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May 11, 1962

Technical Evaluation
of E-N Demonstration Loading
(Interim Report, Production Test IP-350-C)

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TECHNICAL EVALUATION OF E-N DEMONSTRATION LOADING
(Interim Report, Production Test IP-350-C)

INTRODUCTION

This report presents the technical observations and conclusions to date of the full-reactor demonstration E-N loading for the combined production of plutonium and tritium.

A major incentive for such a loading is the increase in conversion ratio resulting from three factors:

1. A "blacker" lattice, in which a combination of "E" metal fuel elements (uranium enriched to 0.95% U²³⁵) and "N" metal target elements (enriched lithium-aluminum alloy) approximately matches the reactivity properties of a natural uranium lattice, results in a smaller proportion of parasitic capture of neutrons in the moderator and other reactor components.
2. The capture of neutrons in fringe target material results in the formation of useful product by a significant fraction of the neutrons otherwise lost by leakage.
3. The burnout of product atoms (plutonium and tritium) is reduced in the blacker E-N lattice, thereby increasing the net yield of product atoms per MWD of heat generation compared to the natural loading.

Potential advantages for plant-wide E-N production include the routine processing of a single metal stream for Hanford, increased product quality at reduced throughput, and increased production and recovery of higher isotopes in recycled streams.

A demonstration test, authorized by the Atomic Energy Commission for irradiation in the H Reactor^{1/} consisted of two parts corresponding to the above conversion-efficiency incentives - a "striped" E-N core loading of two irradiation cycles completed in early 1962, and a single-cycle "fringe blanket" loading to be discharged in late 1962.^{2,3/}

Objectives of the test were to provide the following information for use in further evaluation of the incentives for this type of loading for Hanford production reactors:

1. Its Operational Feasibility - Reactivity and flux distribution factors affecting reactor efficiency and nuclear safety, handling requirements for the safe and efficient charging and discharging of large quantities of enriched fuel, assurance of flow distribution compatible with the observed power distribution, and determination of E-N irradiation environmental effects on fuel elements and other reactor lattice components.
2. Its Conversion Efficiency - Large scale data on both the plutonium buildup and tritium formation in the striped core loading, and small scale data on columns or batches representative of particular geometries and exposures in the core, the E-ring, and the fringe poison.

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SUMMARY

Operational feasibility of the E-N concept has been demonstrated. Short term and long term reactivity transients observed in the E-N loading, although obscured in part by water leak experience characteristic of recent operation in the older reactors, were lower in magnitude than in a natural uranium loading and were thus compatible with current operating techniques.

Radial power distribution was in good agreement with expected conditions. The expected degree of longitudinal flattening was not obtained; however, the demonstration case provides a satisfactory basis for future column charge pattern planning. Charge-discharge of enriched fuel was carried out under safe practices at rates limited during the demonstration test by enrichment pickup requirements in the discharge chute; pickup rates were only slightly more limiting, however, than initial bumper element charging rates. Aside from factors not attributable to the E-N elements as such, performance of bumper fuel elements, utilized in this loading simultaneous with their initial large-scale demonstration in a natural loading at D Reactor, was consistent with the current rupture model as it applied to specific E-N fuel power, temperature, and exposure conditions. No evidence was observed that the higher specific fuel element heat generation rates affected other lattice component parameters such as graphite temperatures and process tube corrosion rates more than would be calculated. The net difference in production efficiency which might be attributed to the E-N load itself showed no significant trend (changes $\leq \pm 1\%$).

Separations data, from the tritium recovery at Savannah River and the plutonium recovery in Redox, show a significant increase in conversion ratio over that of natural uranium. However, plutonium recovery measurements from bulk E-metal samples from the first core are considerably lower than previous single-column samples from a small-scale block test had indicated.^{4/} Table I on page 6 shows expected conversion ratios in natural uranium, those obtained from the first core loading, and those indicated by the previous small-scale test. Corrections for the effect on conversion ratio of compromises in the first core loading made to ensure operational flexibility during an initial run have been calculated and included in the Demonstration-Test data; the expected relation between Redox head-end alpha-counting indications and actual recovery results is also included in the latter figure.

Demonstration test data shown in the table indicate an H Reactor gain in over-all reactor conversion efficiency of 15 per cent for an E-N load relative to a normal load (85 per cent heat generation in natural uranium and 15 per cent heat generation in E columns) assuming both are irradiated to 800 MWD/T goal exposure. Comparison of 800 MWD/T E-N results to data from a more realistic current natural uranium goal of 625 MWD/T would show an increase of 12 per cent with an E-N loading in H Reactor.

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Table I

Comparative Conversion Data (at 800 MWD/T)

	Plutonium (gm/MWD)	Tritium (gm-equiv/MWD)	Net Reactor Conversion Eff. (gm-equiv/MWD)
<u>Natural Uranium Loading (current CR Tables)</u>			
a. Natural Uranium	.818	0	
b. Supporting E (CR Tables adjusted for Redox Recovery observations)	.697	0	
c. Pile Average (85% natural, 15% E)	.800	0	<u>.800</u>
<u>Previous Small-Scale Data</u>			
a. 30-Tube block	.767	.204	.971
b. Previous blanket test*	.723	.200	.923
c. Pile Average (72% core, 28% blanket)	.755	.203	<u>.958</u>
<u>Demonstration Test Data</u>			
a. Adjusted core data**	.722	.213	.935
b. Previous blanket test*	.705	.200	.905
c. Pile average (68% core, 32% blanket)	.717	.209	<u>.926</u>

* Data include weighting of outer blanket E results by core data at the time.

** Adjusted to 800 MWD/T and for expected change in optimized final E-N load (from improved E to N ratio, reduced excess in control elements, use of E metal in mixer pieces, etc.).

In summary, irradiation experience with the E-N load has demonstrated its operational feasibility. However, in order to further improve the efficiency of large-scale operational loadings and the basis for economic planning, additional demonstration-testing on a full-reactor basis would be desirable prior to the initiation of a major-scale Hanford production program for the following reasons:

1. To provide conversion ratio data in addition to that already reported and to be available from the second cycle. Past experience with natural loading recovery data had indicated that even bulk results should be averaged over a large number of reactor loadings.
2. To provide increased knowledge of the loading's reactivity and fuel performance parameters under more normal operating conditions.
3. To continue incentives for the development of operational practices and fuel handling techniques specific to an E-N load.

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DISCUSSION

A. Background of E-N Demonstration Test

This section of the report describes qualitatively and chronologically the development of incentives and technology leading up to the authorization for a demonstration E-N loading and the planning and execution of that loading.

1. Development of Incentives and Technology

a. Operating Precedents

The discussion of this topic parallels a brief introductory section labeled "Historical" in a document previously routed to the Atomic Energy Commission.^{/5/}

Enrichment, striped loadings, blanket loadings, and other operational features of the E-N program have considerable precedent in Hanford operating history. Previous large-scale tritium production programs were carried out at Hanford in 1950 and again in 1954 utilizing U²³⁵-aluminum alloy (J metal) fuel and natural lithium-aluminum alloy target elements. Enrichment to increase reactivity and extend the degree of radial flattening in natural uranium loads was begun as an operating practice in 1952; U²³⁵-aluminum alloy (C metal) fuel elements were used in this case. Fringe poisoning employing Li-Al target elements was initiated as a production practice in 1955. And the changeover from weapons-grade U²³⁵-Al alloy C metal, formerly used to support natural loading production, to E metal (uranium enriched to 0.95% U²³⁵) was largely completed during 1957. Thus, the mechanics of large-scale E-N loadings at Hanford represent an extension of operating practices previously developed and applied on a smaller scale.

b. Preliminary E-N Irradiations

As mentioned previously, the "blacker" lattice provided by an E-N load appeared attractive for improving the over-all conversion efficiency of a Hanford reactor. Furthermore, the use of E metal as the fuel offered certain operational advantages relative to J metal including an inherent negative metal temperature coefficient, higher melting temperature, fail-safe reactivity characteristics on meltdown, reduced graphite temperatures, and lower U²³⁵ burnout costs.

On the basis of the above incentives, a small block irradiation was carried out in the B Reactor in early 1957 utilizing solid and cored enriched fuel elements. Following concurrent economic studies and interest by the AEC in a proposal^{/6/} of a full-reactor demonstration test in early 1959, a small-scale test was irradiated in H Reactor late in 1959 which included both a central striped block utilizing I&E elements and certain fringe columns; hazards analyses showing the acceptability of the E-N concept were concurrently carried out and presented to the Advisory Committee on Reactor Safeguards.^{/7,8/} Subsequent to publication of plutonium and tritium recovery data from the 1959 H Reactor irradiation test^{/4/}, the Atomic Energy Commission gave final approval^{/1/} for proceeding with the reported full-reactor demonstration test.

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2. Preparation for Demonstration Test

a. Striped-Core Reactivity Matching

Advance calculations indicated initial cold, clean reactivity requirements of the full E-N load to be quite high, largely because of the possible decrease in moderator coefficient and the potential long-term reactivity losses; failure to provide sufficient initial reactivity might result in inadequate equilibrium control flexibility later in the cycle. On the other hand, excess reactivity flexibility held in supplementary control would reduce over-all reactor conversion efficiency and would detract from transient-control flexibility, or operating efficiency. Incentives were therefore strong for careful advance determination of the lattice reactivity of the loading. Cold, dry tests were carried out in the Hanford Test Pile; a small (15 column) wet test was carried out under near xenon-free conditions in the H Reactor; and a 250-tube block was loaded into the H Reactor two weeks in advance of the bulk of the first core loading as a check on hot reactivity effects.

Additional flexibility, partially at the expense of basic lattice conversion efficiency, was provided by the holding back of a number of columns in the central flattened region for PCCF and permanent poison columns and by the loading of sufficient E-ring enrichment to compensate the additional excess reactivity potential to be held in poison columns, control rods, and/or poison splines (the intended activation of the spline system was expedited at the H Reactor in advance of the first striped-core cycle). Tritium yield rates were recovered in part by the charging of up to 18 partial columns of N metal in the core during an appreciable part of the first core cycle. Absolute and transient reactivity observations of the first core load indicated that the control flexibility could have been reasonably reduced by .7-.8% k; the loading pattern was correspondingly revised for the second core load. Some additional reduction in control flexibility requirements would appear to be feasible in future loads.

"Black" N elements, containing lithium enriched to 38.5 per cent lithium 6, were used in the loading and in the reactivity-matching tests in preparation for the demonstration loading.

b. Blanket Load Preparations

Earlier experience with fringe poison (as a means of reducing shield operating temperatures) had indicated the requirement for compensating enrichment in adjacent lattice units to be approximately 2-1/2 E columns per fringe N column. Thus, with an E-ring for flattening purposes in H Reactor approximately two lattice units wide and centered on the fifth lattice unit, the region between the first lattice unit and the core load would be almost completely filled with E columns. The blanket loading posed four problems in reactor operation.

