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LARGE HETEROGENEOUS REFERENCE FUEL DESIGN STUDY FINAL REPORT

Hanford Engineering Development Laboratory

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HANFORD ENGINEERING DEVELOPMENT LABORATORY
Operated by Westinghouse Hanford Company
A Subsidiary of Westinghouse Electric Corporation
Prepared for the U.S. Department of Energy
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ABSTRACT

This report describes the Hanford Engineering Development Laboratory's (HEDL) participation in the Large Heterogeneous Reference Fuel Design Study (LHRFDS) and presents the characteristics of the four reactor designs developed in the study.

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I. INTRODUCTION

The fuel pin geometry used in the Fast Test Reactor (FTR) and considered for use in a prototypic small breeder installation has been evaluated through a comprehensive development program during recent years so that an extensive data base exists for extrapolation of the FTR fuel pin and geometry to larger installations. The objectives of the Large Heterogeneous Reference Fuel Design Study (LHRFDS) were to evaluate the performance capabilities of the reference FTR fuel pin in a 1200 MWe reactor plant installation and to identify a corresponding optimum core design. These goals were to be reached by several independent design agencies: Atomics International Division of Rockwell International (AI), Argonne National Laboratory (ANL), Advanced Reactors Division of Westinghouse (ARD), Fast Breeder Reactor Division of General Electric (GE), and HEDL. Combustion Engineering (CE) functioned as Task Coordinator to ensure consistent efforts by each of the other participants. HEDL's contribution to the study was to develop both homogeneous and heterogeneous core designs at both levels of constraint described below.

For purposes of the study, a reference fuel assembly was defined, corresponding to the FTR fuel pin with axial blankets placed in a 217-pin assembly. The reference assembly was used to define a reference core, which approximated FTR technology extrapolated to prototypic small breeder conditions. Performance indices of the reference core were provided by ARD during the study.

Two levels of constraint applied to the HEDL design efforts. The Level I designs were to adhere to the following limitations:

- Fuel pin identical to the reference fuel.
- Fuel assembly identical to the reference fuel.
- Fuel operating conditions identical to the reference core.
- Cladding conditions at two years no worse than the reference core at two years.

The restraints were relaxed somewhat for the Level II designs, corresponding to the following:

- Fuel pin identical to reference fuel except for active core and plenum height.
- Fuel assembly design open.
- Operating conditions no worse than reference core.
- Cladding cumulative damage fraction (CDF) at end-of-life less than 0.75.

Each core design was to be optimized according to an objective function combining pumping power costs, fuel cycle costs, and doubling time. In addition, improvement of safety parameters was to be given due consideration in the selection of a final core design.

The following sections of this report describe the HEDL contribution to the LHRFDS, including design procedures, final designs, and conclusions.

II. SUMMARY AND CONCLUSIONS

The core performance characteristics which could be obtained with the reference fuel pin design were strongly dependent upon the core design options which were exercised. The Level I ground rules allowed very little design latitude and severely restricted the range of performance characteristics which could be obtained. Virtually the only design option of any significance was core configuration. Consequently, the only improvements in the Level I homogeneous design relative to the reference core are those associated with core size effects. The Level I heterogeneous design allowed the average core fertile-to-fissile ratio to be adjusted, making improvements in core breeding characteristics and sodium void worth possible. However, these improvements were made at the expense of fuel utilization, fuel cycle costs, and plant capital costs (due to the larger core).

The freedom allowed for Level II designs produced large improvements in nearly all of the performance characteristics. Relative to Level I constraints, the significant parameters allowed to vary in Level II were:

- Reactor outlet temperature,
- Number of fuel pins per subassembly, and
- Active core height.

In combination, these parameters effectively halved the doubling time for both the homogeneous and heterogeneous designs.

Comparing the four designs, it is clear that those of Level II are superior overall due to greater design latitude which provided more opportunity for optimization. Level III designs, with the restriction of fuel pin cross-sectional geometry removed, would undoubtedly exhibit even better performance.

The primary benefit associated with the heterogeneous cores is reduced sodium void worth. The degree to which this is realized is dependent upon the thickness of internal blanket regions. It is noted, however, that there are many uncertainties associated with the irradiation performance of the internal blankets since there is no irradiation experience. In this design study, it has been assumed that existing fuel pin models accurately predict the performance of the internal blanket pins. Whether this assumption is valid remains to be seen. The irradiation history of an internal blanket pin is reciprocal (in terms of temperature and power) to the histories of those pins which form the data base for the performance models which were used.

III. CORE DESIGN PROCEDURE

A. GENERAL

HEDL's core design procedure, an iterative loop involving three stages of the design sequence, is illustrated in Figure 1. Stage 1 consists of general mechanical design and reactor physics scoping calculations to arrive at an initial core layout. Stage 2 consists of detailed reactor physics calculations for the core configuration defined in Stage 1. Based upon the detailed reactor physics results, a decision is made either to alter the design (Stage 1) or to go to Stage 3. Stage 3 consists of core orificing and detailed component mechanical design calculations. At the end of Stage 3, design adequacy is assessed. If the design is inadequate, the entire procedure is repeated until the design is acceptable.

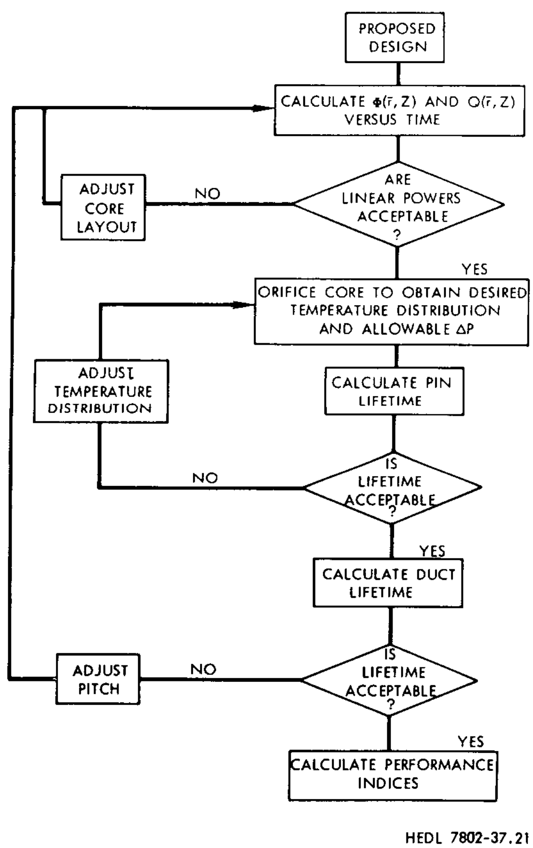

FIGURE 1. HEDL Core Design Procedure.

The initial core configuration is defined by whatever means are expedient, usually taking advantage of past experience. Often a quick-running, scoping code called HAREM (Hanford Advanced Reactor Evaluation Model) is utilized. A flowchart of this code is shown in Figure 2.

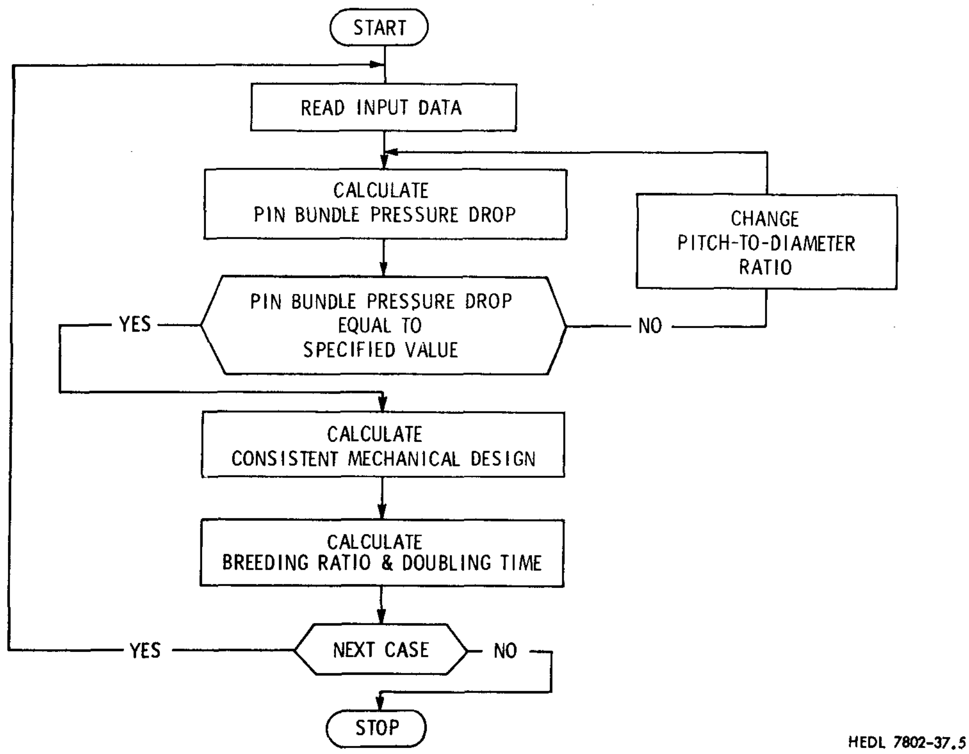

FIGURE 2. HAREM (Hanford Advanced Reactor Evaluation Model) Calculational Flow.

Input consists of design characteristics such as core temperature rise, reactor thermal power, fuel residence time, average pin power, pin bundle pressure drop, and pin size. For the LHRFD study these input quantities, for the most part, were obtained from the ground rules document. Output of the HAREM code includes number of subassemblies (S/A's), S/A spacing, duct geometry, and expected physics performance characteristics. A notable feature of HAREM is that duct pitch and wall thickness can be determined to match fuel residence time and to give optimum doubling time. This option was exercised on the Level II designs but not the Level I designs due to ground rule constraints.

The detailed reactor physics calculations in Stage 2 are performed using 2DB[1], which is a two-dimensional multi-group diffusion code with an isotope depletion module. This analysis provides time-dependent flux and power distributions and breeding performance for use as input to the mechanical design and economic analyses.

