF, and that HCl cannot be handled. Reprocessing of $\text{NpO}_2\text{-ZrO}_2$ targets is not possible at SRL without some modifications to the plant.

A meeting held at Oak Ridge, attended by BCL, ORNL, and SND personnel, led to the conclusions that targets using the zirconia diluent

would be difficult to dissolve, but that a process could be developed and a flow sheet worked out for reprocessing. Several techniques were discussed, including fluoride volatilization to remove zirconium, aqua regia dissolution, and leaching. ORNL personnel indicated that they could possibly handle the target rods (approximately 100) from a demonstration irradiation program, as well as targets from a future production program, once the reprocessing procedures were developed and established.

Difficulties in processing targets using ZrO_2 as the diluent led to the consideration of alternate diluents. One alternate is magnesium oxide, and investigation of that material with regard to reactor licensing yielded information that may be significant. In the late 1950's, reactor thermocouple wires contained MgO as an insulator. A problem developed with these wires in a reactor environment (pressurized water) whenever a small leak occurred in the stainless cladding of the thermocouples. Water entered and caused the MgO to swell, rupturing the stainless steel cladding over great lengths. A first solution to the problem was to replace the MgO with ZrO_2 . Apparently, ZrO_2 does not absorb moisture and swell. Although the swelling problem was overcome by using ZrO_2 , fabrication difficulties forced the manufacturers to return to MgO and improve quality control so that leaks do not occur in the cladding. The above information was verified with Thermo Electric Company of Saddle Brook, New Jersey.

It is doubtful that a reactor containing $\text{NpO}_2\text{-MgO}$ targets could be licensed if the thermocouple information were considered. One would have to show experimentally that (1) no leak could possibly occur in the cladding or that (2) there is no swelling problem. Either of these items would be difficult to prove.

Another possible target diluent that could be considered as an alternative to zirconia is cerium oxide (CeO_2). The rare earth oxide CeO_2 would be expected to be more closely analagous in chemical and physical properties to NpO_2 than to ZrO_2 , hence more compatible with existing commercial reprocessing methods and equipment. A brief check of the effect of the neutron absorption cross-section of cerium (larger than that of zirconium) indicated that CeO_2 would be an acceptable diluent from the physics standpoint. Since cerium was not in the SND cross-section library, the check calculation used a fictitious loading of zirconium that closely simulated the macroscopic cross-section for CeO_2 diluent. The calculated k_{eff} (1.00019) with CeO_2 diluent (as simulated by four times the loading of ZrO_2 diluent) was not appreciably different from that calculated with zirconia diluent ($k_{\text{eff}} = 1.00024$); consequently, the use of a CeO_2 diluent would not affect significantly the production of Pu-238.

V. DEMONSTRATION IRRADIATION AND UTILITY SEMINAR

A. Demonstration Irradiation

A demonstration irradiation program, as contrasted to the previous experimental irradiation, was planned during this report period. The demonstration program was characterized by the anticipated irradiation of a relatively large number of target rods — approximately 100. The total neptunium loading was to be approximately 15 kilograms, which would result in the production of around 4 kilograms of plutonium-238.

In order to determine interest on the part of the electric utility industry in participating in the demonstration irradiation program, the organizations listed below were contacted.

Yankee Atomic Electric Company

Florida Power Corporation

Consumers Power Company

Carolina Power & Light Company

Detroit Edison Company

Florida Power & Light Company

Virginia Electric & Power Company

Yankee Atomic Electric Company was the first utility visited because of an earlier expression of interest on its part and because of its past participation in the experimental irradiation program (Connecticut Yankee reactor).

A meeting in Westboro, Massachusetts, and subsequent telephone conversations indicated that a statement of position regarding participation in the demonstration irradiation would be made in a letter from Northeast Utilities. The following excerpt from the Northeast Utilities' letter summarizes the position:

"Subsequent to this Pu 238 production demonstration at Connecticut Yankee, we have been burdened with more restrictive safety criteria related to emergency core cooling and fuel densification. As a result of evaluations to satisfy these new criteria, Connecticut Yankee operational margins necessary for plant maneuverability at full power have been substantially reduced. We are presently in the process of evaluating the effects of the final emergency core cooling criteria which we expect will have further impact on plant maneuverability and availability.

"In light of the above AEC safety concerns, we find it difficult at this time to provide you any assurance of our continued support or assistance in further Pu 238 commercial production in Connecticut Yankee.

"We will reevaluate the desirability of isotope production after the present safety concerns are fully assessed. In this regard, we intend to closely follow and monitor the technical developments in this area....."

Discussions with Yankee Atomic Electric Company indicated that a reactor may be available about 1977 for a demonstration neptunium irradiation.

Meetings with individuals at Florida Power Corporation to discuss the possibility of using Crystal River Unit No. 3 for the demonstration irradiation led to a request for a presentation of the program to that utility's Nuclear Fuel Study Task Force. The presentation covered work on the overall program to date, future plans for the demonstration irradiation, and plans

for a possible production program. The group addressed was receptive to the presentation, but, at the same time, expressed concern about possible further delays on the Crystal River-3 plant due to licensing considerations. A subsequent letter from Florida Power Corporation stated that the utility had decided not to request consideration as a candidate utility for the demonstration program.

In telephone conversations, Consumers Power Company indicated the possibility that the Big Rock Point reactor might be used for the demonstration irradiation. A visit to their offices revealed that three years would be required for a 12,000 Mwd/mtU irradiation. It was also determined that the cost for irradiating the targets in Big Rock Point would be approximately \$550,000. Big Rock is a relatively low-power boiling water reactor, but has the advantage of containing a number of available locations for neptunium targets. The first targets could be inserted in the fall of 1975. The substantive portion of a letter from Consumers Power, outlining its position on neptunium irradiation, is given below.

".....With this letter I would like to confirm to you our continuing interest in the possibility of a cooperative effort which might lead to a demonstration or experimental irradiation in our Big Rock Point Nuclear Plant. As you well know, we have been engaged in a cobalt-60 production program for over eight years. We have had very substantial experience with all facets of target design, AEC licensing, nuclear reactor operations and radioactive materials shipping. Our extensive experience with the cost aspects of any demonstration or experimental irradiation should be especially useful to your project.

"You had asked for some estimates on what such an irradiation might cost. The following list represents what I'd classify as an upper limit cost estimate:

Overheads - @ \$20,000 per year.....	\$ 60,000
Includes plant handling, GO administration, etc.	
Uranium Inventory Costs - @ \$90,000 per year.....	270,000
Carrying Charges on Enriched Uranium Increment	
Burnup Penalty - @ \$8,000 per year.....	24,000
Charge for burning U 'off-the-top'	
Redesign and Licensing - One-time expense.....	110,000
Safety analysis and new enrichment for fuel	
Second Encapsulation Cost - One-time expense.....	90,000
No singly encapsulated sources in reactor	
Total Cost Estimate - 3 years.....	<u>\$554,000</u>

"We would be delighted to discuss the above cost estimates or any other detail with you or your AEC associates."

The cost figures given in the Consumers Power letter are for items whose cost could not be assumed by the utility.

The contacts with Carolina Power & Light Company, Detroit Edison Company, Florida Power & Light Company, and Virginia Electric & Power Company involved, in some instances, detailed discussions of the overall program and the planned demonstration irradiation. Responses to the proposed program indicated interest, but a reluctance to commit facilities at this time because of stringent licensing requirements and uncertainties in reprocessing techniques.

B. Utility Seminar

Early in the report period, steps were taken to plan a seminar for the electric utility industry on the production of Pu-238 in commercial reactors. The purpose of the seminar, which was to be held following a firm commitment for a demonstration irradiation program, was twofold: (1) to inform the utilities and others of progress to date on the existing program, and (2) to receive information from the utilities that would enable the Government to plan a production program to satisfy the needs for plutonium 238.

Inasmuch as the seminar was predicated on the existence of an active demonstration irradiation program, and such a program was not brought to fruition during the report period, the seminar was not held.

VI. ECONOMIC EVALUATION

The economics of Pu-238 production in commercial power reactors were reviewed. The review included assessments of the effects of Pu-238 costs for several of the options available for the management of Pu-238 production. Although the Pu-238 costs discussed below do not differ greatly from those given earlier,⁴ refinements and alternate management options are now included. The principal differences in the revised costs, as shown in Table VI, result from (1) target irradiation in the equilibrium fuel cycle, using increased enrichment to compensate for reactivity loss; (2) revised reprocessing charges to account for reduced throughput; and (3) the inclusion of interest charges representative of the costs of financing that might be encountered in industry operations. Substantial cost reductions in the long term future appear possible, based on discussions with commercial vendors. The potential reductions in the future (without escalation) derive principally from lower Np material costs and reduced target fabrication costs that can reasonably be expected with an established competitive market for Pu-238.