1. Reactivity matching of the blanket.
2. Radial power distribution and flattening efficiency.
3. Radial flow distribution.
4. Total column heat generation rates in E columns.

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Reactivity matching was approached conservatively by the initial charging of only two E columns in the second and third lattice units per fringe N column; incremental increases in supporting E columns were then made during subsequent outages.

A solid fringe poison layer already existing for top and far-side shield protection at the H Reactor provided prototypical flux and power distribution data which proved to be quite accurate for planning purposes. On this basis, a reduction in effective central tubes of 6-8 per cent was expected. Full-length E columns appeared feasible for use in the second, third, and fourth lattice units, whereas shortened E column charges appeared advisable in the fifth and sixth lattice units if they were not to be limiting to total reactor power. Small gains in central column flow appeared possible from flow re-distribution if the loading were to be continued; however, revision of the flow pattern for the demonstration load as such was not attempted.

c. Use of Bumper Fuel Elements

The combined effects of reduced radial flattening efficiency and reduced fuel length in E-N columns resulted in a specific power increase per unit fuel length of 15 to 20 per cent compared to a complete natural loading; the reduced fuel length is due to displacement of fuel by N slugs and by the higher proportion of endcaps with the shorter E-metal fuel elements. This increase in specific power corresponds to that normally encountered in E columns within the flattened zone of a natural uranium loading. If fuel rupture performance were not to be limiting to operating level under the more severe irradiation conditions expected, an improvement in fuel integrity would be required.

The bumper concept for assuring fuel alignment within ribbed tubes and thereby eliminating hot-spot ruptures had shown sufficient promise on small-scale run-to-rupture tests during the summer of 1960 that large-scale loadings were initiated at D Reactor with natural uranium late that year. The special incentives for improved fuel performance in the H Reactor E-N load led to the decision to provide nozzle and fuel modifications necessary for use of the bumper concept with the first E-N loading concurrent with the early D Reactor bumper program.

The calculated risk taken in acceleration of the bumper fuel program achieved its objective - performance of the bumper E metal elements was consistent with expected rates. Charging damage not specifically related to the use of E metal as such caused a number of ruptures, and several others occurred in rupture-prone or high-irradiation lots whose irradiation was continued as an operating convenience associated with water leak and shutdown crew scheduling problems. Fuel performance experience thus agreed quite well with advanced predictions based on the existing rupture model.

d. Special Fuel Handling

Enriched uranium of .95 per cent U^{235} concentration requires storage and handling limitations to preclude the occurrence of a critical assembly in a water lattice; approximately three tons of E metal would permit a

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critical assembly in optimum lattice spacing and "ideal" shape geometry. Special pickups of up to two tons per increment are carried out with the relatively small proportion of enrichment normally discharged. For the large-scale demonstration test, however, special procedures were devised to allow larger increments to fall directly into buckets arranged in a critically-safe line.

Charging of the E-N core required the proper selection of one of five charge types to provide the intended staggering of target elements throughout the core. Sequence was specified in the charging and discharging of N columns and E columns to ensure that the interim reactivity of any reactor zone would not exceed its final intended startup state.

The net effect of the above considerations was a reduction in charge-discharge rates of approximately 20 per cent relative to natural uranium bumper elements. Continued test loadings would provide the incentive for further improvements which would hopefully increase E-N charge-discharge rates.

Advance preparations for storage of both irradiated and unirradiated fuel were carried out in addition to the development of handling techniques previously described.

B. Operational Characteristics

This section of the report covers those aspects of the preparation and execution of the full-reactor E-N demonstration test related to the operation of the reactor itself. Aspects covered include Reactivity Parameters, Control Measures, Engineering Parameters (including fuel performance), and Over-all Operating Efficiency. Conversion efficiency observations are presented in Section C. Data and conclusions regarding the operational characteristics of the demonstration test have been prepared by R. D. Carter. Fuel performance data and conclusions were prepared by R. D. Shimer.

1. Reactivity Parameters

This sub-section includes the development of loading patterns for obtaining the proper base reactivity of the large-scale load; the evaluation of the long and short term reactivity transients of concern to both operating efficiency and nuclear safety, the development of nuclear safety evaluations and application of their results, and the operating control flexibility requirements and achievements with the E-N load.

a. Base Reactivity Establishment

The calculation of the changes in pile reactivity between a natural and an E-N load consisted of essentially two parts - the fringe blanket load and the flattened zone striped, or E-N core section. Since about 46 fringe poison tubes with supporting enrichment already had been charged in H Reactor for some time for shield protection, it was not necessary to pre-calculate the enrichment support required for the poison in the blanket load. The only significant unknown effect therefore was that of the E-N core loading.

Since even a small uncertainty in small scale lattice reactivity test results can lead to large effects on operating efficiency, a conservative approach was followed in planning advance reactivity-matching tests.

Two tests to determine the reactivity changes were performed prior to the full E-N charge. In one test, loadings with various ratios of E slugs to Li-Al elements were checked for reactivity variations in the Hanford Test Pile (305 Bldg.). The data obtained from the test was used in conjunction with flux calculations to estimate the reactivity changes with the E-N loading. These calculations^{9/} showed a long-term decrease of up to 5 mk in the operating reactor over the full exposure range of the core (flat zone) subsequent to samarium saturation, and indicated that the over-all temperature coefficient of reactivity could be zero or even negative. These calculations were weighted for observed data for natural uranium. Although it was not believed that the changes would be this extreme, it was considered necessary for operational planning to assume that they were correct. Accordingly, the original core enrichment level was made sufficiently high to compensate for the calculated changes; until the extra reactivity was shown by operating experience to be needed it was to be compensated by spike poison columns.

The above estimates, based on test pile data and calculations subject to considerable interpretation, left considerable uncertainty in the reactivity

status of the load; therefore, a second preliminary reactivity test was performed at H Reactor using the proposed E-N charge makeup.^{10/} Because of a water leak, data obtained from this test was not conclusive but was of sufficient worth to show that calculations were not in error to the extent that reactor safety or production would be compromised.

As a further check on the reactivity status of the E-N loaded reactor the charging of the first core was initiated by charging about 250 striped charges in a square block in the exact center of the reactor on May 4, 1961; the remainder of the first cycle core was charged on May 22, 1961. This method had several advantages. It allowed a check on the gross "hot" reactivity effect and on the axial flux shaping effect of the striped loading so that modifications of the rest of the core load could be made if needed. In addition, some samarium fission product poison buildup occurred in the 250 tubes before the rest of the core load was charged so that the large poison requirements during the startup of a samarium-free reactor were lessened. No extreme effects were found, and the remainder of the core was charged as planned.

During subsequent irradiation the reactivity of the first core was found to be about 10 milli-k greater than necessary, requiring compensation by too many splines and spike poison columns for optimum operation. The possible 5 mk long-term loss in operating reactivity did not materialize; instead a gain of about 2.5 milli-k was observed. In addition, temperature coefficients were larger than the most pessimistic (lowest) value estimated. The second load was therefore adjusted by decreasing the ratio of enriched uranium to Li-Al to decrease the reactivity of the core by an estimated 9 mk maximum. The actual reduction observed in hot (operating) reactivity was 5 mk.

The ratio of enriched uranium to lithium-aluminum in the striped charge was changed as shown in the following table.

Table II

Ratio of Enriched Uranium to Li-Al

	<u>Inches U/Inches Li-Al</u>	<u>r_U^2/r^2 Li-Al</u>
1st Core	19.37	19.47
2nd Core	18.14	17.22

The second terms are reactivity-weighted ratios using slightly skewed flux distributions. The "inches of uranium" does not include endcaps. Actual charge make-up and patterns are described in Section B.2.c.

That the change would probably be less than 9 milli-k was recognized since the same data evaluation methods were used in making incremental adjustments as for the original load. The 5 mk observed change is open to interpretation since there is a difference in the shape of the observed hot long term gains curves between the two cores.

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b. Reactivity Transients

The net long-term hot reactivity change observed in the first E-N load subsequent to samarium saturation was a gain of approximately 3 milli-k; planning allowances had been made for a 5 milli-k loss. The difference was not unexpected since the latter figure represented the worst case for optimum operation. As described above, the E-N core makeup in the second load decreased the ratio of E to N.

A hot long term gain of 5 mk subsequent to samarium saturation was derived from second core data, a value approximately two-thirds greater than observed in the first core. It is possible that operating data from the second load is of poor quality; long term gains during the short operating periods showed only a very slight increase, the largest changes coming between equilibrium operating periods.

The abundance of water leaks made the determination of temperature coefficients difficult due to poor cold (startup) reactivity data. It appears, however, that the maximum over-all coefficient contribution (metal plus moderator effects at full level) was less than + 10 mk compared to a maximum of about +16 mk with a regular load. The metal temperature coefficient was measured during scram transient tests as about -0.0025 mk per MW¹¹; the metal coefficient thus provides a reactivity-damping effect of approximately 1/2% $\Delta k/k$ during the rise to operating level.

c. Nuclear Safety Evaluations

Preliminary hazards evaluations of the E-N parameters of the E-N lattice were consistent with Hanford criteria for speed and amount of poison control. Specific evaluation of supplementary control requirements in the reactor as loaded were carried out at the time of the test.¹² The latter study included the establishment of flux distribution parameters for blanket loadings to assure that their loading patterns were within total limitations; in actual practice the loading of sufficient fringe enrichment to exceed the derived control parameters would also result in higher E column heat generation rates than would be operationally desirable.

Primarily to aid in making reactivity coefficient evaluations, the horizontal control rod system at H Reactor was partially calibrated during the second E-N cycle.¹¹ The indicated rod values were up to 25 per cent lower than observed during a more complete calibration test in the normal load in 1957. A calculated 4 per cent reduction in local rod strength would be expected in the blacker E-N lattice. It is quite unlikely that the larger difference would be associated with the E-N core loading itself; numerous discrepancies were observed in the fitting of operational data to rod calibrations prior to the E-N loading.¹³ A check on this anomaly is being planned by a repeat calibration in the present natural uranium core surrounded by the blanket load.

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d. Operating Control Measures

This item refers to the flexibility requirements for control rod and supplementary poison control, and reactivity and control parameters of the E-N load.