The next step of the design procedure is to orifice the core using the orificing scheme shown in Figure 3. This step uses the core orificing code ORIFIS, which distributes a specified coolant flow to obtain a desired subassembly outlet temperature distribution across the core. This distribution is subject to the discretion of the core designer and, for this study, varied depending upon the core design (see Section III, B, 2, page 18). Generally, the desired distribution is related to lifetime considerations and is selected using trial and error methods and iterating between the ORIFIS code and the lifetime codes.

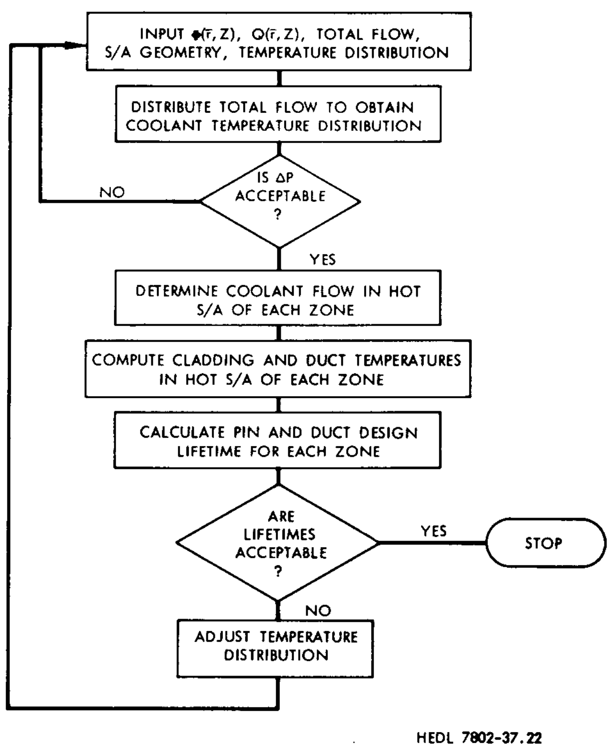

FIGURE 3. Calculational Flow for Establishing the Orificing Scheme.

Fuel pin lifetimes are calculated using the computer code SIFAIL, which uses the fuel pin model common to SIEX[2] and is illustrated in Figure 4. The code calculates fuel and cladding temperature, gas release rate, cladding stresses due to gas pressure loading, and cladding changes due to wastage, swelling, thermal creep, and irradiation creep. SIFAIL also calculates the cladding cumulative damage fraction based on stress rupture properties. This latter parameter was used as the fuel pin life-limiting parameter in accordance with the LHRFDS ground rules.

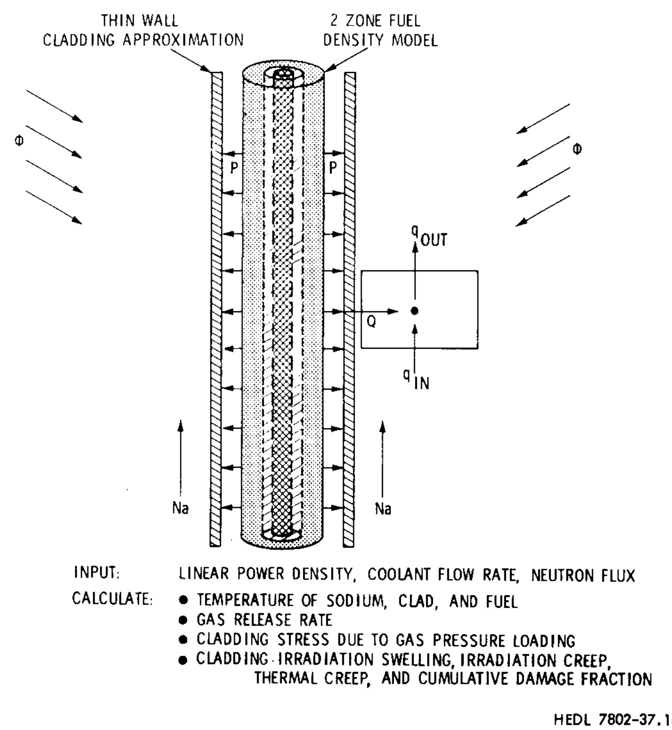

FIGURE 4. SIFAIL Fuel Pin Model.

The duct end-of-life is considered to occur when the duct outside diameter (OD) equals the lattice pitch, i.e., axial duct bowing is not included in the LHRFDS ground rule definition of duct lifetime. The time at which this occurs is calculated with the code DEFLECT, which uses the thin plate elastic deflection equations in conjunction with the method of Wire and Straalsund[3] for calculating irradiation creep. The model is schematically illustrated in Figure 5.

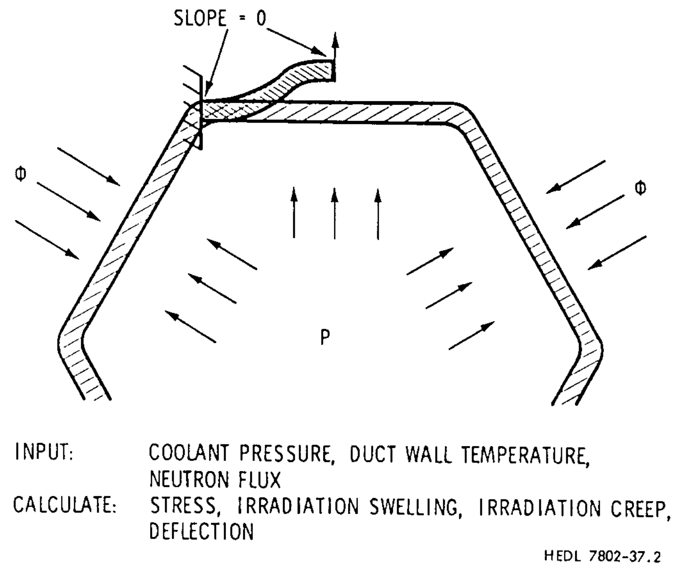

FIGURE 5. HEDL Duct Dilation Model (DEFLECT).

The computer code POROSTY is used to calculate bundle/duct interaction across the flats. The flux and temperature in the pin and duct are held constant at the worst case conditions, and no credit is taken for duct dilation due to irradiation creep. The degree of interaction across the flats is evaluated with respect to GE data relating experimentally measured interaction to actual pin-to-pin clearance.

B. LHRFDS PROCEDURE

The preceding section described the general design procedure and tools employed at HEDL for core design studies. The following sections describe the adaptation of those procedures and specific judgment decisions pertinent to the reactor physics and mechanical design of the homogeneous and heterogeneous LHRFDS cores.

1. Reactor Physics Design

The goal of the reactor physics calculations in the iterative loop is to provide the flux and linear power as a function of position for use in the thermohydraulics and fuel pin and duct lifetime calculations. In the reactor physics calculations, the core layout and subassembly designs (Level II only) for both fuel and blanket may be changed to meet the following non-mechanical design criteria:

- The linear pin power must be within the ground rule limits during the equilibrium cycle,
- The power distribution shall be reasonably flattened during the equilibrium cycle,
- The enrichment will be sufficient to maintain criticality during the entire equilibrium cycle with no excess reactivity at the end of equilibrium cycle ($k_{eff} = 1$), and
- The above criteria shall be met with the minimum number of subassemblies in order to minimize the fuel cycle cost.

When a converged design has been achieved--that is, one which satisfies the above criteria as well as the thermohydraulics, fuel pin lifetime, and duct lifetime criteria--the reactor physics calculations are also used to calculate the breeding performance, safety performance, and the fissile and heavy metal flow necessary to determine fuel cycle costs.

Cross section sets for use in the analyses were obtained from FTR Set 300[4] using 1DX[5], a one-dimensional diffusion code which collapses and self-shields multi-group cross section sets. FTR Set 300 is the reference data base used in FFTF design[6,7] and corresponds very closely to ENDF/B-III. It has been extensively verified through analyses of the FTR Engineering Mockup Critical experiments[8-18].

a. Homogeneous Cores

Using the reactor outlet temperature specified by the ground rules, a plant thermal efficiency of 36% was calculated for the Level I design using the ground rule efficiency relationship. For an electric power level of 1200 MWe, this efficiency corresponds to a gross thermal power of 3333 MW. It was assumed that pumping power would contribute 15 MW_t, so the reactor thermal power was assumed to be 3318 MW.

An iterative technique was employed to determine the enrichment corresponding to an end-of-cycle multiplication factor of 1.000 and to determine the enrichment distribution corresponding to an acceptable power distribution. The multiplication factor was affected by varying the total fissile mass. The power distribution was affected by varying the relative cross-sectional areas and fissile enrichments of the two core enrichment zones.

The iteration consisted of two initial depletion calculations selected to bracket the desired enrichment. A plot of effective multiplication factor at end-of-cycle versus beginning-of-cycle enrichment was constructed, and a linear interpolation was used to select a third enrichment estimate. If the third estimate did not meet the k_{eff} criterion, a curve was constructed using the three calculations, and a second interpolation, using either graphic techniques or a Lagrangian interpolation, was performed to establish the correct fissile content. The power distribution was then examined by zone to determine if the peak linear powers were acceptable. If the peak linear powers did not coincide to within 10%, a decision was made to vary either the ratio of the enrichments of the core zones or the ratio of the cross-sectional area of the two zones. In either case, some change in absolute magnitude of the enrichment was induced by any variation; if the power-balanced core did not meet the equilibrium k_{eff} criterion, the initial enrichment search was repeated.

Calculations were continued through this inner iteration until a converged design was developed from the reactor physics viewpoint. The characteristics of this design were then compiled for use in the mechanical design analysis (refer to Figure 6 for calculational flow).