As a result of the economic review and study of potential options (including discussions with Government and private reprocessors), there are several general conclusions that may be made:

Table VI ECONOMIC EVALUATION OF PU-238 PRODUCTION AT 120 kg/yr
(Numbers are costs in \$/gram Pu-238)

	<u>Original estimate</u>		<u>Option 1*</u>	<u>Revised costs (1980)</u>		<u>Potential objective</u>	
	<u>short term (1980)</u>	<u>long term</u>	<u>special reprocessing (1980)</u>	<u>1 cycle irradi.</u>	<u>2 cycle irradi.</u>	<u>1 cycle</u>	<u>2 cycle</u>
Neptunium							
a) Material cost	75.0	75.0	75.0	75.0	47.6	30.0	19
b) Recovery value	53.0	53.0	53.0	57.0**	27.0	22.8	10.8
Net	22	22	22	18	20.6	7.2	8.2
Target fabrication	45	35	45	45	32.7	31.3	22.7
Ship and assembly	1.5	1.5	1.50	1.5	1	1.5	1
Fuel cost penalty	40	30	40	27.6	35.0	27.6	35
Reprocessing	4	4	67 + 3.3 shipping	6-14	6-14	6	6
Ship Pu-238	<u>10</u>	<u>10</u>	<u>—</u>	<u>10</u>	<u>10</u>	<u>0-10</u>	<u>0-10</u>
Subtotal	122.50	102.50	178.80	108.1-116.1	105.3-113.3	73.6-83.6	72.9-82.9
Interest	NI	NI	NI	21.9	24.4	11.3	12.8
TOTAL***	—	—	—	130-138	129.7-137.7	84.9-94.9	85.7-95.7

* Reprocessing in special pilot plant to be constructed at SRL.

** Corrected to 98% recovery in target reprocessing.

*** Corrected for 2% loss of Pu-238 in processing.

- The previous cost analysis⁴ remains substantially correct as an estimate of Pu-238 production costs for the immediately foreseeable future, although minor adjustments result from refinements and from selecting different options. Substantial reductions appear possible in the future, when some of the potential cost reductions may be realized.
- Pu-238 costs are sensitive to the assumed annual production rate. Appreciable cost reductions result with larger annual production rates in a commercial market.
- There appears to be an adequate supply of neptunium potentially available to support a sizable Pu-238 industry.
- Target reprocessing using the fluoride dissolution process in new facilities constructed at a Government site would result in an appreciable cost penalty and, therefore, does not appear to be an economically attractive option. Reprocessing targets in commercial reprocessing plants remains the most economic method of recovering Pu-238 and neptunium.
- To achieve acceptable costs of Pu-238 production, it will be necessary to develop a target diluent material that is compatible with existing commercial reprocessing plants. There are several candidate materials and/or target configurations that may accomplish this objective, including modification of the zirconia diluent investigated to date.

- Increased irradiation does not result in appreciably lower costs of Pu-238 production for the assumptions employed in this review.
- On the basis of a review of the sensitivity of Pu-238 production costs to several of the more significant contributors, reductions in several cost factors in the future appear feasible depending upon the magnitude of the future market for Pu-238. Costs for neptunium target material and fabrication are the principal factors for which appreciable cost reductions are potentially achievable in a future large scale competitive market.
- Interest charges have now been included for the first time. Such costs depend upon the management philosophy ultimately employed for commercial Pu-238 production, and the values shown here represent a reasonable allowance based upon an assumed representative method of financing.

The principal contributors to the cost of commercial production of Pu-238 are discussed in the following paragraphs.

Costs for Pu-238 production are sensitive to the assumed annual production rate, largely because of the reductions in unit fabrication and reprocessing costs with increased throughput. Figure 53 shows the cost of producing Pu-238 for production rates ranging between 50 and 500 kg of Pu-238 per year, for one-cycle target irradiation in the equilibrium

fuel cycle. For 1980, the potential availability of neptunium target material appears adequate to support an industry producing 200 to 500 kg of Pu-238 per year.

At the present time, neptunium costs approximately \$15 per gram, based upon firm quotations to utilities by at least one reprocessor. This value of \$15 per gram represents an upper limit to the cost of neptunium material and has been used in the estimate of Pu-238 production costs for the 1980 period. For a one-year irradiation period, with 20% conversion to Pu-238, the neptunium target material contributes \$75 per gram to the cost of Pu-238 produced. However, the majority (approximately 76%) of the neptunium is recoverable in target reprocessing. Assuming 2% loss of neptunium in the reprocessing phase, the salvage value of neptunium at \$15 per gram would be equivalent to \$57 per gram of Pu-238 produced. The net amount of neptunium burned (and lost in reprocessing) amounts to \$18 per gram of plutonium 238. The overall consumption of neptunium over a one-year irradiation cycle amounts to 1.20 grams for each gram of Pu-238 produced.

For a two-year irradiation cycle, the quantity of neptunium consumed is 1.37 grams for each gram of Pu-238 produced. At \$15 per gram of neptunium, this could contribute ~\$21 per gram to the cost of Pu-238. Irradiation for a third cycle would lead to further increases in the cost component associated with neptunium consumption.

The \$15 per gram cost for neptunium is based on current firm quotations to utilities. Actual costs of recovery of neptunium in one reprocessor's plant (General Electric Company) is expected to be less than \$5 per gram of neptunium, assuming the neptunium is shipped immediately and there are no charges for storage at the reprocessor's site. In a future competitive market, with neptunium demand approximately equal to the potential reprocessing plant output, recovery costs of neptunium might reasonably be expected to approach \$6 per gram (not including escalation). Contacts with other fuel reprocessors revealed comparable costs for neptunium recovery assuming an established market. Material recovery costs in future reprocessing plants can be significantly reduced by including the neptunium recovery line as part of the initial plant construction. Addition of a neptunium recovery process after the plant has been completed would be more difficult and is estimated to be perhaps three times more expensive.

At the present time, NFS has promised utilities recovery of neptunium at \$8 per gram. However, since the recovery lines are not now installed in the West Valley plant, the actual costs of neptunium recovery to meet these commitments would probably exceed the \$8 per gram figure. Alternative methods of neptunium recovery are available, although the percent of neptunium recovered would be considerably lower. For example, NFS

estimates that if 50% recovery of neptunium were acceptable, the cost of such a recovery process would amount to approximately \$5 per gram of neptunium. If the amount of neptunium thus recovered were sufficient to meet the predicted demand, this process affords an immediately-available method for reducing the cost contribution of neptunium to overall plutonium-238 production costs.

Based upon discussions with the various reprocessors, a future projected cost of neptunium of \$6 per gram would appear to be a reasonable objective (without escalation). Low-cost neptunium, however, clearly depends upon the establishment of a firm market for Pu-238 and a corresponding demand for neptunium.

Estimates of the cost of fabricating neptunium target rods are based upon the established costs for UO₂ rods and projected costs for PuO₂-UO₂ (mixed oxide) fuel rods. Based upon a cost of fabrication for UO₂ of \$200 per rod in a one-metric-ton-per-day throughput fabrication facility, the fabrication costs for neptunium rods can be estimated by scaling upward with a three-tenths power factor scaling rule. A common industry practice is to assign a 20% increase to fabrication cost for the complexity of handling and processing PuO₂-bearing fuel rods. This factor, plus an additional factor of 1.2 to allow for the possibility of even greater penalties incurred through handling neptunium, was applied to the basic cost of fabricating UO₂.

The net cost of neptunium target rod fabrication then becomes the following:

$$C = 1.2 \times 1.2 \times \$200 \left(\frac{T}{T_o} \right)^{-0.3}$$

where T is the annual throughput rate and T_o is the throughput of UO_2 in the reference one-metric-ton fabrication facility.

At the reference production rate (120 kg Pu-238 per year), fabrication of 600 kg neptunium targets per year would be required at a cost of about \$1800 per rod as calculated from the equation above. In the reference case, 20% conversion, this fabrication cost contributes \$45 to the cost of each gram of Pu-238 produced. For a two-year irradiation cycle, only 1905 rods per year would be needed at a cost of about \$2000 per rod to yield 120 kg Pu-238 per year. Despite the increase in the unit fabrication cost per rod, the higher yield of Pu-238 (31.5% conversion) reduces the fabrication contribution to \$32.74 per gram of Pu-238. In the future, as experience is gained in processing and fabricating neptunium target rods, it would be reasonable to expect that some of the penalty assigned relative to UO_2 fabrication costs could be eliminated. If the entire penalty factor (1.44) for fabricating NpO_2 rods compared to UO_2 rods could be eliminated — either by experience or by improved fabrication techniques — the fabrication contribution would amount to \$31.25/gm of Pu-238 produced for a one-year irradiation cycle and \$22.74/gm of Pu-238 for a two-year cycle. This latter cost would appear to be a reasonable cost objective

for the fabrication contribution projected into the long term future when a substantial market for Pu-238 has been established.

Because of the scaling law applied, the contribution from fabrication costs is sensitive to the assumed annual production weight of Pu-238.

Table VII shows the contribution of fabrication costs to Pu-238 production costs for various annual production rates.