As mentioned earlier physics calculations for the E-N loading had indicated the possibility of a long term reactivity loss of up to 5 mk between samarium saturation and goal exposure. The reactivity of the loading pattern was increased by this amount, and lithium-aluminum poison charges were to be loaded during early irradiation to compensate for the increased initial reactivity; the poison charges were then to be discharged as the reactivity decreased to maintain an optimum rod configuration. During the operation from May 3 to May 21, lithium-aluminum poison pieces were charged in 3 PCCF tubes to compensate excess reactivity within the first 250-tube block. These charges consisted of 13 pieces of 4-inch lithium-aluminum centered somewhat downstream. These charges were found to be weaker than required and were discharged on May 21 and replaced by charges containing either 29 pieces of 6-inch lithium-aluminum or 42 pieces of 4-inch lithium-aluminum. In addition seven more PCCF lithium-aluminum charges of similar makeup were loaded on May 21 for compensation in the rest of the striped load.

Operation following the May 21 outage required the loading of 40 to 50 splines during equilibrium. Since this semi-permanent investment in splines reduce their availability for startup control as well as decreasing reactor conversion ratio, and since no long term losses appeared, 8 E-N charges were discharged during the outage of June 27 and replaced by lithium-aluminum columns of identical makeup to the 10 already charged into PCCF channels.

The requirements for the control rod and supplementary control systems were most stringent during the xenon-free, samarium-free startups with the green E-N load. With variable goal discharging supplementary control requirements would not be nearly so great as during the initial startup following a block discharge. The spline system is ideal for providing additional poisoning capacity on occasion without invoking an efficiency penalty when they are not needed.

Direct comparison of control requirements of the cold green (samarium-free) E-N core to a cold green natural load is difficult because of poor cold reactivity data; however, the following listing of supplementary control usage indicates the order of difference. With a regular loading the maximum supplementary control requirements are about the equivalent of 24 PCCF poison columns plus 10 to 15 splines on samarium-free startups. With the samarium-saturated natural loading the requirements are about the equivalent of 24 PCCF columns. The initial startup with the first E-N core load required 85 splines and 24 PCCF poison columns and the startup level was limited to 400 MW. The startup with the (essentially) cold, green E-N second core required 63 splines and 24 PCCF poison columns, and the startup level was a more normal 1250 MW. Future E-N loads at H Reactor should have reduced requirements for supplementary control and lower startup losses.

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The effect of differences in short-term transients between E-N and natural loadings during subsequent startups would depend on the desired length of outage. The smaller E-N temperature coefficients would decrease the amount of supplementary control needed; however, the cold reactivity of the reactor would be greater. This means that the E-N minimum outage time would be less. With the short minimum outage time the xenon content of the reactor would be greater; startups at this time would have a greater xenon burnout rate than with a natural load, tending to nullify the smaller temperature coefficient transient. On extended outages, with an identical amount in xenon, the increased cold reactivity of the E-N reactor would more than nullify the effect of the change to a smaller over-all temperature coefficient. In essence, for short outages close to minimum downtime, about the same amount of supplementary control would be required with the E-N load, and on extended outages up to 2 mk more in supplementary control would be required. More definitive data will be available following the present H Reactor operation without the E-N core loading. This amount can be supplied conveniently with the spline system provided that an excessive amount of spline flexibility is not needlessly sacrificed to equilibrium control.

2. Engineering Parameters

The primary mechanical concern in carrying out the demonstration load was the performance of the fuel elements under higher-than-average severity conditions. Power distributions would also affect other reactor engineering operating parameters including flow distribution, graphite temperatures, and tube corrosion rates. Finally, the special charging and special pickup required an unusual degree of attention on a large-scale basis.

a. Flow and Power Distribution

(1) Radial Flux and Flow Distribution

The full E-N loading decreased the flattening efficiency (ECT) from 1625 ECT to 1500 ECT or by 8.3 per cent. Of this about 5.4 per cent occurred following the charging of the fringe blanket and about 2.9 per cent after the subsequent charging of the core load. The second effect is associated with the difference in core flux (to which blanket powers are normalized) in changing from a less "black" to a "blacker" core. The entire loss in flattening due to the E-N load was thus due to the relative loss in power in the fringe. The flattening efficiency within the flat zone remained approximately constant following the charging of the E-N load.

The variation in tube power between the five types of charges because of different numbers and positioning of E and N pieces was estimated to be less than one per cent in an equivalent flux.

That reorificing which was done during the E-N operation for bulk limit considerations was independent of E-N aspects. Reorificing of the fringe poison tubes to obtain greater central zone flow had been

contemplated but was not considered economical for only two E-N cycles because of the large outage time requirements.

(2) Longitudinal Distribution Aspects

Slight flattening of the axial flux by distribution of the lithium-aluminum pieces in a preferential manner within the E-N striped charge was attempted in order to reduce rupture potential. However, determination of the proper mixture of Li-Al and enrichment pieces for optimum pile reactivity was given priority during the two cycles of the demonstration test. Future loadings with a better known base reactivity match will permit increased attention to longitudinal flux distribution flattening considerations.

The actual effects of the E-N load on the axial flux shape at H were difficult to determine due to the problem of measuring flux distributions along enough tubes to obtain a representative over-all pile flux shape and due to the smallness of the actual change relative to measurement uncertainties. Skewing of the flux peak appeared to be from 0 to 1 slug farther upstream than before the E-N load and the peaking was reduced by an estimated 2 to 5 per cent.

The Li-Al slugs in the charge not only replaced power-generating material but also reduced the flux in adjacent enriched slugs. Since the striped charges were very nearly the same length as a natural uranium column, the specific power in the uranium had to be increased to maintain equivalent column power. This increase in specific power, measured from spline flux-traverse traces, averaged 7.3 per cent for striped charges with three Li-Al slugs and 8.2 per cent for those containing four Li-Al slugs. A further increase of 2.6 per cent must be made to compensate for the reduction in length of uranium due to the greater number of end caps in the E-N charges. Therefore, since the flattening was reduced by 8.3 per cent with the full E-N load the specific power in the E slugs had to be increased by 18.2 to 19.1 per cent to maintain equilibrium pile power level identical with that with the natural loading. Although this increase appears to be quite large, spike enriched charges in normal loadings generate up to 20 per cent higher specific power than adjacent natural uranium charges.

Mixer pieces were placed approximately ten natural uranium slug lengths from the downstream end. (All mixers used in the demonstration test contained natural uranium.) The recommended placing of mixers thirteen natural uranium slug lengths from the downstream end was employed in the second core charge. Positioning in their staggered pattern of the N slugs was considered to have precedence over specific mixer location. Placement of at least one enriched uranium slug between the nearest N slug and the mixer account for the slight variation in mixer positions seen in Figure 2.

Charge lengths were only one to three inches shorter than a natural uranium charge, thus affecting total bulk flow only slightly.

(3) Graphite Temperatures

The effect of the E-N load on the heating of the graphite was not significant. For a given column power the gas composition and graphite temperatures remained approximately the same.

(4) Tube Corrosion

The tube corrosion rate may be described by: /14/

$$CR = ke \cdot 0.057^t \quad \text{where } t = \text{tube temperature and an } r \text{ value of } 1.000 \text{ is assumed.}$$

The tube flow remained approximately the same since there was little change in charge length, and the outlet water temperature of the process tubes increased by 8.3 per cent (due to the drop from 1625 to 1500 ECT). The relative change in corrosion rate between the previous load and the E-N load can be described by

$$\text{Rel CR} = \frac{ke \cdot 0.057^{t_2}}{ke \cdot 0.057^{t_1}} = e^{0.057(t_2 - t_1)}$$

If $t_1 \approx 100C$, then $t_2 = 108.3C$ and,

$$\begin{aligned} \text{Rel CR} &= e^{0.057 \times 8.3} = e^{.473} \\ &= 1.61 \end{aligned}$$

This means that the corrosion rate of the aluminum process tubes in H Reactor was about 60 per cent greater with the E-N load at the same power level. It should be noted, 1) that optimization of operation with the E-N load might call for lower equilibrium levels, for reorificing of fringe tubes, etc., for reducing corrosion rates, and 2) that this difference would not be so high relative to a natural uranium loading with less radial enrichment initially.

Since the E-N load consists of two separate parts, the effect of each part on the tube corrosion rate should be considered. Of the total ECT loss, 65 per cent occurred with the fringe poison blanket and a natural uranium core. On this basis, the tube corrosion rates would be increased a calculated 36 per cent by a fringe blanket loading only or 18 per cent by a central-zone striped loading only (assuming equilibrium power levels are held constant).

The above figures apply specifically to H Reactor. Charging of an E-N load in a reactor with a smaller flat zone initially should affect the ECT less; consequently the effect on corrosion rates should be somewhat smaller.

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b. Fuel Performance

Rupture experience during the two E-N core cycles is compared in this section with predictions. In addition, estimates of rupture rates for future E-N loadings at an old reactor are given. As mentioned in Section A, bumper fuel elements were used in the striped core loading concurrent with their pilot irradiation in a natural uranium load at D Reactor.

(1) Discussion of Rupture Experience

The two core loadings were irradiated under a block discharge system; i.e., all of the 1050-1100 striped-core columns (for charge patterns see Figures 2 and 3) were charged and discharged together. Some deviation from a straight block discharge occurred; however, these deviations were minor. The first core loading was charged during May, 1961, and discharged during September. The second core was charged immediately after discharge of the first and was discharged in January, 1962. The average discharge exposures for the first and second core loadings were approximately 780 and 810 MWD/ton, respectively.

A total of 15 ruptures were sustained in H Reactor during the two core loadings. Fourteen of these failures occurred in striped charges. The remaining failure occurred in a 34-piece enriched bumper (O-III-EB) charge. A summary of the rupture experience is given in Table III.

(a) Prevention of Hot-Spot Failures

Examination of the failures experienced in the E-N load (see Table III under Failure Type) shows strong evidence that bumper slugs do, as predicted, reduce the side failure rate. No accelerated corrosion failures due to slug misalignment (side hot-spots) were observed. As a matter of interest, up to May 2, 1962, no side hot-spot failures had been observed in any bumper material. With the exception of mechanical damage failures all of the failures were end or hole failures. This also is as predicted; certainly, bumpers should not increase or decrease the end and hole failure frequency.