Breeding performance of the core was evaluated by modeling a three-cycle burn using 2DB. A completely fresh core was loaded and burned at full power for 255.5 days, which corresponds to one year using the LHRFDS ground rule capacity factor of 70%. At that time, one-half the inventory of the core and axial blanket was discharged and replaced with fresh fuel and axial blanket material. The radial blanket was not refueled. This new core was then burned for an additional 255.5 days at full power, at which time portions of the core and axial blanket that had not been previously discharged were refueled. Again, the radial blanket was not refueled. At this point, the core material densities were quite close to those expected at the beginning-of-equilibrium cycle (BOEC). This configuration was burned for 255.5 days, thereby simulating an equilibrium cycle.

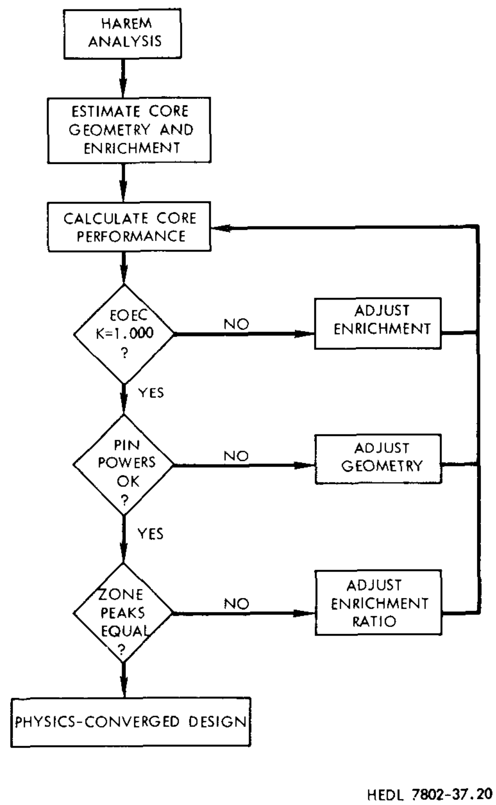

FIGURE 6. Physics Calculational Flow for the LHRFDS Homogeneous Cores.

The advantage of not explicitly refueling the radial blanket was that an equilibrium cycle could be obtained in three burns. A comparison of exposure accumulated by a five-batch radial blanket during annual refueling and the exposure accumulated by the radial blanket during this simulation showed that the equilibrium cycle burn approximated a five-batch blanket with respect to total power and fissile gain. Power and flux distributions in the blanket were not accurately modeled, however, so they had to be estimated.

The additional criterion of "lowest practical sodium void worth" was adopted at the beginning of the Level II design effort. A number of previously constrained variables were allowed to float in Level II, so parametric studies were required to follow the optimization path.

A major parametric study was undertaken for the Level II homogeneous design to evaluate the effects of active fuel height on whole-core sodium void worth. Sodium void worth was calculated for three core heights between 32.5 and 48.0 inches with all fresh fuel and the results were tabulated. Extrapolation of the results to +1$ sodium void worth indicated that a core height of approximately 20 inches would lower the void worth to that value. It was also noted that shortening the core--while holding the coolant pressure drop constant--increased the breeding ratio and decreased the doubling time.

The starting point for the Level II homogeneous design was chosen to be the waffle core, which was thought to incorporate some of the incoherence effects which give the heterogeneous core its low sodium void worth. The waffle core is diagrammed in Figure 7. A reactor physics analysis of this core was performed and the sodium void worth evaluated. It was found that the sodium void worth was roughly equal to that of the 32.5-in. uniform core; therefore, the waffle core was abandoned in favor of the final 24-in. pancake core configuration.

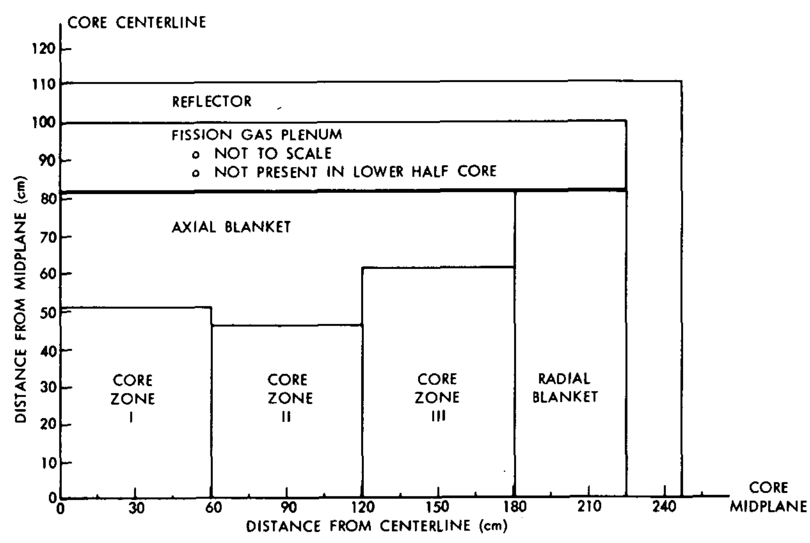

FIGURE 7. Schematic Illustration of the "Waffle" Core Concept.

The 24-in. active fuel height of the pancake core was selected as a compromise between the 20-in. core, which was thought to have a +1$ sodium void worth, and the 32.5-in. core, which represented the lower limit specified in the ground rules. Although EOEC sodium void worth calculations had not been performed for the pancake core, it was estimated that the final figure would be close to the +3$ value associated with the reference core, a value that results in a relatively benign LOF accident and is considered licensable[19-22].

In an attempt to attain a three-year residence time, the outlet temperature was reduced to the ground rule minimum. This change decreased plant thermal efficiency to 32%. For an electric power level of 1200 MWe and 16 MW_t of pumping power, the thermal power of the reactor was calculated to be 3734 MW.

The preliminary analysis of the pancake core indicated that the equal volume core zone concept was not the most effective way of arranging the enrichment distribution. Several geometries and enrichment splits were evaluated; the resulting final design had an outer enrichment zone with three rows of subassemblies. The power distribution across the inner zone of 498 fuel subassemblies was virtually flat, resulting in very good fuel utilization in that zone. Power gradients in the outer zone were steep; however, such gradients cannot be avoided at the core-blanket interface.

A fuel residence time of three years was used for the first iteration in the analysis of the pancake geometry. The mechanical design analysis, however, required that the residence time be reduced to two years because of excessive fuel pin diameter changes. Consequently, the second iteration produced a smaller subassembly gap. This resulted in slight improvements in breeding ratio and doubling time.

Greater improvements might have been realized if the coolant temperature had been increased to give a fuel pin lifetime of two years. This would have increased the thermal efficiency and, consequently, decreased the number of subassemblies and the core power. The fuel cycle cost would have been reduced accordingly. Alternatively, the subassembly dimensions could have been adjusted to allow the three-year residence time; however, the schedule prevented an exploration of these alternate paths.

b. Heterogeneous Cores

The approach used to develop a heterogeneous design was similar to--but more complicated than--that described previously. For the Level I design, the procedure was to use a single enrichment for all fuel assemblies and then to determine a proper internal blanket loading pattern which would satisfy the power distribution criteria. The calculation flow is shown in Figure 8.

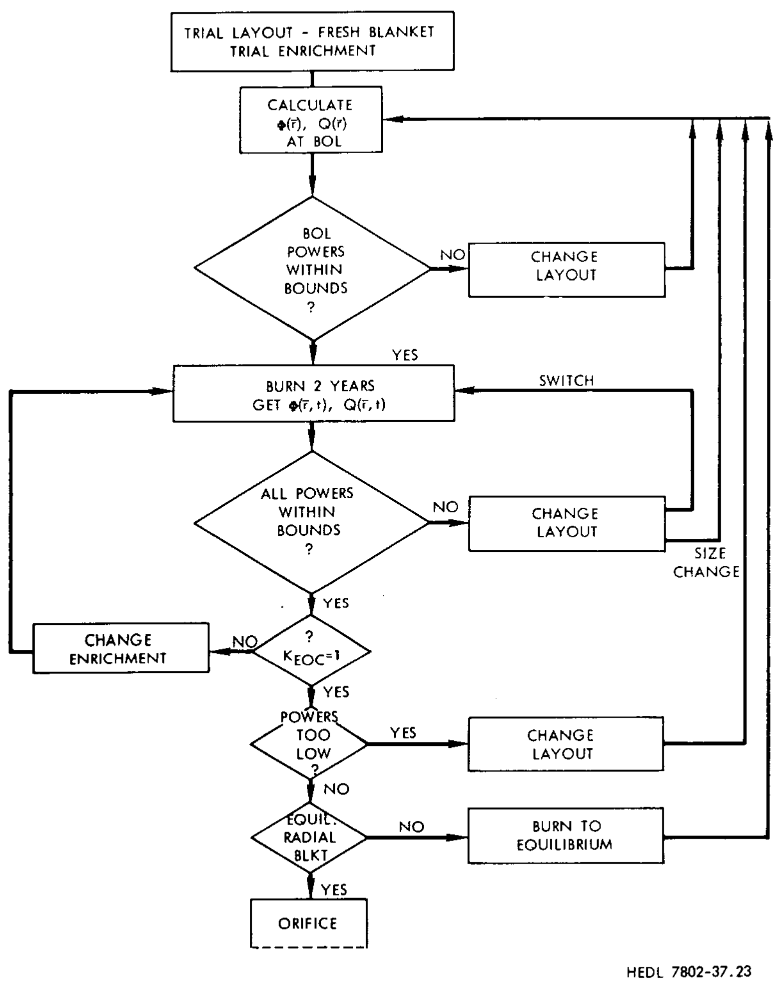

FIGURE 8. Physics Calculational Flow for the LHRFDS Level I Heterogeneous Core.

Starting with a trial core layout and enrichment, the flux and linear power distributions were calculated for the completely fresh core and blanket. If the peak linear powers were not within the ground rule limits, the core layout was modified either by adding assemblies or by moving blanket assemblies. Once satisfactory BOL linear power distribution was obtained, a burnup calculation was performed to determine the power shifts in the fuel during depletion and also the end-of-cycle internal blanket powers.