Table VII FABRICATION COSTS OF Np TARGET RODS

Annual production rate, kg Pu-238/yr	Target fabrication cost, \$/gm Pu-238		
	1-cycle irradi.	2-cycle irradi.	Projected future cost
50	58.5	42.6	41
75	51.8	37.7	36
100	47.5	34.6	33
120	45	32.7	31
150	42.1	30.6	29
200	38.6	29.3	27
500	29.3	21.3	20

Two optional procedures for producing plutonium 238 have been considered. The first was irradiation in the initial fuel cycle of a power reactor, sacrificing fuel burnup to achieve the desired Pu-238 production. The second alternative was irradiation of neptunium in the equilibrium

fuel cycle, with U-235 feed enrichment increased to compensate for the reactivity loss. Prior estimates of the fuel cost penalty of irradiating neptunium targets were based upon the reactivity loss, and hence the reduced fuel burnup. For continuous production of Pu-238, it is probably more reasonable to assume that the U-235 enrichment would be increased to compensate for the neutrons absorbed in the target rods. For the reference case, zero-dimensional physics calculations indicate that the fuel enrichment should be increased from 3.2% to 3.44% to compensate for the presence of the neptunium target rods. On the basis of this enrichment increase, and assuming \$30 per separative work unit and \$8 per pound U_3O_8 cost, the higher feed enrichment would increase the fuel cost (burnup charge plus initial enrichment minus salvage value) by about \$341,000 per year. In addition, interest charges on the higher-valued fuel (assuming an annual interest rate of 12%) would contribute an additional \$321,000 per year in the reference reactor design, producing approximately 24 kg of Pu-238 in a one-year irradiation cycle. The net contribution to Pu-238 production costs from the fuel cost penalty thus amounts to \$27.60 per gram of Pu-238. The actual fuel cost penalty incurred in any given reactor will differ from this value because the interest rates and other economic factors are unique to the utility owning and operating the reactor.

The fuel cost penalty incurred in irradiating neptunium target material for the production of Pu-238 derives from factors that are not directly

related to the target irradiation, such as ore costs, separative work charges, and interest rates. Thus, in a Pu-238 production program there are no recognized options available to reduce the cost of neutrons, other than to insure that installation of target rods does not cause additional penalties by unacceptable perturbations in power operations (e.g., power peaking factors, etc.).

Perhaps the greatest uncertainty in estimating the cost for Pu-238 production results from the uncertainty in costs of reprocessing target rods to recover neptunium and Pu-238. Earlier estimates of the reprocessing costs (\$4/gm of Pu-238) assumed that the targets could be reprocessed in an annual campaign in commercial reprocessing plants. This estimate assumed that, with allowances for reduced throughput and plant cleanup time, a period of 12 days would be adequate to process target materials containing 120 kg of Pu-238. Subsequently, in discussions with various fuel reprocessors, the difficulty of processing target rods with zirconia diluent became evident, largely as a result of the low dissolution rates of zirconia in the plant head-end and the possible requirement for a fluoride dissolution process.

One potential alternative studied was the possibility of constructing a small reprocessing plant using a fluoride dissolution technique in one of the canyons at SRL. Results of the economic evaluation of this process

yielded significantly higher target reprocessing costs — amounting to approximately \$67/gm of Pu-238 compared to \$4/gm in earlier estimates. Consequently, the principal conclusion of this study is that reprocessing target rods in a specially-constructed plant is not an economically-attractive technique. In addition to the high cost of a special processing plant for zirconia-diluent targets, the opinion of various reprocessors inevitably leads to the conclusion that a target diluent material other than zirconia is needed: one that would be compatible with the head-end dissolution processes now employed in commercial reprocessing plants. Several candidate materials can be suggested, including alumina, magnesia, ceria, and, perhaps, unusual materials such as bismuth oxide. The dissolution rate of alumina (ceramic) is not known, and the use of ceria would involve an additional parasitic neutron absorption penalty. While magnesia may not be stable under postulated accident conditions (cladding rupture), the addition of magnesia to zirconia could possibly increase the dissolution rate of zirconia in conventional reprocessing schemes, while maintaining the magnesia content sufficiently low to avoid unacceptable rapid chemical reaction with hot water under accident conditions. Other diluent materials remain to be studied, but the economic incentive to develop a diluent compatible with commercial reprocessing plants is sufficient to justify an investigation.

Assuming a diluent compatible with commercial reprocessing plants can be developed, costs for target reprocessing have been re-evaluated. Under this assumption, commercial reprocessors would be able to accommodate the neptunium target material after irradiation, although the allowable throughput would be less than that for conventional reactor fuel. The reduced throughput permitted is a result of three factors: (1) the higher neutron radiation from Pu-238; (2) the greater heating rate due to the higher alpha specific activity of Pu-238; and (3) the higher off-gassing from Pu-238 decay. In most reprocessing plants, the concrete shielding provided to protect against gamma radiation also provides considerable shielding from the neutron radiation of Pu-238. Although the neutron radiation from Pu-238 is some twenty to forty times that from the plutonium in ordinary reactor fuel, the neutron dose is not the criterion for design of reprocessing facility shielding, and the permitted Pu-238 processing rate would not be directly proportional to the higher neutron dose.

The greater heating load and off-gassing from Pu-238 would tend to limit the quantity of Pu-238 that could be processed at any one time. The small fission-product inventory associated with neptunium target rods, however, would tend to compensate for this effect, since the fission product heating load would be less. Without a detailed and expensive evaluation of

the capability of a commercial reprocessing plant to handle neptunium-plutonium 238 target material, the maximum processing rate can only be approximated. NFS has estimated that a one-mtU conventional reprocessing plant would be able to safely accommodate approximately 10 kg of Pu-238 per day without additional shielding. In addition, estimates of time required for plant cleanup in a once-a-year target reprocessing campaign range from two days (GE Morris plant) to as much as ten or twenty days (NFS, West Valley). At \$40,000 per day reprocessing charge, estimates of the contribution of target reprocessing costs for various annual production rates of Pu-238 are given in Table VIII for several combinations of processing rates and cleanup times. Because of the uncertainty in reprocessing charges, the contribution to Pu-238 costs at the rate of the reference production rate of 120 kg per year are estimated to range from \$6 to \$14/gm of Pu-238 produced.

In commercial Pu-238 production managed by the utility industry, interest charges would accrue on the capital invested by the utility in certain stages of the production plan. Under this method of financing, interest charges during fabrication, irradiation, cooling and reprocessing would be applicable. For the purposes of the present estimates, it has been assumed that interest charges on neptunium target material are included in the target fabrication costs. Interest charges on the capital

Table VIII COST CONTRIBUTION FROM REPROCESSING

Production rate kg Pu-238/yr	<u>Minimum</u> 10 kg/day processing 6 day cleanup	<u>Nominal</u> 10 kg/day processing 8 day cleanup	<u>Maximum</u> 5 kg/day processing 18 day cleanup
50	8.8	10.4	22.4
75	7.2	8.3	17.6
100	6.4	7.2	15.2
120	6.0	6.7	14.0
150	5.6	6.1	12.8
200	5.2	5.6	11.6
500	4.8	4.6	9.44

invested in the neptunium material and for the target fabrication were computed on the basis of an assumed 12% interest rate applicable during the in-reactor irradiation period and for about six months thereafter.

The contribution of interest charges to Pu-238 production costs on the basis of the assumptions discussed here are listed in Table VI.

The General Electric Company has projected that 2000 to 3000 kg of neptunium 237 are potentially available from commercial power reactor fuels by the year of 1980, and that the minimum annual yield could amount to some 550 kg. The GE Morris plant anticipates a production rate of nearly 100 kg per year by 1980. NFS anticipates that it would be capable

of producing up to 490 kg of neptunium per year in the current West Valley plant. To achieve low neptunium recovery costs (about \$5 to \$6/gm), only 50 or 60% of the neptunium would be recovered — or approximately 250 kg per year by NFS.

On the basis of an annual irradiation cycle, estimates have been made of the total quantity of neptunium 237 required in irradiation and the annual neptunium replacement rate required for various annual production rates of Pu-238. These values are given in Table IX. Since approximately 76% of the neptunium is recoverable, the annual replacement needs for neptunium are considerably less than the inventory in irradiation. For example, at a Pu-238 production rate of 120 kg per year, 600 kg of neptunium would be loaded for irradiation. At the end of the irradiation, however, some 456 kg of neptunium would be recovered, necessitating the replacement of 144 kg of neptunium to continue the cycle for the next irradiation. For a 120 kg annual production rate, an additional neptunium inventory would be needed out-of-pile in cooling, reprocessing, and refabrication. If recovery and refabrication could be accomplished in one year, the out-of-pile inventory would be equal to that under irradiation at any point in time. On the basis of these figures, it would appear that, by the year of 1980, annual production rates of Pu-238 ranging between 200 and 500 kg per year would be possible depending upon the availability of neptunium at that time.

Table IX ANNUAL REQUIREMENTS FOR Np TARGET MATERIAL

<u>Production rate</u> <u>kg Pu-238/yr</u>	<u>kg Np</u> <u>in-pile</u>	<u>Total inventory</u> <u>kg Np</u>	<u>Annual supply of</u> <u>Np required*</u>
50	250	500	60
75	375	750	90
100	500	1000	120
120	600	1200	144
150	750	1500	180
200	1000	2000	240
500	2500	5000	600

* Equals kg Np consumed annually.

VII. PROGRAM REVIEW

At a program-review meeting in the Space Nuclear Systems offices in Germantown on May 30, 1974, SNS personnel expressed concern that the program may be delayed principally because of (1) the potential impact of licensing requirements on the irradiation of neptunium target material; (2) delays in operation of commercial nuclear power plants now being encountered by most utilities as a result of the Regulatory requirements and the licensing review processes; and (3) an apparent reluctance of utilities to make a firm commitment until Regulatory requirements are clearly defined and until a market for Pu-238 can be established with some degree of confidence. These concerns on the part of SNS can be answered only by (1) continuation of the program as proposed and (2) positive action on the part of SNS to define more clearly projected Government requirements for Pu-238, in order to give some assurance of the potential marketability of commercially-produced Pu-238 and to establish a base requirement for the recovery of neptunium 237.