(b) Mechanical Damage Failures

Of the 15 ruptures sustained in the E-N loadings, four were mechanical-damage failures. The observed mechanical damage probably occurred during charging. Since the charging damage to a given slug (such as the O-III-EB) should be no different whether that element goes into an E-N striped charge or a uniform enriched or natural charge, it is believed that the mechanical damage failures were not directly related to the E-N load itself. The basis of this judgment is evidence from

TABLE III

E-N RUPTURE EXPERIENCE

<u>Tube Number</u>	<u>Date Failed</u>	<u>Charge Date</u>	<u>Exposure MWD/Ton</u>	<u>Average Tube Power</u>	<u>Failure Type</u>	<u>Lot Number</u>	<u>Charge Type</u>	<u>Downtime** Charged to Failure, hrs</u>	<u>Remarks</u>
2867-H	6/18/61	5/3/61	314	1135	End	KH-289-D	E-N	4.0	
2476-H	8/8/61	5/3/61	663	898	Hole	KH-289-D	E-N	16.2	2nd fail. in lot
1379-H	8/30/61	5/21/61	745	1030	Hole	KH-273-A	E-N	2.5	
2667-H	9/2/61	5/3/61	843	1001	Hole	KH-274-A	E-N	6.6	
1581-H	9/6/61	6/18/61	544	915	MD	KY-081-D	E-N	6.1	
1978-H	10/5/61	9/10/61	109	832	MD	KY-057-D	E-N	2.0	
2864-H	12/13/61	9/10/61	605	1074	MD	KY-087-D	E-N	9.1	
4177-H	12/23/61	8/3/61	1050	972	MD	KY-030-D	O-III-EB	1.8	
2872-H	12/28/61	9/2/61	760	1045	Hole	KH-830-D	E-N	5.9	
2667-H	12/31/61	9/2/61	793	1077	Hole	KH-831-D	E-N	4.0	
2580-H	1/7/62	9/10/61	746	1036	Hole	KY-097-D	E-N	4.4	
2576-H	1/9/62	9/10/61	738	909	Hole	KY-097-D	E-N	1.0	2nd fail. in lot
1968-H	1/16/62	9/10/61	890	991	Hole	KY-057-D	E-N	1.9	
2669-H	1/21/62	9/2/61	880	1016	Hole*	KH-830-D	E-N	3.2	2nd fail. in lot
3174-H	1/25/62	9/24/61	829	1081	Hole*	KH-831-D	E-N	1.5	2nd fail. in lot

Legend

End - Failure on the end of the fuel element probably caused by water penetration through a weld.

Hole - Failure on the internal surface of an I&E fuel element probably caused by water penetration through a weld.

MD - Failure probably caused by mechanical damage to the fuel element prior to irradiation.

E-N - Striped charge consisting of O-III-EB and Li-Al fuel elements.

O-III-EB - Charge consisting of O-III-EB Fuel elements.

* Examination in the Radiometallurgy Laboratory indicates the cause of failure was accelerated corrosion of the fuel element spire. Personal communication from R. Teats to R. D. Shimer.

** During CY-1961, the average downtime charged per failure was 10.9 hours.

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from non-bumper I&E element observations that such failures are not random, but that they are related to fuel element length and the individual reactor/^{15/}; the fact that all non-bumper failures have occurred in six-inch enriched slugs and predominantly at one reactor is probably due to differences in charging equipment.

Similarly, bumper mechanical-damage failures may be related to fuel element length and the individual reactor. As of May 2, 1962, there had been six bumper mechanical-damage failures. Five of the six had occurred in six-inch enriched material, all at H Reactor, whereas approximately two-thirds as many bumper E metal slugs have been charged to date at the older reactors as have been charged at H (~5000 columns altogether). With the improved bumper charging equipment and procedures being developed, mechanical-damage failures are expected to be essentially eliminated.

(c) Rupture-Prone Lots

Four of the 11 end-closure defect failures observed (hole and end failures) occurred after the parent lot had been designated "rupture-prone." A fuel element lot sustaining one end-closure defect failure is ordinarily discharged at the first opportunity as "rupture-prone".^{16/} During the second E-N block, four rupture-prone fuel element lots were left in the reactor to avoid charging replacement material near the end of the exposure cycle. This replacement material would have been discharged with the block because of 1) a large tube-replacement outage scheduled following discharge of the block, and 2) continuance of block discharge operation at H Reactor. Additional considerations were the process tube leak frequency and the type of fuel element failures expected. Because of the leak frequency, only short periods of operation were expected during January, 1962. Manpower for tube replacement was not available until January 27. The type of failures expected, hole and end failures, seldom stick in the process tube; therefore, the downtime required per failure is low. The failures experienced during January were, relatively speaking, inexpensive (2.4 hours per rupture). Three of the four rupture-prone lots subsequently sustained failures (2776, 2669, and 3174-H). An additional failure (2476-H) was sustained in rupture-prone material in the first cycle because of an error in charging records. These four failures do not have any particular significance in evaluating future E-N operation; certainly, they have no direct relationship to E-N operating conditions. Although it would not be possible to completely eliminate additional failures in "rupture-prone" fuel element lots, these failures should be, and have in the past, been comparatively rare.

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(2) Agreement with Expected Performance

A total of 15 ruptures were sustained during the two E-N core loadings at H Reactor. Four of these failures were caused by mechanical damage during charging. With the improved bumper-charging equipment and procedures being developed, these failures should be essentially eliminated. In any case, mechanical damage failures have no direct relationship to E-N core operating conditions.

Three or four of the remaining 11 failures would have been prevented by normal operating procedures. A fuel element lot sustaining one end-closure-defect failure is ordinarily discharged at the first opportunity as "rupture-prone." Four rupture-prone fuel element lots were left in the reactor to avoid charging replacement material near the end of the second core irradiation. Three of these lots subsequently sustained failures. An additional failure was sustained in rupture-prone material because of an error in charging records. Exclusive of the seven or eight failures that should ordinarily have had no direct relationship to the E-N loadings, the metal performance was quite acceptable; based on the observed operating conditions (power, outlet temperature, and exposure) and assessed metal performance during the two core loadings, about 10 failures were expected. Over-all, the enriched bumper fuel performed as anticipated.

(3) Future E-N Rupture Predictions

Figure 1 shows the estimated rupture performance in future E-N loadings; failures sustained in the 34- and 43-piece enriched charges supporting the Li-Al blanket loading were included in the analysis. This rupture estimate also includes failures not related to reactor operating conditions; a twenty per cent adjustment was used to account for these failures. Average goal exposures of 925 and 1000 MWD/ton were assumed for 34- and 43-piece blanket zone enriched charges, respectively. Block discharge, average inlet temperature (12C), a bulk outlet temperature of 93.5C and present assessed metal performance^{/17/} were used to calculate the expected ruptures. In general, slightly fewer ruptures would be expected if a variable goal plan were used. The use of present assessed metal performance is conservative, since, in the past, a strong improvement trend has been observed in metal performance.^{/17/} It should be noted, however, that the first two core loadings were irradiated under less severe conditions than used in obtaining Figure 1. The first core loading was irradiated during a period of high inlet temperature; consequently, the power level was relatively low although the bulk temperature was close to 93.5C. The second core load was operated considerably below a bulk temperature of 93.5C.

c. Charge-Discharge Effects

(1) Charge Makeup

The enriched uranium charges in the fringe blanket were of standard types. The entire load of productive material was either enriched

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FIGURE I

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EXPECTED RUPTURES
OLD REACTOR BLOCK E-N LOAD

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uranium or enriched lithium in aluminum. The enriched uranium (0.947% U^{235}) was fabricated as a 6.54 inch, I&E slug. The composition of the Li-Al (N or Boll) was 3.1 per cent by weight enriched lithium (38.5 per cent lithium-6). Two types of Li-Al slugs were used - a 4.2 inch I&E slug in the E-N or striped central zone charges and a 6.11 inch solid slug in the fringe poison blanket. Both types were used in "spike" poison charges in the flat zone of the reactor.

The same standard downstream dummy pattern (SDSDP) was used for E-N charges that is used with the natural uranium charges.

In the core charge enriched uranium and lithium-aluminum were mixed with slightly more lithium-aluminum in the center of the charge to flatten the axial flux. To preclude vertical "sheets" of lithium-aluminum through the reactor, these pieces were positioned differently between charges (see Figure 2) so that five different charges resulted. Each group of five types was arranged as compactly as possible in order to obtain the greatest homogeneity of the entire mixture. However, in order to minimize charging problems, only two different types of charges were placed in each row. (See Figure 3)

The column charge makeup of the various parts of the loading were as follows:

Fringe Blanket and Enrichment Ring

- 1st (outermost) Lattice Unit - 47 pieces, 6.11 inch, solid enriched lithium-aluminum
- 2nd, 3rd, 4th L.U. - 43 pieces enriched uranium
- 5th and 6th L.U. - 33 pieces enriched uranium plus a natural uranium mixer, or 34 pieces enriched uranium (except for some striped charges also in the 6th L.U.)

Flattened Zone

First Loading (Mixture of enriched uranium and 4.2 inch I&E enriched lithium-aluminum)

Charge

<u>Type</u>	<u>Charge (Beginning with Downstream End)</u>
1	SDSDP*: 2-E; 1-N; 9-E; 1-Mixer; 2-E; 1-N; 12-E; 1-N; 12-E; 1-N; 2-E
2	SDSDP: 4-E; 1-N; 8-E; 1-Mixer; 4-E; 1-N; 13-E; 1-N; 11-E
3	SDSDP: 7-E; 1-N; 5-E; 1-Mixer; 7-E; 1-N; 13-E; 1-N; 8-E
4	SDSDP: 9-E; 1-N; 3-E; 1-Mixer; 9-E; 1-N; 13-E; 1-N; 6-E
5	SDSDP: 11-E; 1-N; 2-E; 1-Mixer; 11-E; 1-N; 13-E; 1-N; 3-E

*SDSDP: Standard Downstream Dummy Pattern

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FIGURE II - E-N CHARGE MAKE-UP

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FIGURE III

E-N CHARGING PATTERN

■ Fringe poison Li-Al
G represents graphite stringers
Numbers specify the type of E-N charge
Blank tubes in numbered region are PCCF columns

□ Tubes used for PT IP-255-A-9-FP
□ Tubes used for PT IP-350-C

Second Loading

<u>Charge Type</u>	<u>Charge (Beginning with Downstream End)</u>
1	SDSDP: 2-E; 1-N; 12-E; 1-N; 1-E; 1-Mixer; 9-E 1-N; 11-E; 1-N; 4-E
2	SDSDP: 4-E; 1-N; 9-E; 1-Mixer; 2-E; 1-N; 11-E; 1-N; 11-E; 1-N; 2-E
3	SDSDP: 7-E; 1-N; 8-E; 1-Mixer; 2-E; 1-N; 11-E; 1-N; 12-E
4	SDSDP: 9-E; 1-N; 6-E; 1-Mixer; 5-E; 1-N; 11-E; 1-N; 9-E
5	SDSDP: 11-E; 1-N; 4-E; 1-Mixer; 7-E; 1-N; 11-E; 1-N; 7-E

Make-up of the striped charges was accomplished with special charge boxes and special templates. The templates were used to position the Li-Al pieces for a given type of charge, and the remaining holes were filled with E-metal. Arrows on the boxes showed the proper sequence for transferring the slugs from the box to the charging machine.