If either internal blanket or fuel linear powers exceeded the ground rule limits at any time during the cycle, the core layout was modified to make them acceptable. If the trial enrichment did not provide criticality for the whole of the cycle or provided excess reactivity at the end, it was changed and the cycle burnup repeated. After the enrichment change, if powers did not meet the ground rule limits, the layout was modified again to develop a design with the minimum number of assemblies consistent with pin linear power limits. Finally, an additional burnup calculation was made to bring the radial blanket to equilibrium composition, and the design process was repeated to assure compliance with the criteria.

The Level I heterogeneous design used a cartridge reload in which the fuel assemblies and all internal blankets were fully replaced at the end of each two-year cycle. This caused large power tilts in the fuel regions during the equilibrium cycle and resulted in a low average discharge exposure. To reduce the power swings, a two-batch system was used in the Level II heterogeneous design. Further, it was found that the power tilting in the Level I heterogeneous design was caused by the non-uniform internal blanket loading pattern. This caused the local conversion ratio to vary from place to place in the core; the differential local fissile content changed with burnup resulting in flux tilts and, hence, power tilts. To alleviate this condition, a different internal blanket loading pattern was used in the Level II design.

A rather uniform blanket loading pattern was used in the Level II heterogeneous design to minimize the variation in local conversion ratio across the core. To produce the desired power distribution, a higher enrichment was used in the outermost fuel--in a manner similar to the homogeneous designs. The blanket loading pattern consisted of single rows of internal blanket separated alternately by single and double rows of fuel. This pattern caused one problem insofar as the linear power was concerned--the core could not be enlarged radially in small increments while preserving the desired uniform loading pattern. In this case, it had to be enlarged by five rows at a time. This problem was solved by enlarging the core axially rather than radially in order to keep pin linear powers within limits.

For a 15-row core, a 4-ft active core height was required. A sodium volume fraction of over 40% was also required to keep the pressure drop within limits. Alternatively, one could have opted for a 20-row core for which core height would have been a little over 2 feet. A tighter pin pitch could then have been used, thus reducing the sodium void effect; however, the larger number of assemblies required in this case would have greatly increased the fuel cycle cost.

The calculational method used in Level II design is shown in Figure 9. It is very similar to the Level I method, the principal difference being that the enrichment ratio and core height, rather than core layout, were changed to meet the ground rule criteria.

An additional degree of freedom allowed by Level II ground rules was that of internal blanket redesign. If internal blanket pin powers were too high, the number of pins in a blanket assembly could be increased. Consequently, the internal blanket of the Level II design contained 127 pins.

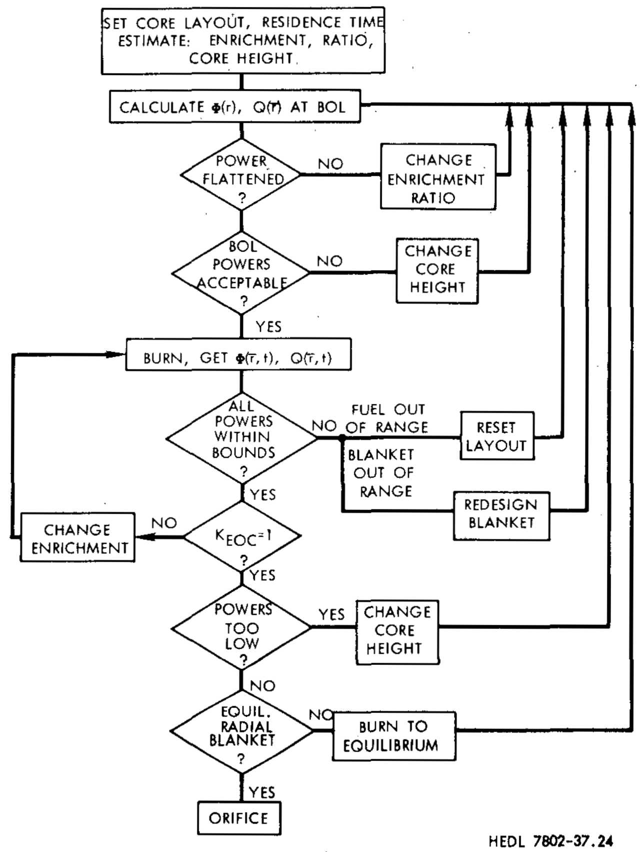

FIGURE 9. Physics Calculational Flow for the LHRFDS Level II Heterogeneous Core.

2. Mechanical Design

a. Homogeneous Cores

The following decisions and optimization criteria determined the orificing strategy and, consequently, the mechanical performance of the homogeneous cores. First, in accordance with established design procedures[23], 80% of the total reactor flow was passed through the core in the Level I designs. For Level II, 85% of the total flow was passed through the core. Second, a primary objective of the Level I homogeneous design was to optimize the inherent safety characteristics of the core. Core safety was enhanced by orificing in such a way that the highest subassembly power-to-flow ratios occurred in the lowest sodium void worth regions of the core--which would cause the most negative regions to void first during an LOF accident.

Third, the subassembly bundle pressure drops and the fuel pin and duct lifetimes had to be within ground rule specifications. Also, in the case of the Level II designs, a criterion of a minimum fuel pin pitch-to-diameter ratio of 1.17 was imposed to preclude designs for which no experimental data base existed.

Relaxation of constraints in the Level II ground rules allowed safety improvement through core shortening. Consequently, the core was orificed to provide equal fuel pin lifetimes throughout. Initially, the core was designed for a three-year fuel residence time by lowering the core outlet temperature to the minimum allowable level according to the ground rules.

Although the design met fuel pin lifetime ground rules, the combined calculated cladding swelling and irradiation creep was much higher than prudent design criteria would allow (>40% $\Delta D/D$). In the final core design iteration, the fuel residence time was reduced to two years. The core outlet temperature was held at the lower level.

b. Heterogeneous Cores

Generally, the same mechanical design considerations were applied to both the homogeneous and heterogeneous cores. One notable difference in the heterogeneous orificing procedure was that the flow available to the core (85% of total flow) was split to obtain ground rule specified power-to-flow ratios in both fuel and internal blanket assemblies.

In the Level I design, safety considerations were the overriding factor in determining the core orificing pattern. For this reason, the internal blanket assemblies were orificed to void before the fuel assemblies. It was later determined that the fuel assemblies should have been orificed to void first in an accident, but the Level II design was orificed without regard to voiding pattern because of its low void worth. In the Level II design the fuel assemblies were orificed to provide fuel pin lifetimes which were equal throughout the core. Internal blanket assemblies were similarly orificed to provide pin lifetimes equal to the internal blanket residence time.

3. Sodium Void Calculations

Sodium void worths were calculated for each of the designs using statics calculations and voiding only the flowing sodium in the active core region. The cross section set utilized for this analysis consisted of 12 energy groups--rather than the four used in the design analysis--and an additional cross section set was developed with sodium out of the core. Sodium void worths were calculated from the difference in k_{eff} obtained from two multi-group calculations. The difference in reactivity between the sodium-in and the sodium-out cases--with a fuel composition corresponding to the state of the core at that time--determined the whole-core sodium void worth. The calculational scheme is shown in Figure 10.

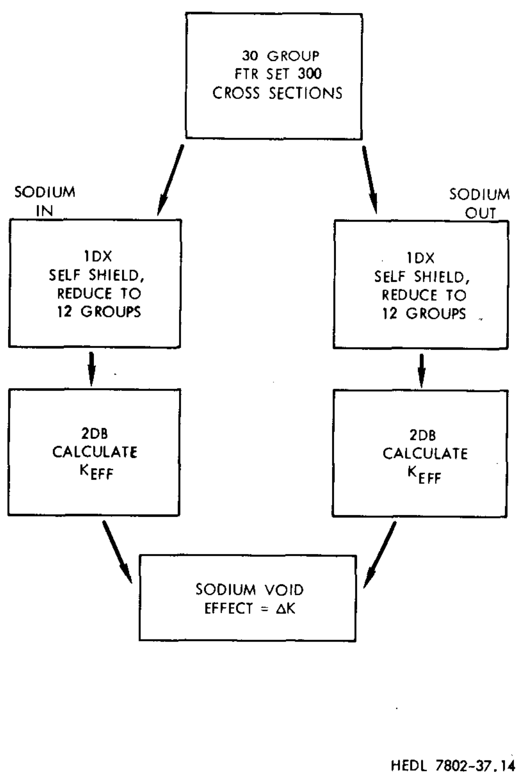

FIGURE 10. Sodium Void Calculational Flow.

The heterogeneous design exhibited a markedly lower sodium void worth--caused by voiding fuel assemblies--than did the homogeneous design. This was due to the contribution of the absorption, spectral, and leakage effects to the total sodium void worth. Figure 11 shows the small sample sodium void worth for a homogeneous design with the spectral, absorption, and leakage components specifically illustrated. The worth was dominated over most of the reactor by the spectral component with absorption providing a small additional effect. Only near the radial blanket did the leakage component become important.

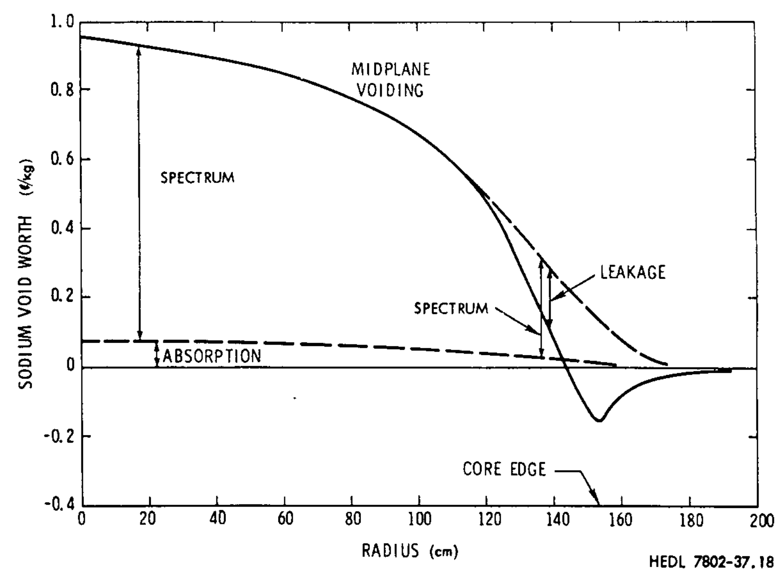

FIGURE 11. Small Sample Sodium Void Worth for a Homogeneous Core.