VIII. CONTINUING PROGRAM

A program providing for the orderly continuation of the study of Pu-238 production in commercial nuclear power plants should place principal emphasis upon resolution of the two major outstanding problems revealed by the previous studies. These are (1) careful review and evaluation of licensing problems that may arise in assuring AEC Regulatory that irradiation of neptunium target material in power reactors does not introduce any new or significant safety problems that might constitute a hazard to the health and safety of the public; and (2) detailed investigation of the problem of Pu-238 recovery from irradiated target material, including reprocessing techniques and the availability of reprocessing capability adequate to meet the projected need. Secondary areas that must also be considered include (1) resolution of the discrepancy in chemical analyses of the irradiated target materials revealed in the previous program, and appropriate adjustment, as needed, of the analytical model for predicting Pu-238 production; (2) engineering design and physics analyses in support of the licensing review discussed above; (3) continued investigation of target fabrication techniques, including improvements in $\text{NpO}_2\text{-ZrO}_2$ pellet preparation, as well as techniques applicable for alternate diluent materials that may be considered; (4) test irradiations with several alternate diluent materials (compatible with reprocessing and recovery of Pu-238/Np-237)

to demonstrate feasibility and mechanical stability of such alternate diluents; and (5) continued liaison with industry (utilities and fuel reprocessors) to develop interest in the program and to resolve any residual concerns that industry may have.

A logical objective for the program outlined above would be a demonstration irradiation of perhaps 100 target rods in one of the larger operating power reactors. To do so would require direct assistance to the utility involved in safety analyses and licensing applications. In addition, detailed engineering design and coordination with a selected fuel fabricator and reprocessor would be needed. A considerable portion of the preparatory work for the demonstration irradiation can, and should, be performed concurrent with the R&D programs discussed above. Negotiations to assure the timely availability of neptunium target material should commence well in advance of the demonstration irradiation program.

If initiated promptly, the total R&D program, culminating in the demonstration irradiation, would require approximately five years and would result in the availability of several kilograms of Pu-238, produced commercially, in approximately seven years. Long-range planning should include (1) consideration of neptunium availability and early commitments that may be necessary to assure the adequacy of future supply, and (2) consideration of facilities (perhaps in conjunction with existing or projected reprocessing facilities) for neptunium recovery, target fabrication,

Pu-238/Np-237 reprocessing, and Pu-238 encapsulation on a single site, sharing such support facilities as may be economically desirable.

REFERENCES

1. K. E. Roach, S. E. Turner, and W. A. Maxwell, The Production of ^{238}Pu in Commercial Light Water Reactors, USAEC Report SNE-53, Southern Nuclear Engineering, Inc., December 1969.
2. K. E. Roach, W. L. Mullett, G. M. Brown, W. A. Maxwell, S. E. Turner, and W. Mitchell, III, Plutonium-238 Production in Commercial Water Reactors Experimental and Analytical Program, USAEC Report SNE-101, Southern Nuclear Engineering, Inc., February 1972.
3. K. E. Roach, W. L. Mullett, G. M. Brown, W. A. Maxwell, S. E. Turner, and W. Mitchell, III, Plutonium-238 Production in Commercial Water Reactors Experimental and Analytical Program, USAEC Report SNE-119, Southern Nuclear Engineering, Inc., July 1972.
4. K. E. Roach, S. E. Turner, W. L. Mullett, W. Mitchell, III, and M. D. Gurley, Plutonium-238 Production in Commercial Water Reactors Experimental and Analytical Program, Annual Report for Fiscal Year 1973, USAEC Report SNE-129, Southern Nuclear Engineering, Inc., July 1973.
5. Y. S. Kim, NUTRIX — A Digital Computer Program for Three-Dimensional Analysis of Time-Dependent Operating Reactor, Report NUS-657, Rev. 1, NUS Corporation, March 1972.
6. D. L. Delp et al., FLARE — A Three-Dimensional Boiling Water Reactor Simulator, USAEC Report GEAP-4598, General Electric Company, July 16, 1964.
7. W. R. Cadwell, PDQ-7 Reference Manual, USAEC Report WAPD-TM-678, Bettis Atomic Power Laboratory, Westinghouse Electric Corporation, January 1967.
8. C. K. Paulson and E. J. Hennelly, Cross-section Measurement of Plutonium-236 Formation in Plutonium-238 by ^{237}Np (n,2n) Reactions, Nuclear Science and Engineering, 55(1): 24-27 (September 1974).

REFERENCES (Continued)

9. E. Hellstrand, J. Phelps, and C. Sastre, Studies of the Capture Cross Section of ^{237}Np and ^{241}Am in Different Reactor Spectra, USAEC Report BNL-50242 (T-577), Brookhaven National Laboratory, June 1970.
10. R. P. Schuman and J. R. Berreth, Resonance Integral Measurements, USAEC Report IN-1296, Idaho Nuclear Corporation, May 1969.

Fig. 1 Critical boron concentrations as a function of burnup calculated in the NUTRIX-1 model of the Connecticut Yankee reactor.

9426	10814	9780	11245	10154	10882	9513	8728
10817	10276	11458	10743	11547	10377	11160	7847
9792	11462	11447	10446	11280	9503	9669	
11287	10751	10359	9942	11138	11020	8025	
10296	11617	11286	11142	10213	8220		
10926	10499	9533	11032	8225			
9460	11113	9667	8030				
8400	7602						

Fig. 2 Assembly-wise burnup in the NUTRIX-1 model of Connecticut Yankee Core 3 at a core average burnup of 10192 Mwd/mtU.

.8980	1.044	.9427	1.100	.9917	1.077	.9409	.8788
1.045	.9888	1.121	1.042	1.142	1.018	1.123	.7826
.9436	1.121	1.119	1.009	1.114	.9309	.9663	
1.104	1.042	1.003	.9602	1.101	1.103	.7964	
1.002	1.145	1.112	1.100	1.006	.8114		
1.072	1.021	.9278	1.099	.8101			
.9150	1.098	.9547	.7906				
.8072	.7332						

Fig. 3 Connecticut Yankee Core 3, NUTRIX-1 model -
x, y power distribution, BOL.

.9103	1.056	.9515	1.105	.9928	1.072	.9341	.8672
1.056	1.000	1.128	1.049	1.140	1.016	1.111	.7746
.9525	1.128	1.125	1.015	1.111	.9281	.9578	
1.109	1.049	1.008	.9652	1.098	1.095	.7910	
1.005	1.146	1.110	1.098	1.004	.8084		
1.071	1.023	.9275	1.093	.8078			
.9161	1.094	.9508	.7878				
.8090	.7342						

Fig. 4 Connecticut Yankee Core 3, NUTRIX-1 model -
x, y power distribution, 849.3 Mwd/mtU.

.9188	1.062	.9572	1.107	.9936	1.068	.9298	.8591
1.063	1.007	1.131	1.053	1.138	1.015	1.102	.7692
.9585	1.132	1.128	1.020	1.108	.9266	.9516	
1.112	1.054	1.012	.9691	1.096	1.088	.7873	
1.008	1.145	1.108	1.096	1.003	.8063		
1.072	1.025	.9284	1.089	.8064			
.9184	1.092	.9487	.7864				
.8120	.7364						

Fig. 5 Connecticut Yankee Core 3, NUTRIX-1 model -
x, y power distribution, 1698.7 Mwd/mtU.

.9245	1.066	.9609	1.109	.9944	1.066	.9277	.8539
1.066	1.012	1.132	1.056	1.137	1.015	1.096	.7660
.9623	1.133	1.129	1.024	1.107	.9264	.9476	
1.113	1.057	1.014	.9722	1.094	1.084	.7850	
1.010	1.145	1.108	1.095	1.002	.8049		
1.072	1.028	.9298	1.085	.8056			
.9212	1.090	.9474	.7857				
.8154	.7389						

Fig. 6 Connecticut Yankee Core 3, NUTRIX-1 model -
x, y power distribution, 2548.0 Mwd/mtU.

.9281	1.068	.9630	1.108	.9953	1.065	.9273	.8510
1.068	1.014	1.132	1.058	1.135	1.015	1.092	.7647
.9644	1.132	1.129	1.026	1.106	.9272	.9453	
1.113	1.059	1.016	.9747	1.093	1.080	.7839	
1.011	1.143	1.107	1.094	1.001	.8042		
1.072	1.029	.9315	1.082	.8051			
.9240	1.090	.9467	.7855				
.8189	.7416						

Fig. 7 Connecticut Yankee Core 3, NUTRIX-1 model –
x, y power distribution, 3397.3 Mwd/mtU.