Some increase in efficiency could occur if the E-N load were on a continuing basis. More charge boxes could be obtained and the charges made up ahead of time, releasing more personnel for front face and pick-up chute work.

(2) Charge-Discharge Limitations

The discharge rate of the E-N charges was limited by Process Standard H-020 to meet nuclear safety standards. The metal was discharged directly into buckets in the discharge chute. The following rate figures/¹⁸ indicate observed limitations and potential improvements. The maximum discharge rate using this method is estimated to have been about 28 tubes per hour; this rate compares with about 65 tubes per hour with natural uranium standard I&E elements prior to the E-N load. However, the charge-discharge rate without the special pick-up problems would have been reduced with bumper slugs to a maximum of about 35 tubes per hour.

3. Over-all Operating Efficiency

Over-all operating efficiency may be broken into three major factors: Time-operated Efficiency, Level-operated Efficiency, and Flattening Efficiency. The following discussion covers the first two; flattening efficiency observations were discussed under item B.2.a.(1).

a. Time-Operated Efficiency Effects

The principle difference in time-operated efficiency which might be expected with an E-N loading would be in charge-discharge rates. During the

demonstration test irradiation, however, a much more significant contributor to outage time was the tube leak problem common to all of the older reactors during that period. A positive contribution to time-operated efficiency was made by the shorter minimum downtime and longer allowable scram time with the lower E-N temperature coefficients of reactivity.

(1) Charge-Discharge Time

The major difference in shutdown work at H Reactor between the E-N load and the natural load (pre-bumper slug) was the decrease in charge-discharge rate. The reduction in charge-discharge rate with the E-N load, as shown previously, was from about 65 tubes per hour to about 28 tubes per hour (maximum rates).

There were approximately 1080 E-N charges in the flat zone of the reactor. Prior to the E-N load there was practically no enrichment in the flat zone; hence the increase in outage time caused by reduced charge-discharge rates in the flat zone was directly proportional to to the number of tubes in the flat zone.

There were approximately 295 more enrichment columns in the fringe blanket than prior to the E-N load. There were also about 126 fewer uranium columns in the first lattice unit because of the additional Li-Al charged for the E-N fringe blanket. However the first lattice unit is discharged so infrequently it will not affect operating efficiency significantly. The 295 fringe tubes averaged approximately 70 per cent of the power of the flat zone tubes; therefore, assuming approximately the same goal exposure, it was assumed that the throughput per tube in the fringe is 70 per cent that in the flat zone. It was then calculated, using the data in Table II, the change in outage time per core loading as:

$$(1080/28 - 1080/65) + .70 (395/28 - 295/65) = 26.2 \text{ hours or } 1.1 \text{ days.}$$

Since there were two core loadings the effect on production efficiency was:

$$(1.1 \times 2 / 245 \text{ days}) \cdot 70 = 0.6\% \text{ (assuming 70\% production efficiency),}$$

or about 0.6 per cent decrease in production efficiency due to lower charge-discharge rates.

However, the indicated conclusion would be misleading. The introduction of bumper slugs in itself imposed a maximum charge-discharge rate of about 35 slugs per hour. On this basis the effect on production efficiency was a decrease of about 0.2 per cent, a more realistic value.

During the demonstration load irradiation there were six separate special pickups for conversion ratio determinations; each consisted of up to 24 charges to be picked up in three groups of eight tubes each. These six special pickups each required an average of about four hours or a total of one full day's outage time.

$$(1 \text{ day}/245 \text{ days}) \times .70 \text{ oper. eff.} \approx 0.3\%$$

Since a continuing E-N load would not be expected to require such special pickups, its production efficiency need not include such time as taken during the demonstration load for special pickup.

(2) Other Downtime Effects

The various changes in reactivity parameters as noted before resulted in a decrease in minimum downtime. This normal minimum downtime is in the neighborhood of 32 to 35 hours. With the E-N second core loading the minimum time was reduced to 23 to 25 hours. (The first core loading minimum downtime was even shorter, but base reactivity was excessive in that case.) A number of unscheduled outages occurred during which advantage was taken of the decreased downtime.

Precise calculation of the outage time saved with the E-N load is difficult. However, a conservative calculation shows that at least 39 hours were saved by earlier startups during 245 days of the full demonstration load.

For the same reason that the minimum downtime is reduced, scram recovery time is extended. Once again, the determination of whether a scram recovery was made with the E-N load that could not have been made with the natural load is difficult. The best estimate is that the equivalent of about 40 hours downtime was saved due to one equilibrium scram recovery and two nonequilibrium scram recoveries.

The total decrease in outage hours due to E-N reactivity effects over the E-N operation therefore was about 80 hours. This indicates an increase in production efficiency of $(80/(24 \times 245)) \times .70 = 1.0$ per cent directly attributable to E-N operation. It should be noted that the magnitude of this increase was large because of so many unscheduled outages, a condition hopefully non-representative of normal E-N operation.

The production efficiency of H Reactor with the E-N load should be compared with that of the other old reactors and with the change from the previous year when H Reactor operated with a standard loading. In the following table H Reactor is compared with averages for B, D, DR, and F and also with averages for B, D, and F (to ensure that non-representative data from DR Reactor operation after the Ball 3X drop during the summer of 1961 did not affect the relative status of each group).

TABLE IV

Production Efficiency Comparison

	Avg. 1961	Avg. Last Half 1961	Avg. 1960	1961 vs. 1960	Last Half 1961 vs. all 1960
B,D,IR,F	70.9%	66.2%	77.2%	-6.3%	-11.0%
B,D,F	71.0%	66.5%	76.9%	-5.9%	-10.4%
H	68.0%*	68.0%*	76.3%	-8.3%*	-8.3%

*Represents E-N operation June, 1961
through January 1962 only.

Table III indicates that the reduction in production efficiency experienced at H Reactor with the E-N load was generally experienced at the other reactors also and that the E-N load was not the primary cause of the reduced production efficiency. The general reduction in production efficiency resulted from deterioration of process tube quality during 1961. The E-N load probably had some accelerating effect on tube corrosion rates as shown previously, but most of the tube corrosion had already occurred prior to the charging of the E-N load. No attempt has been made to estimate a share of the tube leak outages to be attributed to the probable increase in corrosion rate under the more peaked E-N conditions.

b. Level-Operated Efficiency

The level operated efficiency of a reactor measures how well the reactor operates while it operates or essentially how closely it comes to its maximum power level during all the time it operates. Since the process of starting up a reactor requires limiting the reactor power to some extent, the level-operated efficiency is affected by both the startup efficiency and the frequency of startups. In measuring of the effect of a special loading the frequency of startups may be independent of the type of loading and dependent on other effects (such as tube leaks and rupture rates). Therefore a more basic measure of the effect of a loading on the level-operated efficiency may be in comparing the startup efficiency between two loadings.

The startup efficiency of a reactor is a measure of how rapidly equilibrium power level is reached following the first indication of criticality. Startup efficiency is affected adversely by higher equilibrium levels, limitations of the control systems, restrictions on rates of power changes, and higher pile reactivities.

During the E-N load the equilibrium levels and the rate-of-power-rise restrictions remained essentially the same. The E-N loading increased the basic pile cold reactivity. For up to 15 to 20 hours after minimum down-time this did not essentially change startup conditions, although it did

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change startup conditions for a given time after shutdown. Concurrent with the E-N loading the poison spline supplementary control system was added at the H Reactor to increase the flexibility of the control system. It is difficult to compare the startup efficiency between reactors since there is a difference between the number of control rods, equilibrium power levels, flattened zone sizes, etc. In general, it might be said that the H control system was essentially not limiting startup efficiency before the E-N loading nor during it except for startups following block discharges. The following table shows average startup efficiency as a function of equilibrium days lost per startup (based on a theoretically perfect instantaneous jump to equilibrium level at first indication).

TABLE V

	<u>Average Loss Per Startup (in Equilibrium Days)</u>			
	<u>1960</u>	<u>1961</u>	<u>Last Half '61 (E-N)</u>	<u>Last Quarter '61 (E-N)</u>
B,D,DR,F	-	.26	.23	.25
B,D,F	-	.25	.22	.24
H	.26	.29	.22	.25

Comparison with other reactors was omitted for 1960 since the 9-rod reactors were control-limited prior to acquiring the poison spline system. Because the first E-N core load was deliberately charged to have higher basic reactivity than the second core load, the first load should have had more adverse effects on the startup efficiency, if any could be charged to the E-N load. This does not appear to be true even if it is generally conceded that higher equilibrium power levels in the fourth quarter normally would tend to increase losses somewhat. Therefore it appears that the E-N load had little effect on the level operated efficiency.

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C. E-N Conversion Ratio Test Measurements

Plutonium and tritium recovery measurements have been made on a test basis of two small scale E-N block tests irradiated in 1957 and 1959 respectively and of the bulk material processed from the first cycle of the demonstration test 2-cycle core loading discharge in 1961. Variations occurred in fuel and target geometries and materials among the three tests and in the sampling techniques used to determine product yields. Because these differences were to a large extent characteristic of irradiation, separation and analytical practices for natural uranium loadings at the respective times, results in each case are compared to natural uranium load conversion efficiencies rather than to the other E-N conversion ratio test results directly.

The data and conclusions on the reported E-N conversion ratio observations to date have been prepared by W. S. Nechodom. The data are presented under two major headings - Small-Scale Tests and Large-Scale Test.

1. Small-Scale Tests

The 24-column test using solid and cored enriched fuel elements in the B reactor in 1957 and the 26-central-column and 10-fringe-column tests with I&E geometry in H reactor in 1959 provided irradiation experience and conversion ratio data prior to the full-reactor demonstration load. In order to provide a reasonably complete background of E-N technology to date in a single document, results of the previous tests are also summarized herein.

a. Small-Scale Test Results

Initial testing of E-N loads was undertaken in 1957^{19/}. This test consisted of two 12-tube block loadings of striped E-N charges in the B reactor flat zone. One block utilized solid enriched (.947 w/o U²³⁵) uranium fuel and the other block utilized cored enriched uranium fuel. The lithium-aluminum target elements were 3.1 w/o natural lithium (6.94 w/o Li⁶) in aluminum. The length ratio of enriched uranium fuel to lithium-aluminum target elements was 9.5/1 in the solid columns and 9.8/1 in the cored columns. The yields obtained from these loadings are summarized in Table 1 and compared to the natural uranium yields in the same geometries which were best-value data at the time of the test data analysis.