In contrast, the Level II heterogeneous design exhibited a strong leakage component not only near the radial blanket but also near the internal blankets. This effect is shown in Figure 12.

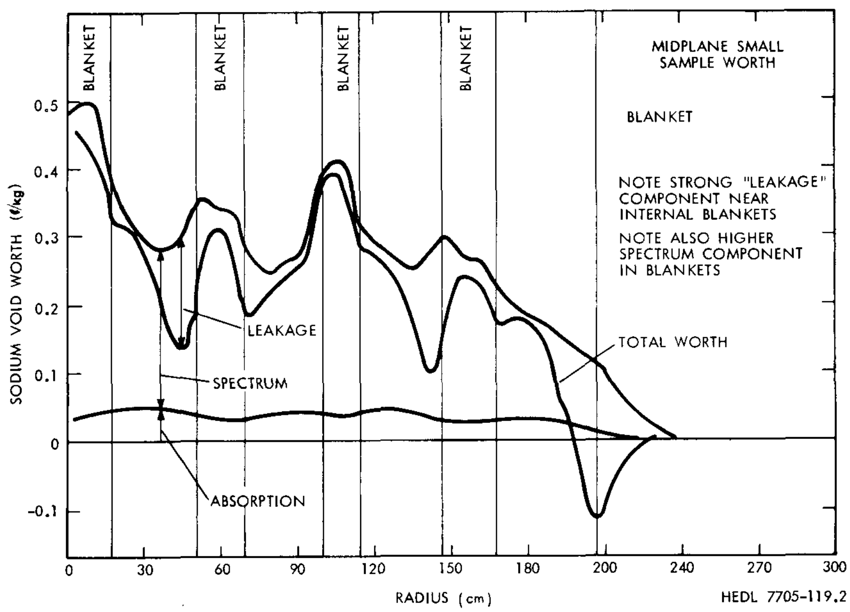

FIGURE 12. Small Sample Sodium Void Worth for a Heterogeneous Core.

The thicker internal blankets were strikingly more effective in reducing the sodium void worth than thinner internal blankets; the sodium void worth was smaller near the double row of blankets by a factor of more than two. In addition, this design exhibited a generally smaller spectral component than did the homogeneous design (due to the higher enrichment). The higher enrichment was beneficial in two ways:

> For a given pin design and linear power, the fluxes are lower, while
>
> The larger plutonium concentration produces an environment in which spectral hardening causes a reactivity reduction.

4. Cost Calculations

Fuel cycle costs were calculated for all four core designs using the data in the ground rules. The fuel cost model HPC, which is the cost model from ALPS[24], was used. All plants were assumed to have the same capital costs and operating costs, leaving only fuel cycle phenomena to produce a cost differential. Fabrication costs for the four designs are shown in Table I.

TABLE I

FUEL FABRICATION COSTS

Core Design	Costs ($K/Assembly) Driver Assemblies	Blanket Assemblies
LEVEL I CORES		
Homogeneous	32.4	29.1
Heterogeneous	33.2	29.9
LEVEL II CORES		
Homogeneous	32.8	29.5
Heterogeneous	39.1	35.2

IV. DESIGN DESCRIPTIONS

The designs developed in this study are considered to be reasonably well-matched in terms of thermohydraulic, mechanical, and reactor physics parameters, but they are not optimally matched. Given sufficient time, additional design iterations and parametric studies would be desirable to optimize the designs. However, it is thought that additional iterations would only serve to "fine tune" the designs and would not significantly alter the performance characteristics summarized below.

A summary design description of the Level I homogeneous core is given in Table II, and a complete description appears in ALSDAWG format in Appendix A. The core layout is shown in Figure 13.

TABLE II

LHRFDS LEVEL I HOMOGENEOUS DESIGN - SUMMARY OF MAIN PARAMETERS

GENERAL REACTOR DATA	
Reactor Power (MW_t)	3333
Gross Electric Power (MWe)	1200
Reactor Vessel ΔT (°F)	249
Reactor Vessel Outlet Temperature (°F)	965
Core Enrichment (Pu/Pu + U) (%)	
Inner Zone	18.1
Outer Zone	21.8
Total Fissile Inventory (kg-BOEC)[1,2]	3548
Total Heavy Metal (kg-BOEC)[2]	40278
Number of Subassemblies	
Drivers - Zone 1	354
Drivers - Zone 2	324
Control	43
Radial Blanket	420
Volume Fractions in Active Core	
Fuel	0.3240
Sodium	0.4239
Steel	0.2316
Control	0.0205
Number of Core Orifice Zones	8
Driver Residence Time (calendar yr)	2
Radial Blanket Residence Time (calendar yr)	6
Peak Discharge Exposure (MWd/kg)	100
Average Discharge Exposure (MWd/kg)	69
Peak Neutron Flux, E > 0.1 MeV (n/cm^2-s)	4.23×10^{15}
Peak Fluence, E > 0.1 MeV (n/cm^2)	1.87×10^{23}
Peak Cladding Temperature (°F)	
Nominal	1189
2σ	1338
Peak Linear Power (kW/ft)	
Nominal	10.8
3σ + 15%	14.2
GENERAL REACTOR DATA (continued)	
Sodium Void Worth ($)	
Fresh Core	+2.58
EOEC	+4.28
Doppler Coefficient	N/A
Breeding Ratio	1.17
Compound System Doubling Time (yr)	36.7
Maximum CDF	0.52
FUEL ASSEMBLY PARAMETERS	
Pins per Assembly	217
Duct Wall Thickness (in.)	0.120
Duct Outside Flat-to-Flat (in.)	4.575
Fuel Pin P/D, Compressed	1.24
Wire Diameter (in.)	0.056
Assembly Pitch (in.)	4.780
Nozzle-to-Nozzle ΔP (psi)	137
Maximum Mixed Mean Outlet Temperature (°F)	1044
DRIVER PIN PARAMETERS	
Fuel Height (in.)	36
Plenum Volume ($in.^3$)	1.287
RADIAL BLANKET ASSEMBLY PARAMETERS	
Pins per Assembly	61
Duct Wall Thickness (in.)	0.120
Duct Outside Flat-to-Flat (in.)	4.575
Pin OD (in.)	0.506
Pin P/D, Compressed	1.07
Assembly Pitch (in.)	4.780
Assembly Fueled Height (in.)	64
Plenum Volume ($in.^3$)	7.29
Peak Linear Pin Power (kW/ft)	15

[1] $^{235}U + ^{239}Pu + ^{241}Pu$

[2] Driver fuel region including axial blankets.

FCC = 3.18 mills/kW-hr
PP = 0.41 mills/kW-hr
OF = 3.18

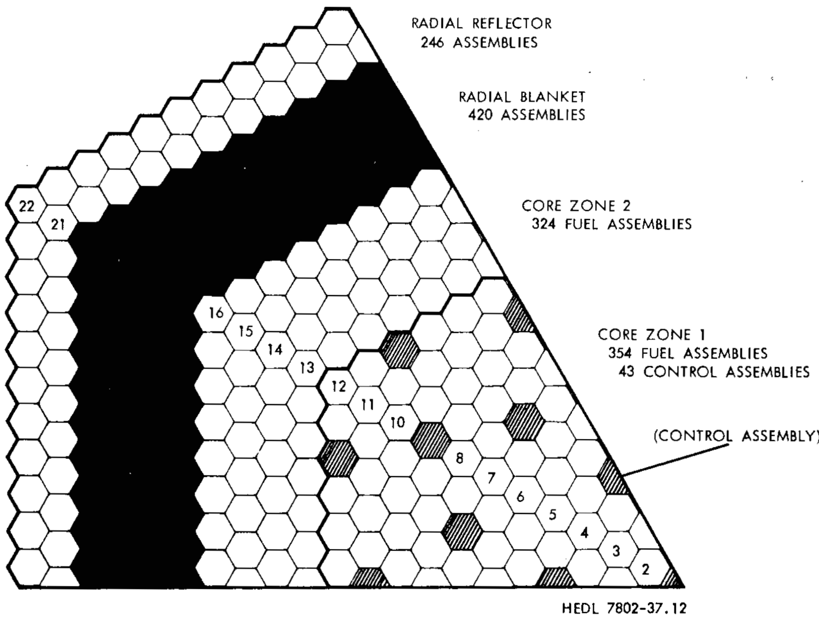

FIGURE 13. *LHRFDS Level I Homogeneous Core Map.*

The geometry is a scale-up of the reference core design with the exception of duct spacing. The core consists of 678 driver assemblies, 43 control assemblies, and 420 radial blanket assemblies. A unique feature of the design is that the core was orificed for safety. The core-wide subassembly mixed mean outlet temperatures, plotted in Figure 14, and the corresponding power-to-flow ratios were chosen to directly relate to the core-wide sodium void worth (Figure 15) in such a way that the core regions of lowest sodium void worth would void first in a loss of coolant accident because of the higher operating temperatures in those regions. The accident potential of this design therefore may be less than that indicated by the whole-core sodium void worth since the core is designed to void incoherently. The active fuel length for this design is 36 inches, and the end-of-equilibrium cycle sodium void worth is 4.28$. The fuel cycle cost is 3.18 mills/kW-hr.