.9302	1.068	.9639	1.107	.9962	1.064	.9281	.8498
1.068	1.015	1.130	1.058	1.134	1.016	1.090	.7646
.9653	1.130	1.128	1.028	1.105	.9288	.9441	
1.112	1.059	1.018	.9767	1.092	1.078	.7837	
1.012	1.142	1.106	1.093	1.000	.8041		
1.073	1.031	.9334	1.080	.8051			
.9269	1.089	.9465	.7857				
.8224	.7444						

Fig. 8 Connecticut Yankee Core 3, NUTRIX-1 model —
x, y power distribution, 4246.6 Mwd/mtU.

.9312	1.066	.9641	1.106	.9970	1.065	.9297	.8497
1.067	1.014	1.127	1.058	1.132	1.017	1.088	.7654
.9653	1.128	1.126	1.029	1.105	.9308	.9438	
1.110	1.059	1.019	.9784	1.091	1.076	.7841	
1.012	1.140	1.106	1.092	1.000	.8043		
1.073	1.032	.9356	1.079	.8053			
.9299	1.089	.9467	.7863				
.8260	.7472						

Fig. 9 Connecticut Yankee Core 3, NUTRIX-1 model -
x, y power distribution, 5096 Mwd/mtU.

.9315	1.065	.9637	1.104	.9978	1.066	.9318	.8505
1.065	1.013	1.124	1.058	1.130	1.019	1.088	.7668
.9648	1.125	1.123	1.030	1.105	.9331	.9441	
1.108	1.058	1.020	.9797	1.090	1.075	.7850	
1.013	1.138	1.106	1.091	1.000	.8049		
1.073	1.033	.9378	1.078	.8058			
.9328	1.089	.9471	.7872				
.8296	.7501						

Fig. 10 Connecticut Yankee Core 3, NUTRIX-1 model -
x, y power distribution, 5945.3 Mwd/mtU.

.9314	1.062	.9630	1.102	.9985	1.066	.9343	.8518
1.063	1.011	1.121	1.056	1.129	1.020	1.088	.7686
.9641	1.121	1.121	1.030	1.105	.9356	.9448	
1.106	1.057	1.021	.9809	1.090	1.074	.7861	
1.013	1.136	1.106	1.090	1.001	.8057		
1.073	1.034	.9401	1.077	.8066			
.9355	1.089	.9476	.7882				
.8330	.7528						

Fig. 11 Connecticut Yankee Core 3, NUTRIX-1 model —
x, y power distribution, 6794.6 Mwd/mtU.

.9312	1.060	.9622	1.100	.9991	1.067	.9368	.8534
1.060	1.009	1.117	1.055	1.127	1.021	1.088	.7706
.9632	1.118	1.118	1.030	1.105	.9381	.9457	
1.103	1.056	1.021	.9819	1.090	1.074	.7874	
1.013	1.134	1.106	1.090	1.001	.8068		
1.072	1.035	.9423	1.076	.8075			
.9379	1.089	.9482	.7893				
.8361	.7554						

Fig. 12 Connecticut Yankee Core 3, NUTRIX-1 model -
x, y power distribution, 7644.0 Mwd/mtU.

.9317	1.059	.9621	1.098	.9997	1.068	.9388	.8547
1.059	1.008	1.115	1.054	1.126	1.022	1.087	.7721
.9630	1.115	1.117	1.030	1.105	.9404	.9462	
1.101	1.055	1.022	.9830	1.090	1.074	.7886	
1.013	1.132	1.106	1.090	1.002	.8078		
1.071	1.035	.9441	1.075	.8084			
.9397	1.088	.9484	.7901				
.8386	.7573						

Fig. 13 Connecticut Yankee Core 3, NUTRIX-1 model -
x, y power distribution, 8493.3 Mwd/mtU.

.9313	1.056	.9612	1.095	.9999	1.068	.9415	.8569
1.056	1.006	1.112	1.052	1.124	1.023	1.088	.7744
.9619	1.112	1.115	1.030	1.105	.9428	.9474	
1.098	1.053	1.022	.9840	1.090	1.074	.7903	
1.012	1.130	1.106	1.090	1.004	.8094		
1.070	1.034	.9460	1.075	.8099			
.9416	1.088	.9490	.7914				
.8413	.7596						

Fig. 14 Connecticut Yankee Core 3, NUTRIX-1 model – x, y power distribution, 9342.6 Mwd/mtU.

.9318	1.055	.9609	1.094	1.000	1.069	.9436	.8585
1.055	1.005	1.109	1.051	1.122	1.023	1.087	.7761
.9615	1.110	1.114	1.029	1.105	.9449	.9481	
1.096	1.051	1.022	.9851	1.091	1.074	.7917	
1.011	1.127	1.105	1.091	1.005	.8111		
1.068	1.034	.9475	1.075	.8114			
.9428	1.087	.9491	.7924				
.8433	.7612						

Fig. 15 Connecticut Yankee Core 3, NUTRIX-1 model —
x, y power distribution, 10192 Mwd/mtU.

Fig. 16 Critical boron concentrations as functions of burnup for Connecticut Yankee Core 3, with NUTRIX-1 G value as a parameter.

.5788	.7444	.6523	.8638	.7990	.9537	.8498	.6837
.7339	.8790	.8010	.9923	.9291	1.093	1.018	1.029
.5316	.6641	.6192	.8412	.8402	1.113	1.135	1.274
.1226	.1835	.2042	.4117	.5599	1.150	1.594	2.474
.7484	.7099	.8535	.8360	.9566	.9270	1.129	.8979
.8788	.8315	.9828	.9383	1.101	1.034	1.261	.9294
.6630	.6295	.7934	.7945	1.034	1.054	1.478	1.172
.1788	.1784	.3033	.3934	.7820	1.101	2.459	2.414
.6647	.8641	.8864	.8712	1.024	.9361	1.180	
.8003	.9817	.9994	.9232	1.092	.9536	1.087	
.6152	.7892	.8294	.8014	1.050	.9816	1.272	
.1862	.2838	.3481	.4288	.8348	1.021	2.042	
.9010	.8646	.8832	.9102	1.127	1.279	1.097	
.9912	.9348	.9148	.9085	1.121	1.204	.9088	
.8305	.7829	.7879	.8303	1.136	1.345	1.078	
.3539	.3378	.3884	.5321	1.060	1.768	1.734	
.8799	1.023	1.067	1.150	1.223	1.237		
.9326	1.096	1.084	1.116	1.059	.8972		
.8277	1.011	1.028	1.123	1.119	1.014		
.4493	.6300	.7112	.9660	1.131	1.319		
.8799	1.033	1.003	1.334	1.263			
1.059	1.024	.9414	1.192	.8928			
1.027	1.014	.9481	1.313	1.001			
.7641	.8232	.8146	1.509	1.202			
.9723	1.283	1.302	1.169				
.9671	1.029	1.059	.8934				
1.030	1.361	1.204	1.037				
1.054	1.696	1.532	1.388				
.9322	1.018	— PDQ-7					
.9261	.8470	— 0.5 - G					
1.100	1.014	— 1.0					
1.597	1.504	— 2.0					

Fig. 17 Connecticut Yankee Core 3 — relative power distribution from the 3-D PDQ-7 calculation compared with NUTRIX-1 calculations having different transport kernels — BOL.

.4997	.6441	.5685	.7739	.7452	.9530	.9168	.8321
.7731	.9207	.8320	1.014	.9346	1.078	.9901	.9869
.6071	.7503	.6824	.8915	.8525	1.076	1.058	1.153
.2170	.3064	.2877	.4757	.5072	.8314	.9666	1.310
.6448	.6089	.7425	.7440	.8970	.9257	1.264	1.132
.9208	.8698	1.016	.9601	1.103	1.023	1.221	.8952
.7502	.7067	.8674	.8409	1.044	1.023	1.371	1.068
.3124	.2929	.4301	.4506	.7212	.8107	1.500	1.316
.5709	.7442	.7718	.7803	.9629	.9394	1.385	
.8323	1.015	1.027	.9410	1.089	.9417	1.057	
.6822	.8667	.8949	.8395	1.052	.9528	1.194	
.3097	.4528	.4964	.5057	.8164	.8320	1.409	
.7839	.7504	.7790	.8275	1.072	1.335	1.351	
1.016	.9589	.9321	.9183	1.114	1.181	.8874	
.8924	.8386	.8299	.8525	1.128	1.295	1.022	
.5523	.5112	.5378	.6377	1.081	1.550	1.335	
.7867	.9218	.9734	1.077	1.211	1.398		
.9443	1.104	1.086	1.112	1.050	.8844		
.8604	1.043	1.046	1.124	1.106	.9871		
.6412	.8812	.9438	1.183	1.258	1.290		
.8427	.9754	.9614	1.349	1.404			
1.058	1.024	.9370	1.176	.8826			
1.032	1.019	.9449	1.288	.9842			
1.001	1.065	1.028	1.820	1.418			
.9656	1.333	1.436	1.376				
.9588	1.190	1.043	.8798				
1.012	1.324	1.172	1.011				
1.287	2.024	1.840	1.669				
1.019	1.174	PDQ-7					
.9143	.8355	0.5 - G					
1.067	.9830	1.0					
1.866	1.761	2.0					

Fig. 18 Connecticut Yankee Core 3 - relative power distribution from the 3-D PDQ-7 calculation compared with NUTRIX-1 calculations having different transport kernels - 849.3 Mwd/mtU.