A second small-scale E-N test, utilizing I&E geometry fuel elements was initiated in 1957^{20/}. This test consisted of a 26-tube block loading of striped charges placed in the H reactor flat zone, of which the central four charges were utilized in the conversion ratio analysis. The lithium-aluminum target elements were 3.1 w/o lithium enriched to 38.5 w/o Li⁶. The length ratio of enriched uranium fuel to lithium-aluminum target elements was 15/1. The test also included a 10-column lithium-aluminum blanket loading supported by I&E enriched uranium columns. The two central blanket columns were used for tritium yield analysis, and one

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TABLE VI
FIRST AND SECOND SMALL-SCALE TEST RESULTS
(Conversion Ratios at 800 MWD/T, gms Pu/MWD)

	<u>Plutonium</u>	<u>Tritium</u> (gms equivalent)	<u>Total</u>
Solid striped E-N load	.735	.197	.932
Solid Natural Uranium	.8375	- -	.8375
Ratio			<u>1.113</u>
Cored Striped E-N Load	.6978	.203	.9008
Cored Natural Uranium	.8000		.8000
Ratio			<u>1.126</u>
I&E E-N Load With Enriched Lithium	.767	.204	.971
I&E Natural Uranium Load	.834		.834
			<u>1.164</u>

enriched column adjacent to the blanket was used for plutonium yield analysis. Results of this test were reported in references 4, 21, and 22. These data indicated a conversion ratio for an E-N striped loading (extrapolated to 800 MWD/T) of .767 gms Pu plus .235 gms Pu equivalent tritium per MWD. After correction of the tritium yield for the lack of control rods and poison columns in the test region during the irradiation, the conversion ratio became $.767 + .204 = .971$ gms Pu equiv./MWD at 800 MWD/T. These data were compared to accountability-table/23/ values for I&E natural uranium at the same exposure (after correcting for the lack of radial leakage in the flat zone) of .834 gms Pu/MWD, yielding a relative E-N conversion ratio compared to natural uranium conversion ratio of 1.164. The E-N conversion ratio appeared to be about one per cent higher than expected from normalized results of the 1957 reactor test; assuming about a two per cent uncertainty in each of the results, they were in excellent agreement.

Analytical results from the E columns supporting the blanket test indicated a plutonium conversion ratio of 0.679 gms/MWD at 800 MWD/T (extrapolated) in an E column immediately adjacent to the blanket. Since theoretical studies/24/ also indicate a reduction in conversion ratio at the edge of the reactor, (higher leakage), this value was accepted as reasonable. Theoretical studies also indicate that it would be conservative to obtain an average plutonium conversion ratio for the entire blanket-supporting enriched load

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by averaging the value at the outermost E-lattice with the value for an E-column in the flat zone of the reactor. Doing this gives an average plutonium yield of .723 gms Pu/MWD in the blanket-supporting enriched load.

Analytical results of tritium content from two lithium-aluminum columns in the blanket indicated that the tritium yield in a blanket loading was 1.407 gms Pu-equivalent per MWD in the adjacent enriched metal column which is equivalent to .200 gms Pu equivalent per MWD in all of the E columns supporting the blanket loading. Thus, the over-all conversion ratio in the blanket plus supporting E-columns was found to be $0.723 + .200 = .923$ gms Pu equiv./MWD.

Calculations of the power distribution in a full-scale E-N loading plus blanket loading indicate that 27.8 per cent of the pile power would be generated in the blanket-supporting enrichment. The over-all conversion ratio, weighted, for an E-N load plus blanket load, would then be .958 gms Pu equiv./MWD at 800 MWD/T. This value was compared to a normal load (including 15.1 per cent power generation in enriched columns) whose over-all conversion ratio at 800 MWD/T obtainable from accountability tables was .806 gms Pu/MWD. The E-N load conversion ratio was thus indicated to be about 18.9 per cent higher than that for a normal load (natural plus some enriched uranium) at 800 MWD/T exposure.

b. Theoretical Considerations

When the results of the second small-scale test were reported, an attempt was made to rationalize the apparent increase in the conversion ratio of enriched uranium in an E-N loading as compared to the conversion ratio of a single enriched uranium column in a natural uranium loading, and the decreased enriched uranium conversion ratio observed in the enriched columns supporting the blanket^{25/}. It was concluded, from consideration of the changes in the magnitude of the resonance neutron flux in the various loadings, that an increase in enriched uranium conversion ratio in an E-N load of about five per cent would be reasonable. A further adjustment, however, made necessary because the resonance capture probability is reduced in an E-N load by replacing uranium with lithium-aluminum, reduces this increase to about four per cent.

The observed increase, compared to "empirical" conversion ratios from then-existing accountability tables* was 8.5 per cent; a similar comparison to a theoretically-calculated conversion ratio for enriched-uranium^{26/} gives the increase as 4.9 per cent. Since other considerations (see below) indicated that the conversion ratio of flat-zone columns would be higher than the pile-averaged conversion ratio, the correlation between experiment and theory appeared good.

The effect of placing a uranium column next to a non-fissionable lithium-aluminum blanket was considered in a semi-qualitative theoretical manner. The resonance neutron flux in such a column relative to the thermal neutron flux, would be drastically reduced^{27/}, and hence the ratio of resonance neutron captures to thermal neutron captures to form plutonium would be

* IBID HW-50300 E. Accountability table values for enriched uranium conversion ratios were not (and are still not) based on experimental data, but were more-or-less adjusted to CPD-observed values of plutonium yield from all enriched uranium.

reduced. Since the column power level is determined essentially by the thermal neutron flux the result would be a reduction in conversion ratio. This reduction was estimated to be roughly ten per cent. Since the experimental data indicated that the conversion ratio in the columns adjacent to the poison blanket was 88.5 per cent of the conversion ratio in the core, the correlation between experiment and theory again appeared good.

Theoretical calculations of the tritium yield in the striped loading and in the fringe loading were also performed^{25/}. The purpose of these calculations was primarily to provide a method for extrapolating the experimental data obtained from the H reactor test to other types of reactors. However, the results of the calculations agreed reasonably well with experimental data; the calculated tritium yields in the striped and blanket loadings were 86 per cent and 103 per cent respectively of experimental values.

2. Full-Reactor Test

a. General

Based on the attractive results of the small-scale testing, a full-reactor E-N irradiation including a lithium-aluminum blanket was undertaken in 1961. ^{2, 3, 28/}. The major goals of the test were to demonstrate the operability and feasibility of a full-scale E-N loading, and to gather E-N and blanket conversion ratio data on a larger scale and in a more representative fashion than permitted by limited small block testing. Samples for product analyses consisted of one to two-ton batches of uranium selected from various regions of the reactor so that not only would the conversion ratio of the striped loading be determined but also the variation in conversion ratio within the blanket-supporting enriched loading. Test batches were also taken from the normal load prior to the loading of the blanket and striped core loading so that the base-point conversion ratio could be more accurately determined.

b. Analytical Test Results

The analytical program for determining the over-all reactor conversion ratio has not been completed. However, bulk results are available on the first E-N core. The tritium yield from the first core loading has been determined and reported^{29, 30/}. Since the identities of the small test batches selected from the first E-N core for plutonium yield analysis were inadvertently lost, a special run at the REDOX plant was substituted. In this special run, essentially all the enriched uranium discharged from the first E-N core loading was sequentially dissolved at REDOX. Analytical samples were taken from the solute as it was passed from the dissolver. The results of this analysis have also been reported^{31/}. Remaining plutonium analyses are required from small batch samples of E metal from the second core loading (six batches), the E metal supporting the blanket (15 batches), and of both natural and enriched uranium taken from the fringe of H reactor prior to charging the first E-N core. Tritium analyses are also required from the second core loading and from the blanket. The analyses and correlations described

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below are based on data already available.

The tritium yields from the first E-N core reported in References 29 and 30 were expressed as a reactor conversion ratio by converting the reported volumes of He^4 to moles, calculating the equivalent number of grams of plutonium, and dividing by the MWD generated in the loading. It had been hoped that the core could be divided into several separate batches upon which separate extractions could be made. Such data would tend to reveal the accuracy of the Savannah River extraction and analysis process. However, the identity of the lithium-aluminum pieces from the first coreloading was confused after they were discharged from the reactor, so that the Savannah River data represents the over-all tritium yield from the entire core loading. This yield, expressed as grams of equivalent plutonium, is 0.192 grams/MWD. The MWD includes the power generation of the natural uranium mixer pieces which were included in the striped charges. This value is based on the He^4 volume measurement, which is more accurate than the T^3 volume measurement. The value thus represents the undecayed tritium yield. (All tritium yields in this report must be corrected for decay before utilization in economic analyses).

The plutonium yields reported in reference/31/ were converted to reactor conversion ratio by multiplying the reported values of grams of plutonium per pound of uranium by the number of pounds of uranium in the sample and dividing by the number of MWD assigned to the sample by the Production Scheduling Operation. These data then yield an over-all average plutonium conversion ratio for the entire first E-N core loading of .7006 gms Pu/MWD at 829 MWD/T. Extrapolated to 800 MWD/T the value would be .7031 gms Pu/MWD.

c. Interpretation of Test Results and Extrapolation to Full-Scale Reactor Conditions.

The results reported in /b/ above indicate a disturbingly low over-all conversion ratio for an E-N striped load when compared to earlier test results, and to normal reactor loading conversion ratios. However, before such a comparison can be valid, the results must be placed on the same comparative basis. Several adjustments to the reported data are necessary before it can be meaningfully compared to accountability table values for a normal loading. The discussion under the following six sub-headings describes the calculational adjustments which were made.

1) Water-Mixer Pieces

Natural uranium water-mixer pieces (OIIWM geometry) were utilized in the E-N core loading, and are also used in a normal natural uranium loading. However, in an ideal E-N loading, E metal mixers would be used. This change will result in a slight increase in tritium yield (see Section 5 below). Using accountability table values for natural uranium pieces and the natural uranium mixer pieces, the average conversion ratio in an OIIIN natural uranium column is .8180 gms Pu/MWD at 800 MWD/T. Again, normal E columns, utilized in a natural uranium loading, contain natural uranium water mixer pieces; utilizing accountability table values for both the E pieces (OIIIE geometry) and the mixer pieces gives a conversion ratio of .7122 gms Pu/MWD at 800 MWD/T.