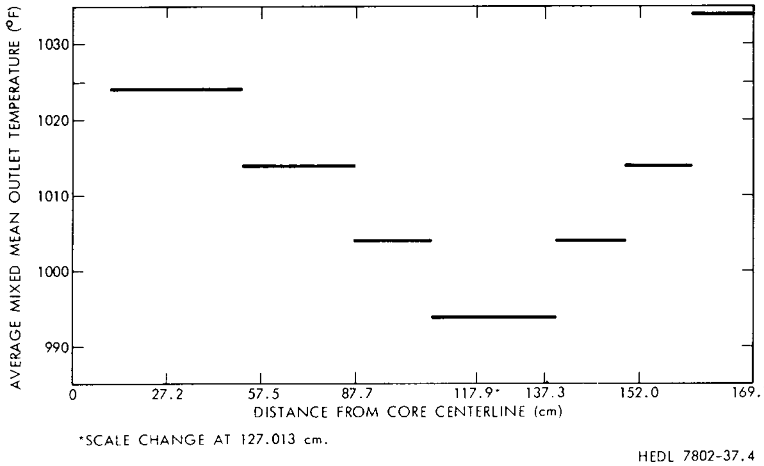

FIGURE 14. Assembly Coolant Mixed Mean Outlet Temperature at End-of-Life (EOL) for the LHRFDS Level I Homogeneous Core Design.

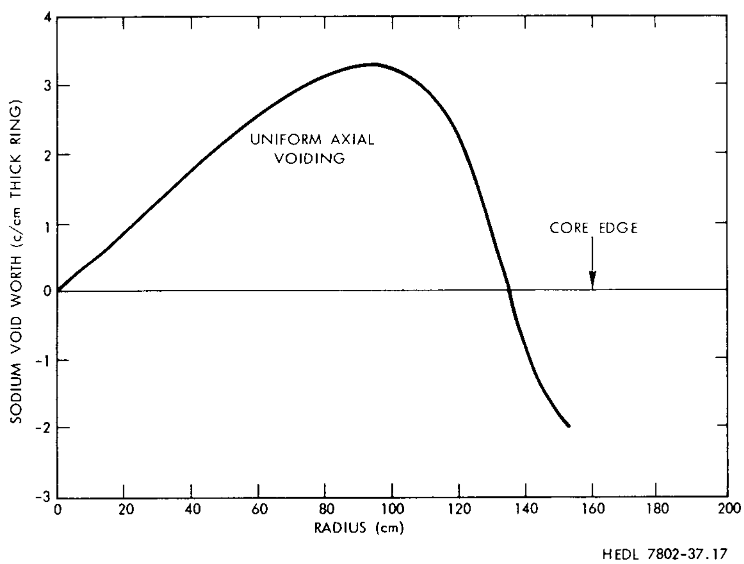

FIGURE 15. LHRFDS Level I Homogeneous Core Local Sodium Void Worth.

The Level I heterogeneous design is summarized in Table III and is described completely in ALSDAWG format in Appendix B. The core layout is shown in Figure 16 and consists of 684 driver assemblies, 313 internal blanket assemblies, 66 control assemblies, and 366 radial blanket assemblies. The significant feature of this design is the low sodium void worth--i.e., +1.61$ at the end-of-equilibrium cycle. The active fuel length for this design is 36 inches, and the fuel cycle cost is 5.53 mills/kW-hr. Thus a very significant improvement was made in the sodium void worth for the Level II heterogeneous design at a very significant increase in the fuel cycle cost.

TABLE III

LHRFDS LEVEL I HETEROGENEOUS DESIGN - SUMMARY OF MAIN PARAMETERS

GENERAL REACTOR DATA	
Reactor Power (MW_t)	3333
Gross Electric Power (MWe)	1200
Reactor Vessel ΔT (°F)	249
Reactor Vessel Outlet Temperature (°F)	965
Core Enrichment (Pu/Pu + U) (%)	29.9
Total Fissile Inventory (kg-BOEC)[1,2]	5353
Total Heavy Metal (kg-BOEC)[2]	41540
Number of Subassemblies	
Drivers	684
Internal Blankets	313
Control	66
Radial Blankets	366
Volume Fractions in Active Core	
Fuel	
Zone 1	0.3075
Zone 2	0.2940
Zone 3	0.3154
Zone 4	0.3236
Sodium	0.4215
Steel	0.2326
Control	
Zone 1	0.0384
Zone 2	0.0519
Zone 3	0.0305
Zone 4	0.0223
Number of Orifice Zones	
Driver Fuel Region	8
IB Region	4
Residence Time (calendar yr)	
Driver Fuel	2
IB	2
Radial Blanket	6
Peak Discharge Exposure (MWd/kg)	88
Average Discharge Exposure (MWd/kg)	62
Peak Neutron Flux, E > 0.1 MeV (n/cm^2-s)	2.62×10^{15}
Peak Fluence, E > 0.1 MeV (n/cm^2)	1.16×10^{23}
Peak Fuel Pin Cladding Temperature (°F)	
Nominal	1188
2σ	1356
Fuel Pin Peak Linear Power (kW/ft)	
Nominal	9.41
3σ + 15%	12.33

GENERAL REACTOR DATA (continued)	
Internal Blanket Peak Linear Power (kW/ft)	
Nominal	13.16
3σ + 15%	17.24
Sodium Void Worth at EOEC ($)	1.61
Doppler Coefficient	N/A
Breeding Ratio	1.29
Compound System Doubling Time (yr)	31.8
Maximum CDF	0.65
FUEL ASSEMBLY PARAMETERS	
Pins per Assembly	217
Duct Wall Thickness (in.)	0.120
Duct Outside Flat-to-Flat (in.)	4.575
Pin P/D, Compressed	1.24
Wire Diameter (in.)	0.056
Assembly Pitch (in.)	4.770
Nozzle-to-Nozzle ΔP (psi)	137
Peak Mixed Mean Outlet Temperature (°F)	1068
DRIVER PIN PARAMETERS	
Fuel Height (in.)	36
Plenum Volume ($in.^3$)	1.287
INTERNAL BLANKET/RADIAL BLANKET ASSEMBLY PARAMETERS	
Pins per Assembly	61
Duct Wall Thickness (in.)	0.120
Duct Outside Flat-to-Flat (in.)	4.575
Pin OD (in.)	0.506
Pin P/D, Compressed	1.07
Assembly Pitch (in.)	4.770
Assembly Fueled Height (in.)	64
Plenum Volume ($in.^3$)	7.29
IB Peak Linear Power	
Nominal	13.2
3σ + 15%	17.3
RB Peak Linear Power	
Nominal	11.2
3σ + 15%	14.7

[1] $^{235}U + ^{239}Pu + ^{241}Pu$

[2] Driver fuel region including axial blankets.

FCC = 5.53 mills/kW-hr
PP = 0.41 mills/kW-hr
OF = 3.87

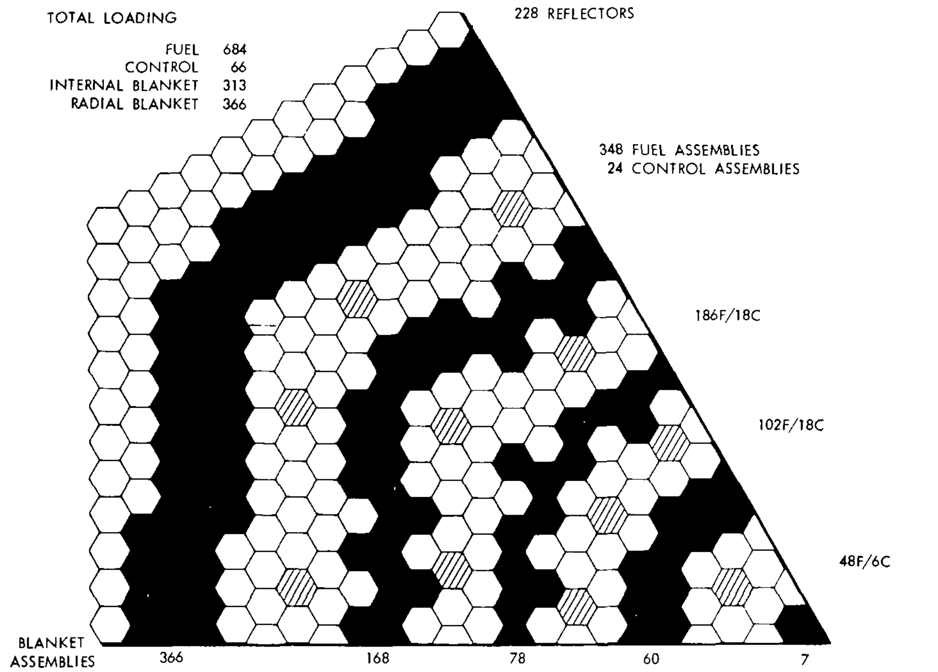

FIGURE 16. LHRFDS Level I Heterogeneous Core Map.

The Level II homogeneous design is summarized in Table IV. The complete description appears in ALSDAWG format in Appendix C. The core layout, shown in Figure 17, consists of 768 driver assemblies, 49 control assemblies, and 444 radial blanket assemblies. The large number of subassemblies in this design is a direct consequence of the 24-in. active fuel length which was selected to minimize sodium void worth. The sodium void worth of this design is 3.16$ at the end-of-equilibrium cycle--a significant decrease from that of the Level I homogeneous design. The fuel cycle cost for this design is 4.43 mills/kW-hr which is a significant increase over that of the Level I homogeneous design. The primary reason for this increase is, of course, the reduction in active fuel length from 36 inches to 24 inches.