.5739	.7319	.6253	.8245	.7615	.9551	.8985	.8469
.8300	.9793	.8756	1.043	.9427	1.058	.9544	.9322
.7244	.8809	.7772	.9646	.8738	1.036	.9726	1.019
.4689	.6379	.5361	.7663	.6890	.9620	.9915	1.251
.7317	.6772	.8093	.7782	.9148	.9127	1.258	1.150
.9799	.9238	1.059	.9900	1.104	1.007	1.167	.8507
.8821	.8235	.9744	.9088	1.058	.9871	1.245	.9519
.6376	.5729	.7628	.6986	.9395	.9179	1.486	1.266
.6251	.8085	.8182	.7976	.9566	.9080	1.379	
.8778	1.061	1.064	.9659	1.086	.9262	1.017	
.7815	.9777	.9867	.8936	1.054	.9171	1.098	
.5351	.7608	.7820	.6987	.9728	.8887	1.382	
.8267	.7782	.7664	.8174	1.033	1.292	1.350	
1.051	.9928	.9570	.9319	1.102	1.146	.8575	
.9778	.9162	.8869	.8799	1.107	1.223	.9488	
.7652	.6937	.6834	.7284	1.136	1.514	1.287	
.7908	.9278	.9585	1.033	1.143	1.363		
.9616	1.116	1.089	1.104	1.036	.8650		
.9022	1.079	1.064	1.114	1.077	.9423		
.6985	.9356	.9604	1.125	1.202	1.235		
.8367	.9459	.9176	1.295	1.363			
1.056	1.025	.9324	1.150	.8668			
1.032	1.018	.9326	1.236	.9481			
.8756	.9174	.8744	1.487	1.220			
.9267	1.302	1.406	1.359				
.9496	1.165	1.020	.8609				
.9810	1.261	1.116	.9627				
.9546	1.442	1.346	1.256				
1.006	1.166	PDQ-7					
.9003	.8214	0.5 - G					
1.013	.9326	1.0					
1.252	1.185	2.0					

Fig. 19 Connecticut Yankee Core 3 — relative power distribution from the 3-D PDQ-7 calculation compared with NUTRIX-1 calculations having different transport kernels — 2,548.0 Mwd/mtU.

.5594	.7126	.6049	.7981	.7350	.9334	.8869	.8881
.8621	1.010	.9002	1.060	.9507	1.052	.9414	.9083
.7945	.9563	.8344	1.010	.8951	1.028	.9441	.9680
.6782	.8992	.7150	.9355	.7600	.9468	.8942	1.069
.7126	.6534	.7827	.7480	.8876	.8882	1.271	1.224
1.010	.9525	1.081	1.008	1.106	1.004	1.142	.8320
.9571	.8910	1.034	.9519	1.073	.9806	1.195	.9079
.8990	.7927	.9906	.8435	1.007	.8934	1.302	1.083
.6052	.7825	.7893	.7679	.9263	.8859	1.431	
.9022	1.082	1.082	.9812	1.086	.9223	.9978	
.8372	1.036	1.036	.9277	1.061	.9070	1.058	
.7149	.9897	.9752	.8058	.9995	.8380	1.206	
.8015	.7491	.7639	.7885	1.001	1.290	1.452	
1.067	1.010	.9713	.9404	1.095	1.127	.8434	
1.020	.9563	.9177	.8955	1.096	1.185	.9152	
.9384	.8416	.9890	.7690	1.088	1.346	1.116	
.7654	.9025	.9296	1.002	1.125	1.425		
.9702	1.118	1.089	1.097	1.027	.8544		
.9196	1.089	1.067	1.100	1.055	.9163		
.7788	1.015	.9660	1.085	1.112	1.101		
.8274	.9238	.8975	1.295	1.426			
1.053	1.024	.9298	1.132	.8565			
1.022	1.008	.9189	1.194	.9201			
.8844	.9135	.8396	1.341	1.097			
.9166	1.318	1.461	1.463				
.9440	1.146	1.004	.8482				
.9537	1.209	1.071	.9249				
.8931	1.305	1.206	1.112				
1.046	1.242	— PDQ-7					
.8918	.8127	— 0.5 - G					
.9723	.8938	— 1.0					
1.119	1.057	— 2.0					

Fig. 20 Connecticut Yankee Core 3 — relative power distribution from the 3-D PDQ-7 calculation compared with NUTRIX-1 calculations having different transport kernels — 4, 246.6 Mwd/mtU.

.5585	.7091	.5959	.7797	.7096	.9016	.8587	.9039
.8798	1.025	.9134	1.068	.9572	1.051	.9381	.8982
.8333	.9951	.8657	1.034	.9100	1.028	.9338	.9447
.7742	1.011	.7958	1.011	.7989	.9494	.8615	.9995
.7096	.6447	.7695	.7264	.8605	.8565	1.256	1.268
1.025	.9667	1.090	1.017	1.107	1.004	1.129	.8246
.9954	.9264	1.063	.9760	1.081	.9805	1.170	.8879
1.011	.8897	1.083	.9116	1.039	.8905	1.226	1.012
.5977	.7706	.7723	.7455	.8972	.8602	1.457	
.9150	1.090	1.089	.9905	1.088	.9235	.9889	
.8672	1.064	1.060	.9480	1.066	.9053	1.037	
.7963	1.082	1.055	.8594	1.014	.8211	1.129	
.7873	.7307	.7436	.7651	.9716	1.283	1.538	
1.073	1.018	.9801	.9463	1.091	1.116	.8370	
1.040	.9775	.9361	.9057	1.088	1.163	.8976	
1.015	.9107	.8414	.7925	1.064	1.263	1.038	
.7463	.8828	.9062	.9758	1.108	1.485		
.9749	1.117	1.090	1.092	1.021	.8490		
.9299	1.093	1.068	1.090	1.042	.9013		
.8207	1.050	1.014	1.063	1.066	1.034		
.8165	.9039	.8809	1.297	1.491			
1.050	1.023	.9300	1.120	.8506			
1.016	1.002	.9130	1.168	.9032			
.8936	.9160	.8269	1.264	1.034			
.9036	1.327	1.511	1.566				
.9419	1.134	.9946	.8409				
.9402	1.179	1.045	.9028				
.8701	1.240	1.137	1.041				
1.078	1.309	— PDQ-7					
.8882	.8086	— 0.5 - G					
.9507	.8730	— 1.0					
1.061	.9997	— 2.0					

Fig. 21 Connecticut Yankee Core 3 — relative power distribution from the 3-D PDQ-7 calculation compared with NUTRIX-1 calculations having different transport kernels — 5, 945.3 Mwd/mtU.

.5059	.6430	.5416	.7149	.6573	.8530	.8318	.9360
.9985	1.031	.9204	1.070	.9620	1.052	.9391	.8947
.8543	1.013	.8825	1.046	.9201	1.030	.9307	.9333
.8058	1.041	.8243	1.039	.8238	.9629	.8594	.9796
.6442	.5831	.7011	.6659	.8040	.8156	1.251	1.353
1.031	.9728	1.091	1.021	1.107	1.006	1.123	.8224
1.014	.9437	1.075	.9886	1.086	.9820	1.156	.8780
1.040	.9188	1.106	.9406	1.057	.8998	1.200	.9905
.5453	.7040	.7090	.6920	.8495	.8374	1.527	
.9215	1.092	1.092	.9958	1.089	.9269	.9848	
.8833	1.076	1.072	.9600	1.070	.9072	1.026	
.8241	1.105	1.078	.8869	1.028	.8234	1.101	
.7279	.6746	.6931	.7240	.9398	1.304	1.712	
1.075	1.022	.9859	.9507	1.089	1.110	.8345	
1.050	.9886	.9481	.9133	1.085	1.149	.8879	
1.041	.9381	.8677	.8098	1.057	1.227	1.006	
.7021	.8358	.8657	.9479	1.114	1.622		
.9776	1.115	1.091	1.090	1.019	.8469		
.9368	1.094	1.071	1.086	1.035	.8932		
.8428	1.064	1.025	1.056	1.048	1.005		
.7939	.8784	.8705	1.329	1.635			
1.048	1.022	.9320	1.113	.8479			
1.013	.9998	.9124	1.152	.8941			
.9023	.9212	.8266	1.226	1.004			
.8978	1.359	1.614	1.765				
.9421	1.126	.9890	.8373				
.9348	1.161	1.031	.8907				
.8643	1.208	1.104	1.006				
1.140	1.432	— PDQ-7					
.8877	.8074	— 0.5 — G					
.9403	.8626	— 1.0					
1.035	.9737	— 2.0					

Fig. 22 Connecticut Yankee Core 3 — relative power distribution from the 3-D PDQ-7 calculation compared with NUTRIX-1 calculations having different transport kernels — 7, 644.0 Mwd/mtU.

Fig. 23 Relative power in location J-14 of Core 3 as a function of burnup.

Fig. 24 Critical boron concentration as a function of burnup from PDQ-7 and NUTRIX-1 calculations for the reference reactor.

17670	16200	18360	16650	18830	17740	18620	16310
16200	18250	17480	18740	17700	18950	17400	17010
18360	17480	18900	17630	18720	17370	17930	14910
16650	18740	17630	18540	16590	17980	16410	12800
18830	17700	18720	16590	18960	15700	15560	
17740	18950	17370	17980	15700	17210	11740	
18620	17400	17930	16410	15560	11740		
16310	17010	14910	12800				

Fig. 25 Reference PWR — NUTRIX-1 model assembly-wise burnup at an average core burnup of 17,000 Mwd/mtU.