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2) Modernization of Fuel Designs

Since the inception of the E-N demonstration loading at the H Reactor both natural and enriched fuel designs have been modernized by altering their dimensions to provide better heat balance and corrosion characteristics. While experimental data are not available on the effects on conversion ratios of these changes, the effects have been computed; it is felt that a difference in values so calculated is very accurate, although both absolute values may be in error by several per cent. The change in conversion ratio in changing from an OIIN to an OIIIN fuel element is calculated to be + .16 per cent; the change in conversion ratio in changing from an OIIE to an OIIIE fuel element is calculated to be +1.4 per cent; both changes are due to small increases in the size of the fuel elements.

Since the plutonium yield in an E-N load is to be compared to that of the current normal load (OIIIN and OIIIE fuel elements), the plutonium yield in the E-N load must be increased to reflect the E metal geometry change; this change gives a plutonium yield for the E-N core loading of $1.014 \times .7031 = .7129$ gms Pu/MWD at 800 MWD/T.

Other fuel geometry changes are being considered; in particular, it has been proposed that the enriched fuel element length be increased to about ten inches instead of six. This change would reduce the relative volume of aluminum end-caps, and would tend to increase conversion ratios. This same change could, however, be applied to natural uranium fuel elements, and so no credit for further conversion ratio increases is applied.

3) CPD-IPD Book-Physical Inventory Differences

The large-scale extraction of the first E-N core load, the results of which are reported in reference^{/31/}, was essentially a head-end measurement of plutonium yield at the REDOX plant; samples were taken at the same location in the plant as normal head-end physical inventory samples, and analyzed in essentially the same manner. REDOX records during FY 1962^{/32/} show a consistent discrepancy between head-end and load-out inventories; the cumulative difference between the two points is 1.3 per cent, with the head-end lower than load-out. By themselves, these data might not be significant; however, a similar phenomenon is apparent at the PUREX plant, where the cumulative difference between head-end and load-out is 1.8 per cent, again with the head-end lower than load-out.

Furthermore, it is apparent that the load-out measurement of plutonium yield from natural uranium at PUREX corresponds almost exactly with IPD accountability book value. It is therefore necessary to adjust the apparent plutonium yield from the E-N load to a load-out value which would be on the same comparative measurement basis as the load-out value (and IPD book value) for natural uranium at PUREX. Adjusting the E-N conversion ratio to reflect this change gives a conversion ratio of

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$$1.013 \times .7129 = .7222 \text{ gms Pu/MWD at 800 MWD/T.}$$

Cumulative data from the REDOX plant load-out measurements during FY 1962 (thru December, thus excluding the E-N core load data) indicate that the load-out measurements of plutonium yield in normal, enriched uranium are two per cent lower than IPD accountability book value. Since normal E metal conversion ratios will be combined with natural uranium conversion ratios to establish a normal reactor conversion ratio, the enriched uranium conversion ratios must be reduced two per cent. Making this correction gives a conversion ratio for normal enriched uranium of .698 gms Pu/MWD at 800 MWD/T.

Close scrutiny of the REDOX plant accountability measurements during the months of January and February, 1962, supports the above conclusions. During these two months, when about 60 per cent of the uranium throughput was discharged from the E-N core loading, the differences between REDOX measurements and IPD back values indicated that the conversion ratio of E metal from the E-N load was between 2.5 and 3.0 per cent higher than the conversion ratio of normal E metal³³.

4) Blanket-Supporting Enriched Metal Conversion Ratio

Analytical data are not yet available on the plutonium yield from blanket enriched uranium. It is necessary to utilize earlier data from the small scale test results. In the 1959 test, a conversion ratio of .679 grams Pu/MWD at 800 MWD/T was obtained for enriched columns immediately adjacent to the fringe poison blanket. It was considered appropriate to average these data with flat zone data to obtain a conversion ratio for the entire enriched blanket load. First converting to OIIIE geometry, then averaging with the flat zone value (Section 3, above), gives an average conversion ratio for the blanket of .7054 gms Pu/MWD.

5) Tritium Yield Adjustments

(a) Core Loading

Analytical data indicate that the tritium yield in the first E-N core loading was .192 grams equivalent plutonium/MWD.

In the first E-N core loading, on the average, about 4.5 mk of excess reactivity was held in supplementary control to be used as a factor of conservatism to insure adequate reactivity over the cycle. In the second core loading this excess reactivity was eliminated²⁸. Since the tritium yield will be directly proportional to the amount of excess reactivity absorbed in productive lithium-aluminum target elements, and since it is calculated that about 100 mk of excess reactivity was so absorbed during the second core loading, the observed tritium yield in the first core may be adjusted by the ratio 100/95.5 to give an adjusted yield of .201 gms pu equiv./MWD.

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Operating experience with the H reactor E-N load, as outlined elsewhere in this report, definitely indicates that the E-N loaded reactor can be successfully controlled with less reactivity invested in control rods and supplementary controls than is feasible for a natural uranium-loaded reactor. It has been conservatively estimated that the reactivity invested in non-productive control systems can be further reduced by about four mk in future E-N loads. This will permit an increase in tritium yield of about four per cent to .209 gms Pu equiv./MWD.

Since the natural uranium mixers in the demonstration E-N core loading would be replaced with E metal mixers in an ideal E-N loading, the effect would essentially be to convert another small portion of the core from a natural to an E-N load. This would result in an increased tritium yield. In the demonstration E-N core loading, 2.17 per cent of the heat production was generated in the natural uranium mixers; if these are converted to E metal mixers, the tritium yield would be increased about the same amount; thus, a further adjustment in the tritium yield is made, to $1.02 \times .209 = .213$ gm Pu equiv./MWD.

(b) Blanket Loading

Tritium yield data are not yet available from the large-scale test loading. It appears reasonable to continue using the results of the small-scale blanket test, which gave a value of .200 gms Pu-equiv./MWD generated in the blanket-supporting enriched uranium columns.

6. Other Considerations

(a) Axial Blanket Loading

A further tritium yield amounting to about a .2 per cent increase in over-all reactor conversion ratio could theoretically be realized by adding lithium-aluminum target elements in front of and behind the standard column. This change would, however, result in a small increase in the peak-to-average power ratio within the column, an undesirable effect from the fuel rupture standpoint. The over-all effect requires further study, and no change in tritium yield is assumed from this source.

(b) Non-productive Fuel Channel Loadings

A survey of H reactor recent loading history indicates that, on the average, the reactor is operated with the equivalent of around 30 to 40 unproductive fuel channels (exclusive of supplementary control columns but including water to dummy segments in such channels). These channels reduce conversion ratio in two ways - by poisoning the reactor, and by softening the flux spectrum seen by surrounding channels. These channels include water channels, air channels, solid aluminum dummy channels, permanent graphite stringers, and channels loaded with bismuth-lead alloy elements

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(which are productive, but which contribute to the effect under discussion). It has been theoretically calculated that the conversion ratio in uranium-loaded channels surrounding such non-productive channels is reduced about 2-3 per cent. This results in a reduction in the over-all reactor conversion ratio of about .5 per cent.

This effect is slightly more pronounced in enriched uranium loadings than in natural uranium loadings, although no credit is taken for the difference in the present analysis. It is apparent, however, that if the other Hanford reactors averaged fewer non-productive columns than did H reactor, then since the over-all IPD-book natural uranium yields closely match PUREX output, the H reactor conversion ratio must be lower, and the other reactors conversion ratio higher, than book values. This effect cannot be accurately calculated, and thus is represented as a possible bias in the conclusions which follow. The bias might be as large as -.5 per cent in the conversion ratio of H reactor natural uranium.

All of the above-described effects have been applied to the data, and the adjusted best values of plutonium and tritium yields in each of the various types of loads are summarized in Table VII.

TABLE VII

FULL-PILE E-N LOAD CONVERSION TEST RESULTS
(Conversion Ratios at 800 MWD/T)

	Pu (gms/MWD)	T (gms Pu/equiv./MWD)	Total
E-N Striped Core Load	.7222	.213	.9352
Fringe Blanket Load	.7054	.200	.9054
Average (31.9 blanket, (68.1% core)	<u>.7168</u>	<u>.2089</u>	<u>.9257</u>
Natural Uranium (OIIIN Fuel)	.818	- -	.818
E Columns (OIIIE fuel)	.698	- -	.698
Average (15% E Columns, 85% Natural Uranium)			<u>.8000</u>

Ratio, over-all E-N plus blanket to over-all normal load = 1.157 + .03*
- .02

* See section d below.

d. Error Analysis

In most instances, an accurate calculation of the probable error associated with a given measurement is not possible, due to the small number of measurements available. This is the case for example, with the tritium yield from the first E-N core loading, where essentially one measurement of the entire core was made. In other instances, data indicate very high accuracy whereas such accuracy is considered fortuitous; for example, the excellent agreement between IPD book values and PUREX loadout measurements might be a temporary condition. In these cases, it is more conservative to assign "best guess" accuracy values to the various factors which make up the relative conversion ratio calculation. This has been done: the various factors are tabulated in Table VIII with their calculated effect on the over-all relative conversion ratio calculation and the estimated accuracy of the effect.

The accuracies in Table VIII are estimated at the 95% confidence level (2 σ). To compute the over-all probable accuracy, the following equation applies:

$$\sigma_{\text{overall}} = \sqrt{\sum \sigma_i^2}$$

Performing the calculation yields an overall probable accuracy of +1.93 per cent at the 95 per cent confidence level (over-all). This result assumes that no hidden bias exists, that all of the errors are random. Obviously, all known biases have already been accounted for in the analysis. The possible bias which might be introduced by the effect of non-productive columns which is discussed in 6.b above would make the over-all range of error about +2.5.

-2.0

TABLE VIII
ERROR ANALYSIS FACTORS

<u>Adjustment or Factor</u>	<u>Effect on Over-all Relative Conv. Ratio</u>	<u>Estimated Accuracy of Effect (95% C.L. = 2σ)</u>
Natural Uranium Conv. Ratio	- -	+ .85 %
Normal E Metal Conv. Ratio	- -	+ .15 %
Normal E B-PID Adjustment	.3%	+ .12 %
Natural Uranium Water Mixer	.1%	+ .005%
Normal E Metal Water Mixer	.03%	+ .0015%
E-N Pu Yield Measurement	- -	+1.06 %
E-N T Yield Measurement	- -	+ .14 %
E-N Pu Yield B-PID Adjustment	1.3 %	+ .6 %
E-N Water Mixer Correction	.2 %	+ .01 %
Modernization, OIIE to OIIIE for E-N	1.4 %	+ .5 %
E-N T Yield Adjustments	1.35%	+ .6 %
Fringe Blanket Pu Yield	- -	+ .75 %
Fringe Blanket T Yield	- -	+ .55 %
Overall*		+1.93 %

*See text.

e. Theoretical Discussion (Full-Pile E-N Load)1) Tritium Yield Calculations

No better technique for calculating tritium yields in striped and blanket loadings has become available since the reporting of the small-scale test results; the technique then utilized was briefly discussed in Section C.1.6. However, an independent approach, utilizing a similar technique, is now being utilized in the computer code Flex II* to calculate the equivalent grams of plutonium in any loading which are lost to non-productive controls; these data confirm the earlier calculations, predicting that about 0.22 gms Pu equiv. of T^3 could be obtained in a striped load per MWD.