TABLE IV

LHRFDS LEVEL II HOMOGENEOUS DESIGN - SUMMARY OF MAIN PARAMETERS

GENERAL REACTOR DATA	
Reactor Power (MW_t)	3750
Gross Electric Power (MWe)	1200
Reactor Vessel ΔT (°F)	300
Reactor Vessel Outlet Temperature (°F)	895
Core Enrichment (Pu/Pu + U) (%)	
Inner Zone	20.7
Outer Zone	26.2
Total Fissile Inventory (kg-BOEC)[1,2]	3665
Total Heavy Metal (kg-BOEC)[2]	58656
Number of Subassemblies	
Drivers	
Zone 1	498
Zone 2	270
Control	49
Radial Blanket	444
Volume Fractions in Active Core	
Fuel	0.3895
Sodium	0.3581
Steel	0.2281
Control	0.0243
Number of Core Orifice Zones	5
Driver Residence Time (calendar yr)	2
Radial Blanket Residence Time (calendar yr)	5
Peak Discharge Exposure (MWd/kg)	108
Average Discharge Exposure (MWd/kg)	79
Peak Neutron Flux, E > 0.1 MeV (n/cm^2-s)	4.57×10^{15}
Peak Fluence, E > 0.1 MeV (n/cm^2)	2.00×10^{23}
Peak Cladding Temperature, BOL (°F)	
Nominal	1093
2σ	1279
Peak Linear Power (kW/ft)	
Nominal	12.11
3σ + 15%	15.9

GENERAL REACTOR DATA (continued)	
Sodium Void Worth ($)	
Fresh Core	+1.27
EOEC	+3.16
Doppler Coefficient	N/A
Breeding Ratio	1.29
Compound System Doubling Time (yr)	17.2
Maximum CDF	0.06
FUEL ASSEMBLY PARAMETERS	
Pins per Assembly	271
Duct Wall Thickness (in.)	0.101
Duct Outside Flat-to-Flat (in.)	4.737
Fuel Pin P/D, Compressed	1.17
Wire Diameter (in.)	0.039
Assembly Pitch (in.)	4.875
Nozzle-to-Nozzle ΔP (psi)	∿ 100
Maximum Mixed Mean Outlet Temperature (°F)	974
DRIVER PIN PARAMETERS	
Fuel Height (in.)	24
Plenum Volume ($in.^3$)	0.804
RADIAL BLANKET ASSEMBLY PARAMETERS	
Pins per Assembly	61
Duct Wall Thickness (in.)	0.101
Duct Outside Flat-to-Flat (in.)	4.737
Pin OD (in.)	0.526
Pin P/D, Compressed	1.07
Assembly Pitch (in.)	4.875
Assembly Fueled Height (in.)	52
Plenum Volume ($in.^3$)	4.946
Peak Linear Pin Power (kW/ft)	17.2

[1] $^{235}U + ^{239}Pu + ^{241}Pu$

[2] Driver fuel region including axial blankets.

FCC = 4.4 mills/kW-hr
PP = 0.4 mills/kW-hr
OF = 2.73

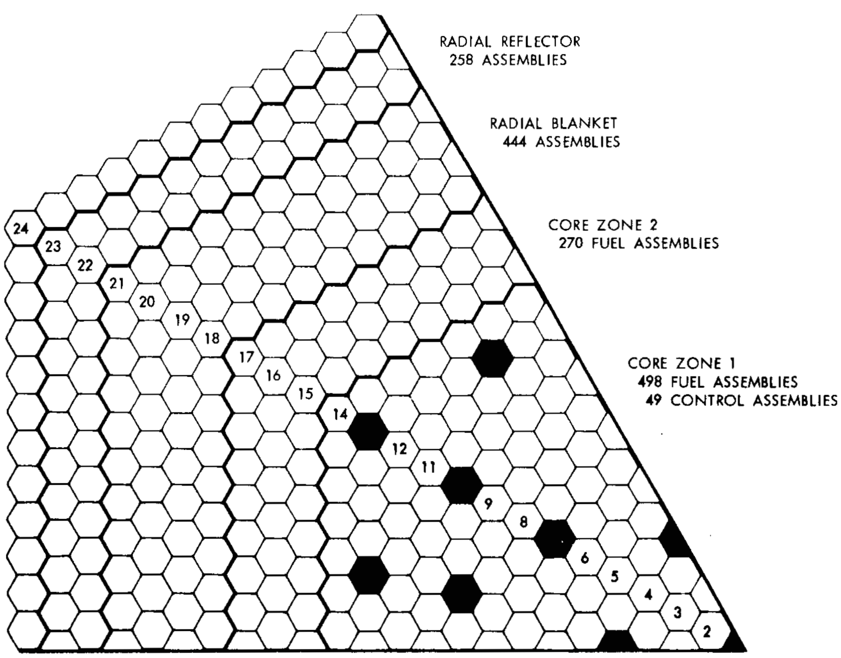

FIGURE 17. LHRFDS Level II Homogeneous Core Map.

The Level II heterogeneous design is summarized in Table V and completely described in ALSDAWG format in Appendix D. As shown on the core map, Figure 18, the core consists of 378 driver assemblies, 36 control assemblies, 217 internal blanket assemblies, and 288 radial blanket assemblies. The active fuel length is 48 inches, and the sodium void worth of this design is 2.53$ at the end-of-equilibrium cycle. The fuel cycle cost of this design is 2.95 mills/kW-hr. The Level II heterogeneous design has a fuel cycle cost lower than that of the Level II homogeneous design primarily due to the 48-in. active fuel length.

TABLE V

LHRFDS LEVEL II HETEROGENEOUS DESIGN - SUMMARY OF MAIN PARAMETERS

GENERAL REACTOR DATA	
Reactor Power (MW_t)	3750
Gross Electric Power (MWe)	1200
Reactor Vessel ΔT (°F)	300
Reactor Vessel Outlet Temperature (°F)	895
Core Enrichment (Pu/Pu + U) (%)	
Inner Zone	25.2
Outer Zone	29.3
Total Fissile Inventory (kg-BOEC)[1,2]	4319
Total Heavy Metal (kg-BOEC)[2]	60892
Number of Subassemblies	
Drivers	
Zone 1	162
Zone 2	216
Control	36
Radial Blanket	288
Internal Blanket	217
Volume Fractions in Active Core	
Fuel	
Band 1	0.3786
Band 2	0.3245
Band 3	0.3786
Band 4	0.3118
Band 5	0.3786
Band 6	0.3506
Sodium	0.4025
Steel	0.2188
Control	
Band 1	0
Band 2	0.0541
Band 3	0
Band 4	0.0668
Band 5	0
Band 6	0.0280
Number of Core Orifice Zones	
Fuel	7
Internal Blanket	6
Driver Residence Time (calendar yr)	2.5
Radial Blanket Residence Time (calendar yr)	6.25
Peak Discharge Exposure (MWd/kg)	122
Average Discharge Exposure (MWd/kg)	86
Peak Neutron Flux, E > 0.1 MeV (n/cm^2-s)	3.5×10^{15}
Peak Fluence, E > 0.1 MeV (n/cm^2)	1.9×10^{23}
Peak Cladding Temperature (BOL) (°F)	
Nominal	1200
2σ	1401
Peak Linear Power (kW/ft)	
Nominal	12.0
3σ + 15%	15.7
Sodium Void Worth (EOEC) ($)	+2.53
GENERAL REACTOR DATA (continued)	
Doppler Coefficient	N/A
Breeding Ratio	1.31
Compound System Doubling Time (yr)	19.0
Maximum CDF	0.25
FUEL ASSEMBLY PARAMETERS	
Pins per Assembly	271
Duct Wall Thickness (in.)	0.102
Duct Outside Flat-to-Flat (in.)	4.889
Fuel Pin P/D, Compressed	1.207
Wire Diameter (in.)	0.048
Assembly Pitch (in.)	5.095
Nozzle-to-Nozzle ΔP (psi)	∿ 170
Maximum Mixed Mean Outlet Temperature (EOL) (°F)	961
DRIVER PIN PARAMETERS	
Fuel Height (in.)	48
Plenum Volume ($in.^3$)	1.71
INTERNAL BLANKET/RADIAL BLANKET ASSEMBLY PARAMETERS	
Pins per Assembly	127
Duct Wall Thickness (in.)	0.102
Duct Outside Flat-to-Flat (in.)	4.889
Pin OD (in.)	0.365
Pin P/D, Compressed	1.110
Assembly Pitch (in.)	5.095
Assembly Fueled Height (in.)	76
Plenum Volume ($in.^3$)	4.79
IB Peak Linear Pin Power (kW/ft)	
Nominal	14.6
3σ + 15%	19.1
RB Peak Linear Pin Power (kW/ft)	
Nominal	11.0
3σ + 15%	14.4

[1] $^{235}U + ^{239}Pu + ^{241}Pu$

[2] Driver fuel region including axial blankets, but not internal blankets.

FCC = 2.95 mills/kW-hr
PP = 0.41 mills/kW-hr
OF = 2.24

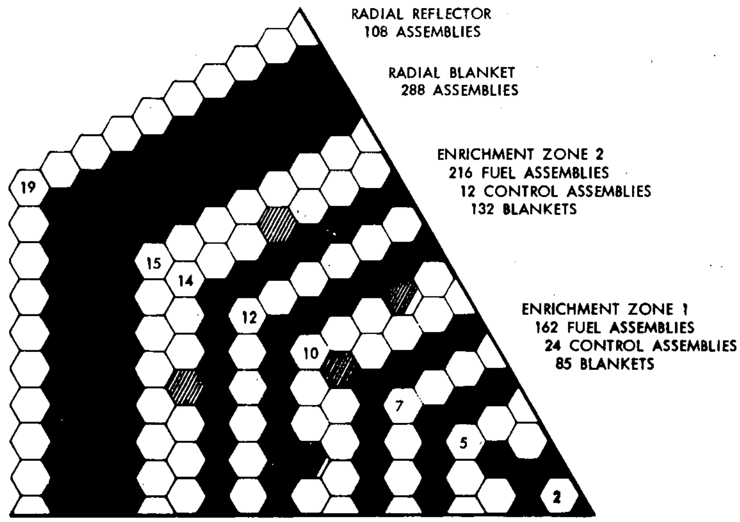

FIGURE 18. LHRFDS Level II Heterogeneous Core Map.

Performance characteristics of the four designs are compared in Table VI. The only advantage to the Level I homogeneous design appears to be minimum development costs. The Level I heterogeneous design has a low sodium void worth but has the disadvantages of both high fuel cycle costs and a long doubling time. The Level II homogeneous design has the shortest doubling time of the four designs. Fuel cycle costs, although high, might be improved by raising the operating temperature and consequently the thermal efficiency of the reactor. The Level II heterogeneous design has the highest core average discharge exposure, highest breeding ratio, lowest fuel cycle costs, lowest value of the optimization function, and appears to be the best design under the ground rules of this study.