18740	18930	19090	19130	19040	18910	16660	13330
18930	19040	19710	19230	19520	18390	17590	13630
19090	19710	19380	19750	18930	18530	16000	12350
19130	19230	19750	19090	18690	17520	15960	10810
19040	19520	18930	18690	19570	16450	14590	
18910	18390	18530	17520	16450	16360	10270	
16660	17590	16000	15960	14590	10270		
13330	13630	12350	10810				

Fig. 26 Reference PWR – PDQ-7 two-dimensional model assembly-wise burnup at an average core burnup of 17,000 Mwd/mtU.

1. 140	1. 041	1. 183	1. 054	1. 176	1. 075	1. 096	. 9292
1. 041	1. 179	1. 122	1. 190	1. 096	1. 144	1. 011	. 9624
1. 183	1. 122	1. 208	1. 100	1. 139	1. 019	1. 018	. 8180
1. 054	1. 190	1. 100	1. 129	. 9718	1. 019	. 8968	. 6792
1. 176	1. 096	1. 139	. 9718	1. 079	. 8520	. 8209	
1. 075	1. 144	1. 019	1. 019	. 8520	. 9049	. 5962	
1. 096	1. 011	1. 018	. 8968	. 8209	. 5962		
. 9292	. 9624	. 8180	. 6792				

Fig. 27 Reference PWR — NUTRIX-1 model, assembly-wise power distribution, BOL.

1. 126	1. 030	1. 170	1. 045	1. 169	1. 073	1. 097	. 9331
1. 030	1. 166	1. 111	1. 180	1. 091	1. 142	1. 104	. 9671
1. 170	1. 111	1. 197	1. 094	1. 136	1. 020	1. 022	. 8241
1. 045	1. 180	1. 094	1. 126	. 9727	1. 024	. 9040	. 6863
1. 169	1. 091	1. 136	. 9727	1. 083	. 8593	. 8301	
1. 073	1. 142	1. 020	1. 024	. 8593	. 9156	. 6049	
1. 097	1. 104	1. 022	. 9040	. 8301	. 6049		
. 9331	. 9671	. 8241	. 6863				

Fig. 28 Reference PWR — NUTRIX-1 model, assembly-wise power distribution, 1,000 Mwd/mtU.

1.111	1.016	1.156	1.035	1.161	1.070	1.099	.9378
1.016	1.151	1.098	1.169	1.084	1.140	1.016	.9727
1.156	1.098	1.184	1.086	1.132	1.021	1.028	.8314
1.035	1.169	1.086	1.121	.9736	1.029	.9124	.6946
1.161	1.084	1.132	.9736	1.089	.8679	.8410	
1.070	1.140	1.021	1.029	.8679	.9281	.6151	
1.099	1.016	1.028	.9124	.8410	.6151		
.9378	.9727	.8314	.6946				

Fig. 29 Reference PWR – NUTRIX-1 model, assembly-wise power distribution, 2,000 Mwd/mtU.

1.094	1.002	1.140	1.024	1.152	1.067	1.101	.9429
1.002	1.134	1.084	1.157	1.078	1.137	1.019	.9788
1.140	1.084	1.170	1.077	1.127	1.022	1.034	.8394
1.024	1.157	1.077	1.116	.9746	1.035	.9217	.7038
1.152	1.078	1.127	.9746	1.094	.8774	.8530	
1.067	1.137	1.022	1.035	.8774	.9420	.6265	
1.101	1.019	1.034	.9217	.8530	.6265		
.9429	.9788	.8394	.7038				

Fig. 30 Reference PWR — NUTRIX-1 model, assembly-wise power distribution, 3,000 Mwd/mtU.

1.078	.9878	1.124	1.014	1.143	1.063	1.102	.9477
.9878	1.118	1.069	1.144	1.070	1.134	1.022	.9848
1.124	1.069	1.156	1.068	1.123	1.023	1.040	.8475
1.014	1.144	1.068	1.111	.9755	1.041	.9312	.7133
1.143	1.070	1.122	.9755	1.100	.8873	.8656	
1.063	1.134	1.023	1.041	.8873	.9564	.6385	
1.102	1.022	1.040	.9312	.8656	.6385		
.9477	.9848	.8475	.7133				

Fig. 31 Reference PWR – NUTRIX-1 model, assembly-wise power distribution, 4,000 Mwd/mtU.

1.063	.9743	1.109	1.003	1.134	1.060	1.102	.9523
.9743	1.102	1.056	1.132	1.063	1.131	1.024	.9905
1.109	1.056	1.142	1.059	1.118	1.024	1.045	.8557
1.003	1.132	1.059	1.106	.9763	1.047	.9408	.7232
1.134	1.063	1.118	.9763	1.106	.8972	.8784	
1.060	1.131	1.024	1.047	.8972	.9712	.6510	
1.102	1.024	1.045	.9408	.8784	.6510		
.9523	.9905	.8557	.7232				

Fig. 32 Reference PWR — NUTRIX-1 model, assembly-wise power distribution, 5,000 Mwd/mtU.

1.049	.9615	1.094	.9920	1.125	1.055	1.103	.9567
.9615	1.086	1.042	1.119	1.055	1.127	1.026	.9960
1.094	1.042	1.129	1.050	1.112	1.025	1.051	.8639
.9920	1.119	1.050	1.100	.9770	1.052	.9504	.7332
1.125	1.055	1.112	.9770	1.111	.9073	.8915	
1.055	1.127	1.025	1.052	.9073	.9863	.6640	
1.103	1.026	1.051	.9504	.8915	.6640		
.9567	.9960	.8639	.7332				

Fig. 33 Reference PWR – NUTRIX-1 model, assembly-wise power distribution, 6,000 Mwd/mtU.

1.036	.9500	1.081	.9821	1.115	1.051	1.102	.9604
.9500	1.073	1.030	1.108	1.048	1.123	1.027	1.001
1.081	1.030	1.116	1.041	1.107	1.025	1.056	.8718
.9821	1.108	1.041	1.095	.9775	1.058	.9597	.7432
1.115	1.048	1.107	.9775	1.116	.9171	.9044	
1.051	1.123	1.025	1.058	.9171	1.001	.6769	
1.102	1.027	1.056	.9597	.9044	.6769		
.9604	1.001	.8718	.7432				

Fig. 34 Reference PWR – NUTRIX-1 model, assembly-wise power distribution, 7,000 Mwd/mtU.

1.024	.9396	1.068	.9726	1.106	1.046	1.101	.9639
.9396	1.060	1.018	1.096	1.040	1.118	1.028	1.006
1.068	1.018	1.104	1.033	1.102	1.025	1.060	.8796
.9726	1.096	1.033	1.090	.9778	1.062	.9688	.7532
1.106	1.040	1.102	.9778	1.121	.9266	.9173	
1.046	1.118	1.025	1.062	.9266	1.016	.6901	
1.101	1.028	1.060	.9688	.9173	.6901		
.9639	1.006	.8796	.7532				

Fig. 35 Reference PWR — NUTRIX-1 model, assembly-wise power distribution, 8,000 Mwd/mtU.

1.014	.9307	1.057	.9640	1.097	1.041	1.100	.9667
.9307	1.049	1.007	1.086	1.033	1.114	1.029	1.010
1.057	1.007	1.092	1.025	1.096	1.025	1.064	.8870
.9640	1.086	1.025	1.085	.9780	1.067	.9774	.7630
1.097	1.033	1.096	.9780	1.125	.9358	.9298	
1.041	1.114	1.025	1.067	.9358	1.030	.7031	
1.100	1.029	1.064	.9774	.9298	.7031		
.9667	1.010	.8870	.7630				

Fig. 36 Reference PWR – NUTRIX-1 model, assembly-wise power distribution, 9,000 Mwd/mtU.

1.006	.9231	1.047	.9560	1.088	1.036	1.098	.9672
.9231	1.039	.9980	1.076	1.026	1.108	1.029	1.013
1.047	.9980	1.082	1.018	1.091	1.024	1.067	.8941
.9560	1.076	1.018	1.080	.9779	1.071	.9857	.7728
1.088	1.026	1.091	.9779	1.128	.9445	.9419	
1.036	1.108	1.024	1.071	.9445	1.044	.7162	
1.098	1.029	1.067	.9857	.9419	.7162		
.9672	1.013	.8941	.7728				

Fig. 37 Reference PWR — NUTRIX-1 model, assembly-wise power distribution, 10,000 Mwd/mtU.

1.000	.9180	1.039	.9494	1.080	1.030	1.095	.9708
.9180	1.031	.9905	1.068	1.019	1.103	1.028	1.016
1.039	.9905	1.073	1.011	1.086	1.023	1.070	.9006
.9494	1.068	1.011	1.075	.9775	1.074	.9930	.7821
1.080	1.019	1.086	.9775	1.131	.9524	.9533	
1.030	1.103	1.023	1.074	.9524	1.056	.7286	
1.095	1.028	1.070	.9930	.9533	.7286		
.9708	1.016	.9006	.7821				

Fig. 38 Reference PWR — NUTRIX-1 model, assembly-wise power distribution, 11,000 Mwd/mtU.