2) Plutonium Yield Calculations

The weakness of all currently-available systematized methods of calculating the plutonium buildup in a lattice is that a uniform, infinite, lattice must be assumed. No accurate theoretical technique is apparent for considering the effect of foreign substances included in a lattice, or for considering discontinuities between types of lattices, such as occur at the boundary between the core and the blanket, or at the boundary between the enriched uranium and poison portions of the blanket. An attempt can be made to solve the problem in a pseudo-theoretical manner by combining experimental measurements of the relative flux in the lithium-aluminum pieces with theoretically-calculated physics parameters for the enriched uranium portion of the lattice. These data may be used in a neutron balance calculation to determine the number of atoms of product buildup per atom of ^{235}U burned out, or per fission. The neutron balance technique does suffer from the added disadvantage that it must be performed for a green lattice; accurate accounting for plutonium buildup and burnout is beyond the scope of a hand calculation. In this section, therefore, a neutron balance calculation is used to calculate the green atom-per-atom conversion ratio of a striped E-N charge compared to a natural uranium charge. The results of the neutron balance are then adjusted to ripe conditions by using computer-calculated buildup and burnout data for simple cases of enriched and natural uranium lattices.

Assumptions:Lattice Parameters:

	<u>OIE**</u>	<u>OIN**</u>
η	1.3807	1.2611
ϵ	1.0361	1.0370
f	.9033	.8948
P	.8866	.8840
k_{00}	1.14567	1.03444
σ_a^{235}	649.2	649.1
$(1 + \alpha)^{25}$	1.20624	1.20404
σ_a^{238}	2.7355	2.7342

* An improved version of the engineering-physics survey code Flex I, written by R. J. Shields.

** Taken from Flex I computer code data - hot, operating, green, parameters.

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When a striped E-N load is constructed, a portion of the enriched column is removed, and replaced with lithium-aluminum pieces. This results in a reduction in uranium inventory, and an increase in the resonance escape probability (P), which may be approximated by:

$$(1 - P)_{EN} = (1 - P)_E \times \frac{R}{1 + R}$$

where R is the length ratio of E to lithium-aluminum fuel. Correcting the OIIE P in this fashion yields a value of $P_{EN} = .8923$. Further assuming that:

$$\eta_{EN} = \eta_E$$

$$\epsilon_{EN} = \epsilon_E$$

$$k_{\infty EN} = k_{\infty \text{ Natural}}$$

we may solve for the effective thermal utilization of the enriched portion of the E-N lattice:

$$f_{EN(E)} = (k_{\infty} / \eta \epsilon P)_{EN} = .8104$$

It has been previously estimated that in this lattice the $f_{E-N}(\text{li-al}) = .0966 / 3^{1/4}$, so that the fraction of thermal neutrons lost in non-productive captures in the lattice is $1 - .0966 - .8104 = .093$.

The neutron balance may now be generated (Figure).

The neutron balance was based on the OIIE enriched uranium fuel because of the lack of detailed lattice parameters on the OIIIE fuel. The IPD accountability tables do indicate, however, that the initial conversion ratio of the OIIIE is 1.65 per cent higher than the OIIE. In the neutron balance, therefore, the plutonium yield for the E-N load is increased by 1.0165, to obtain .3767 atoms of Pu, a total of .4733 productive atoms, and 1.0111 atoms per fission. Assuming that the fast fission ratio (δ) is the same in both fuels, and that the heat of fission in both fuels is the same,* the ratio of initial conversion ratios in atoms (or grams) per MWD is thus $1.0111 / .98698 = 1.0244$. However, the major portion of the gain in E-N conversion ratio compared to natural uranium conversion ratio is obtained by the decreased product burnout in the E-N load compared to the natural load. Using accountability table values for OIIIE and OIIIN fuels, it can be determined that the average Pu yield in OIIIE fuel per MWD at 800 MWD is 89.75 per cent of the initial yield, whereas the average yield is OIIIN fuel at the same exposure is 85.83 per cent of the initial yield. The plutonium values in the neutron balance may therefore be further corrected to obtain for the E-N load .3381 atoms of Pu., .43469 total productive atoms, and .9386 productive atoms/fission. The natural uranium

*The heat of fission for I&E natural uranium fuels has been calculated to be 202.4 Mev, and for I&E E metal 201.1 Mev /35/. However, in an E-N striped load, about .2 $\text{Li}^6 (n, \alpha) \text{T}^3$ reactions occur per fission, each releasing about 4.5 Mev. The overall energy release per fission in the E-N load is thus about 202 Mev per fission, and the difference between the two loads is insignificant ($< .2$ per cent).

FIGURE 1
NEUTRON BALANCE

	E-N	Nat U
Atoms T ³	.09660	0
Atoms Pu	.37059	.46422
Total	.46719	.46422
Fissions	.46812	.47034
Atoms/Fission	.99801	.98698

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values may be corrected to .39844 atoms of Pu and .84713 productive atoms/fission, for an E-N/natural ratio of 1.096/1.

If we now return to the experimental results, we see that the adjusted E-N yield is 1.14 times the natural uranium yield. However, both experimental values must be re-adjusted to correspond to the conditions of the neutron balance; the water mixers must be removed, and, more importantly, the ratio must be adjusted for the fact that the E-N yield represents a flat zone conversion ratio only, whereas the natural uranium yield is pile-averaged, and includes the lower conversion ratio of fringe natural uranium columns. The difference between flat-zone and fringe conversion ratios is estimated to be about five per cent. Assuming that the flat zone natural uranium conversion ratio is five per cent higher than the fringe, it is about one per cent higher than the pile average. Removing the mixer piece decreases the conversion ratio of the E-N striped load about .2 per cent and increases the conversion ratio of the natural uranium load about .1 per cent; thus, the experimental ratio of E-N to natural conversion is readjusted to about 1.125, compared to the neutron balance ratio of 1.096. The experimental ratio is thus 2.6 per cent higher than the theoretical value. This difference is well within the limits of accuracy of the two techniques.

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References

1. HAN-78026, "Full-Pile E-N Test Authorization," AT Gifford to AB Greninger, 3-20-61
2. HW-66300 PT1, "Production Test IP-350-C, E-N Demonstration Load at H Reactor (Part I, Fringe Poison Loading)," RD Carter, 3-22-61
3. HW-66300 PT2, "Production Test IP-350-C, E-N Demonstration Load at H Reactor (Part II, Flattened Zone Striped Loading)," RD Carter, 4-4-61
4. HW-68184, "E-N and Blanket Conversions from Analysis of Tubes Irradiated at H," LW Lang and WS Nechodom, 2-10-61
5. HW-70357, REV 2, "Combined Operation Study - Blanket and E-N Loadings at HAPO," LW Lang, 2-9-62
6. HW-60309, "Proposed E-N Demonstration Load," LW Lang, 5-18-59
7. HW-55704, "Full Reactor E-N Load Nuclear Hazards Review," WE Niemuth and GF Owsley, 4-11-58
8. HW-59133, "Chemical, Physical, and Reactivity Changes in a Full Reactor E-N Meltdown," R. Nilson, 2-9-59
9. HW-69566, "Physics of E-N Load Compared to Natural Uranium Load at H Reactor," DI Monnie, 5-11-56
10. HW-67970, "PT IP-385-C, E-N Reactivity Matching Measurement," RD Carter, 12-30-60
11. HW-72411, "Final Report PT IP-416-C, Measurement of the Horizontal Rod Strength at H Reactor," AD Vaughn, 1-24-62
12. HW-70708, "Current Status of H Reactor Total Control," AD Vaughn, 10-15-61
13. Personal Communication, SL Stewart to RD Carter
14. HW-60123, "Forecast Routine Process Tube Replacements, FY 1960 to 1961, App. A," EF Fairweather et al, 4-22-59
15. HW-73392, "Rupture and Water Leak Memorandum," RW Reid, 4-16-62
16. HW-70446, "On the Discharging of Lots That Incur Ruptures Caused by End-Closure Defects," RR Bloomstrand, 7-20-61
17. HW-73007, "Rupture and Water Leak Memorandum, February, 1962," RW Reid, 3-14-62
18. Private Communication, A.N. Stinson to R.D. Carter
19. HW-58917, "Final Report PT IP-2-A, Enriched Uranium Conversion and Stability Test," LW Lang, 1-16-59
20. HW-60340, "PT IP-255-A-9-FP, E-N Demonstration Load," RE Hall, WS Nechodom, 5-8-59

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HW-73615
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References
(Continued)

21. HW-66775, SUP 1, "E-N Load Conversion Ratios - Measured Tritium Values,"
WS Nechodom, 10-7-60
22. HW-66775, SUP 3, "Semi-Final Report (Report No. 3) E-N Load Conversion Ratios,"
WS Nechodom, 12-22-60
23. HW-50300B, IPD Plutonium Conversion Tables," DW Hoba
24. HW-67809, "Validity of Homogeneous Method, Overbored vs. Present Lattice Conversion
Ratios," CE Bowers, 12-20-60
25. HW-68184, Appendix A - Ibid
26. Flex-I Engineering - Physics Computer Code - written by R.J. Shields
27. HW-68184, Appendix A, Example II - Ibid
28. HW-70740, "Recommended E-N Charge Modification (Second Cycle PT IP-350-C E-N
Demonstration Load)," RD Carter, 8-15-61
29. DPSP-62-1184, "Extraction of Tritium from Hanford Slugs," RM Girdler, 2-23-62
30. DPSOX-5008, "(Test Conclusion No. 2-412)," RE Patterson, 2-8-62
31. HW-73218, "Analysis of Three Batches of 105-H E-N Core Material, February, 1962,"
RP Ringle and TC Steinhauser, 3-30-62
32. HW-73307, "Irradiation Processing Nuclear Materials Control - March, 1962,"
JW Handshuh, 4-19-62
33. Private Communication from DW Hoba
34. HW-63668, "Conversion Ratio Incentive for Using Black Mint in an E-N Load,"
R. Nilson, 1-28-60
35. Private Communication, R. Nilson to WS Nechodom

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