.9933	.9117	1.030	.9423	1.072	1.025	1.092	.9728
.9117	1.023	.9821	1.059	1.012	1.098	1.027	1.019
1.030	.9821	1.064	1.004	1.080	1.022	1.073	.9075
.9423	1.059	1.004	1.070	.9772	1.077	1.001	.792
1.072	1.012	1.080	.9772	1.134	.9606	.9651	
1.025	1.098	1.022	1.077	.9606	1.069	.7418	
1.092	1.027	1.073	1.001	.9651	.7418		
.9728	1.019	.9075	.7920				

Fig. 39 Reference PWR — NUTRIX-1 model, assembly-wise power distribution, 12,000 Mwd/mtU.

.9884	.9072	1.023	.9361	1.064	1.020	1.089	.9744
.9072	1.016	.9752	1.050	1.006	1.092	1.026	1.021
1.023	.9752	1.055	.9982	1.075	1.020	1.075	.9142
.9361	1.050	.9982	1.065	.9764	1.079	1.008	.8017
1.064	1.006	1.075	.9764	1.135	.9681	.9763	
1.020	1.092	1.020	1.079	.9681	1.082	.7546	
1.089	1.026	1.075	1.008	.9763	.7546		
.9744	1.021	.9142	.8017				

Fig. 40 Reference PWR – NUTRIX-1 model, assembly-wise power distribution, 13,000 Mwd/mtU.

.9854	.9044	1.017	.9309	1.056	1.014	1.086	.9756
.9044	1.011	.9695	1.043	.9999	1.086	1.025	1.024
1.017	.9695	1.048	.9924	1.070	1.018	1.076	.9204
.9309	1.043	.9924	1.060	.9756	1.081	1.014	.8112
1.056	.9999	1.070	.9756	1.136	.9749	.9869	
1.014	1.086	1.018	1.081	.9749	1.094	.7672	
1.086	1.025	1.076	1.014	.9869	.7672		
.9756	1.024	.9204	.8112				

Fig. 41 Reference PWR — NUTRIX-1 model, assembly-wise power distribution, 14,000 Mwd/mtU.

.9836	.9026	1.012	.9264	1.049	1.008	1.082	.9764
.9026	1.007	.9647	1.036	.9940	1.080	1.023	1.025
1.012	.9647	1.041	.9870	1.064	1.016	1.078	.9265
.9264	1.036	.9870	1.055	.9744	1.082	1.020	.8206
1.049	.9940	1.064	.9744	1.137	.9811	.9970	
1.008	1.080	1.016	1.082	.9811	1.104	.7795	
1.082	1.023	1.078	1.020	.9970	.7795		
.9764	1.025	.9265	.8206				

Fig. 42 Reference PWR – NUTRIX-1 model, assembly-wise power distribution, 15,000 Mwd/mtU.

.9829	.9018	1.008	.9225	1.042	1.003	1.078	.9771
.9018	1.004	.9607	1.030	.9884	1.074	1.021	1.027
1.008	.9607	1.035	.9820	1.059	1.014	1.078	.9324
.9225	1.030	.9820	1.051	.9732	1.083	1.026	.8301
1.042	.9884	1.059	.9732	1.136	.9868	1.007	
1.003	1.074	1.014	1.083	.9868	1.114	.7917	
1.078	1.021	1.078	1.026	1.006	.7917		
.9771	1.027	.9324	.8301				

Fig. 43 Reference PWR – NUTRIX-1 model, assembly-wise power distribution, 16,000 Mwd/mtU.

.9833	.9019	1.005	.9192	1.036	.9976	1.074	.9777
.9019	1.002	.9575	1.025	.9832	1.068	1.019	1.028
1.005	.9575	1.030	.9775	1.054	1.011	1.079	.9382
.9192	1.025	.9775	1.046	.9717	1.084	1.031	.8395
1.036	.9832	1.054	.9717	1.135	.9919	1.016	
.9976	1.068	1.011	1.084	.9919	1.123	.8036	
1.074	1.019	1.079	1.031	1.016	.8036		
.9777	1.028	.9382	.8395				

Fig. 44 Reference PWR — NUTRIX-1 model, assembly-wise power distribution, 17,000 Mwd/mtU.

Fig. 45 Comparison of measured and predicted self-shielding effect, Targets A and B.

Fig. 46 Comparison of measured and predicted self-shielding effect, Target C.

Fig. 47 Comparison of measured and predicted self-shielding effect, Target D.

Fig. 48 Error Plot of Pu and Np Composition determined by three techniques of chemical analysis.

Fig. 49 Parametric study of cross-section effects on Pu production, Target Segment A.

Fig. 50 Parametric study of cross-section effects on Pu production, Target Segment B.

Fig. 51 Parametric study of cross-section effects on Pu production, Target Segment C.

Fig. 52 Parametric study of cross-section effects on Pu production, Target Segment D.

Fig. 53 Effect of annual production rates on the cost of Pu-238 production.

APPENDIX A
DRAFT OF LETTER OF INVITATION TO UTILITIES
TO ATTEND SEMINAR ON PU-238 PRODUCTION

Dear _____:

As part of the ongoing effort of the U.S. Atomic Energy Commission, the Space Nuclear Systems Division has, for several years, sponsored a development program to determine the technical and economic feasibility of producing plutonium-238 in the private sector, in power reactors owned and operated by members of the electric utility industry. Southern Nuclear Engineering, Inc. of Dunedin, Florida, manages the program for the AEC. Results of the program are highly encouraging and offer the potential for developing a market for neptunium from irradiated reactor fuel, as well as for the plutonium-238 obtained from subsequent irradiation of the neptunium in utility and/or Government owned reactors.

The Space Nuclear Systems Division has determined that further dissemination of the preliminary information to the electric utility industry would be valuable to both the AEC and the industry. There is a high degree of interest on the part of the Commission in exploring the level of interest on the part of the utility industry in participating in further development of the process and in a production program whose purpose is supplying the Government's needs for plutonium-238.

Accordingly, a one-day Seminar will be held on _____ to acquaint you with the program. The agenda will include a briefing on the Government

projects requiring plutonium-238, a complete summary of research efforts including estimates of the costs of production, a discussion of recovery processes, a discussion of licensing considerations, and a summary of its experience by one utility company which has already participated in an experimental irradiation of neptunium.

You or your representative(s) are cordially invited to participate in the Seminar. The program will begin at _____ and will be held in _____ at _____ (time and place to follow) _____.

At the close of the Seminar you will be given a questionnaire designed to enable the Government to assess your interest in a proposed production program. The Seminar will cover the essential topics to be considered by you in response to the questionnaire.

It will be greatly appreciated if you will advise the undersigned whether or not you plan to attend. Also, please indicate the name(s) of your representative(s).

We look forward to meeting with you.

Very truly yours,

APPENDIX B
QUESTIONNAIRE ON PU-238 PRODUCTION

In assessing future requirements for isotopic power sources, the Government anticipates the need for sizable quantities of Pu-238 annually for a substantial period of time. Certain information is needed from you in order to assess the overall interests and capability of the private sector in cooperating in a plutonium-238 production program. This questionnaire is designed to make the transmittal and compilation of information relatively easy. Your cooperation in completing and returning this form is greatly appreciated.

As the attached return envelope indicates, the form should be returned to

by _____.

1. Name of utility _____
2. List NUCLEAR power stations/actual or scheduled startup date
 - a.) _____
 - b.) _____
 - c.) _____
 - d.) _____
 - e.) _____

List additional stations on back of sheet.

3. Are you now having or plan to have the spent fuel reprocessor recover Np-237 for you? _____

4. How much neptunium could you provide annually and at what price? Please provide estimates for each of the years 1975-1984.

5. Would you prefer to sell your neptunium directly to the U.S. Government or would you prefer to relegate that function to your fuel reprocessor?

6. Would you be willing to irradiate neptunium target rods in your nuclear power station(s)? _____

7. In the irradiation of neptunium target material would you prefer to accept a small reduction in the core cycle time? _____

or would you prefer to increase the fuel enrichment to offset the effect of the Np-237? _____

8. Would you prefer to sell the recovered Np-237 to the U.S. Government and provide irradiation space or would you prefer to recover the Np-237 from your spent fuel, fabricate it, irradiate it, recover Np-237 and Pu-238 and sell the recovered Pu-238 to the Government? Refer to Figure 1 attached and indicate below those parts of the Process Flow diagram in which you would be willing to participate.

Check

- () (1) through (6) - Spent fuel through Pu-238 recovery and sell to U.S. Government
- () (1) to (2) - Recover Np-237 from spent fuel
- () Option - Sell Np-237 to Government at this point
- () (2) to (3) - Fabricate Target Rods
- () (3) to (4) - In-Reactor Irradiation
- () (4) to (5) - Ship irradiated rods for chemical recovery
- () (5) to (6) - Recover Np-237 and Pu-238 and sell Pu-238 to U.S. Government

9. What do you conclude is the best overall program arrangement for participation by utilities, spent fuel reprocessors and the Government to achieve the production goal of low cost Pu-238?

10. What was your opinion of the seminar? _____

11. Do you feel that future seminars on this subject would be worthwhile?

Fig. B-1 Process flow diagram.