TABLE VI

PERFORMANCE CHARACTERISTICS FOR LHRFDS DESIGNS

	Level I Homogeneous	Level I Heterogeneous	Level II Homogeneous	Level II Heterogeneous
Active Fuel Length (in.)	36	36	24	48
Core Average Discharge Exposure (MWd/kg)	69	62	79	86
Breeding Ratio	1.17	1.29	1.29	1.31
Compound System Doubling Time (yr)	36.7	31.8	17.2	19.0
Fuel Cycle Costs (mills/kW-hr)	3.18	5.53	4.43*	2.95*
Sodium Void Worth, EOEC ($)	4.28	1.61	3.16	2.53
Optimization Function	3.18	3.87	2.73	2.24
Residence Time (yr)	2	2	2	2.5

*The fuel cycle cost for the Level II homogeneous design is higher than that of the Level II heterogeneous because of the difference in the active fuel length. The active fuel length of the Level II homogeneous design was reduced to 24 inches in order to minimize the Na void effect.

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APPENDIX A

SPECIFICATION OF LHRFDS LEVEL I HOMOGENEOUS CORE DESIGN IN THE ALSDAWG FORMAT

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February 1977
(Revised April 1977)

1.0 CORE AND REACTOR DATA

1.1 Power Information

1.1.1 Plant Thermal Power, MW_t - 3318

1.1.2 Plant Electric Power, MWe - 1200

1.1.2.1 Net Electric Power, MWe 1185

1.1.3 Plant Capacity Factor versus Time - 0.70 (constant)

1.1.4 Power Split, Fraction of Total (MOEC)

1.1.4.1 Core Fuel - 0.9280

1.1.4.2 Axial Blanket - 0.0388

1.1.4.3 Radial Blanket - 0.0332

1.1.4.4 Internal Blanket

1.1.4.5 Control

1.1.4.6 Radial Shielding

1.1.4.7 Other

0

1.1.5 Average Linear Power (BOEC)

1.1.5.1 Core Fuel, kW/ft & (W/cm) - 7.115 (233.4)

1.1.5.2 Axial Blanket, kW/ft & (W/cm) - 0.250 (8.205)

1.1.5.3 Radial Blanket, kW/ft & (W/cm) - 0.670 (21.880)

1.1.5.4 Internal Fertile Assembly, kW/ft & (W/cm) - N/A

1.1.6 Fission Energy and Deposition, MeV/fission - 215, deposited locally

1.2 Temperature Information

1.2.1 Core Inlet Temperature, °F & (°K) - 716 (653)

1.2.2 Core Average Outlet Temperature, °F & (°K) - N/A

1.2.3 Core ΔT, °F & (°K) - N/A

1.2.4 Reactor Inlet Temperature, °F & (°K) - 716 (653)

1.2.5 Reactor Outlet Temperature, °F & (°K) - 965 (791)

1.2.6 Reactor ΔT, °F & (°K) - 249 (138)

1.2.7 Number of Core Orifice Zones - 8

1.2.8 Radial Profile of Assembly Outlet Temperature - Figure A-1

1.2.9 Core Orificing Criteria - Core orificed to provide voiding pattern which yields minimum accident.

1.3 Coolant Information

1.3.1 Peak Power Assembly Pressure Drop, psi & (kPa) - 95 (655)

1.3.2 Reactor Pressure Drop, psi & (kPa) - N/A

1.3.3 Primary System Pressure Drop, psi & (kPa) - 137 (946)

1.3.4 Flow Split, Fraction of Total

1.3.4.1 Core - 80%

$$\left.\begin{array}{ll} \text{1.3.4.2} & \text{Radial Blanket} \\ \text{1.3.4.3} & \text{Internal Fertile Assembly} \\ \text{1.3.4.4} & \text{Control} \\ \text{1.3.4.5} & \text{Radial Shielding} \\ \text{1.3.4.6} & \text{Other} \end{array}\right\} \text{20\%}$$

1.3.5 Total Coolant Mass Flow Rate, lb_m/hr & (kg/hr) - 1.498×10^8 (6.809×10^7)

1.3.6 Maximum Coolant Velocity, ft/s & (m/s) - 25.47 (7.76)

1.4 Geometric Information (see Figure A-2)

1.4.1 Core Height, in. & (cm) - 36 (91.44)

1.4.2 Axial Blanket Height, in. & (cm) - 14 (35.56)

1.4.3 Radial Blanket Height, in. & (cm) - 64 (162.56)

1.4.4 Axial Shield Height, in. & (cm) - N/A

1.4.5 Number of Core Enrichment Zones - 2

1.4.6 Number of Assemblies - 354/324/420

1.4.7 Equivalent Diameters[1], in. & (cm) - 100 (254.0)/134.78 (324.34)/169.55 (430.65)

1.5 Fuel Management

1.5.1 Refueling Interval, calendar days - 365

---

[1]Diameter of equivalent volume cylinder.

1.5.2 Fuel Residence Time, full power days (see Figure A-2) - 511

1.5.3 Blanket Residence Time, full power days (see Figure A-2) - 1533

1.5.4 Fuel Inventory, kg (see Tables A-I through A-III)

1.5.5 Fraction of Assemblies Replaced at Each Refueling

1.5.5.1 Fuel Assemblies by Enrichment Zone - 0.5/0.5

1.5.5.2 Radial Blanket Assemblies - 0.167

1.5.5.3 Interior Fertile Assemblies - N/A

2.0 FUEL ASSEMBLY DATA

2.1 Pins per Assembly - 217

2.2 Pin Pitch-to-Diameter Ratio - 1.24

2.3 Spacer Description

2.3.1 Wire Wrap Diameter (or grid spacer thickness & height), in. & (mm) - 0.056 (1.4)

2.3.2 Spacer Pitch, in. & (cm) - 11.9 (30.226)

2.3.3 Edge Ratio - 0.27

2.4 Overall Bundle Length, in. & (cm) - 114 (289.56)

2.5 Lattice Pitch, in. & (cm) - 4.78 (12.14)

2.6 Duct Inside Flat-to-Flat, in. & (cm) - 4.335 (11.011)

2.7 Bundle/Duct Clearance, in. & (mm) - 0.03 (0.765)

2.8 Duct Wall Thickness, in. & (mm) - 0.120 (3.05)

2.9 Interduct Gap, in. & (mm) - 0.205 (5.21)

2.10 Duct Material

2.10.1 Material Type - 316 SS, 20% CW

2.10.2 Swelling Properties - NSMH Rev. 5

2.10.3 Irradiation Creep Properties - NSMH Rev. 3

2.11 Duct Midwall Axial Temperature Profile[1]

2.11.1 Nominal (see Table A-IV)

2.11.2 2σ (see Table A-V)

2.12 Duct Wall Pressure Differential Axial Profile[1], psi & (kPa) - EOL

2.12.1 Nominal -

$$P_{psi} = \frac{x}{114} \cdot 55 \quad (0 \leq x \leq 114)$$

where x = distance from top of fuel pin bundle (in.)

$$P_{kPa} = \frac{x}{114} \cdot 379 \quad (0 \leq x \leq 114)$$

2.12.2 2σ - N/A

2.13 Neutron Flux Axial Profile[1] (E > 0.1 MeV), n/cm^2-s - EOL

2.13.1 Nominal -

$$\Phi = 4.23 \times 10^{15} \cos\left[1.98 \frac{(x - 18)}{36}\right] \quad (0 \leq x \leq 36)$$

where x = distance from bottom of fuel (in.)

Φ = flux

2.13.2 2σ - N/A

3.0 FUEL PIN DATA

3.1 Fuel Parameters

3.1.1 Fuel Type (oxide, carbide, nitride) - Oxide

3.1.2 Stoichiometry (O/M, C/M, N/M) - 1.94

3.1.3 Plutonium Content (Pu/Pu + U) - 0.1820/0.2184

3.1.4 Fuel Form (powder or pellet) - Pellet

3.1.4.1 Pellet Diameter, in. & (mm) - 0.1935 (4.9)

3.1.4.2 Pellet Dish and Chamfer Dimensions, in. & (mm) - 0 (0)

---

[1]Reported for design limiting duct.

3.1.4.3 Pellet Inside Diameter, in. & (mm) - 0 (0)

3.1.4.4 Pellet Density, g/cm^3 - 10.03

3.1.5 Fuel Smear Density, %TD - 85.5

3.2 Cladding Parameters

3.2.1 Cladding Outside Diameter, in. & (mm) - 0.23 (5.84)

3.2.2 Cladding Wall Thickness, in. & (mm) - 0.015 (0.381)

3.2.3 Diametral Gap, in. & (mm) - 0.0065 (0.16510)

3.2.4 Cladding Material

3.2.4.1 Material Type - 316 SS, 20% CW

3.2.4.2 Swelling Properties - NSMH Rev. 5

3.2.4.3 Irradiation Creep Properties - NSMH Rev. 3

3.2.4.4 Stress-Rupture Properties - LHRFDS Ground Rules

3.3 Stresser Sleeve Parameters - N/A

3.3.1 Sleeve Outside Diameter, in. & (mm)

3.3.2 Sleeve Wall Thickness, in. & (mm)

3.3.3 Fractional Perforation of Sleeve

3.3.4 Sleeve Material

3.4 Equivalent Plenum Volume, $in.^3$ & (cc)

3.4.1 Top Plenum - 1.286 (21.082)

3.4.2 Bottom Plenum - N/A

3.5 Bond Type - N/A

3.6 Fuel Pin Linear Power Axial Profile[1], kW/ft - EOL

3.6.1 Nominal -

$$Q = 7.37 \cos\left[\frac{2.07\ (x - 18)}{36}\right]$$

where x = distance from bottom of fuel column, in.

Q = local linear power, kW/ft

[1]Design limiting fuel pin.

3.6.2 2σ -N/A

3.7 Cladding Temperature Axial Profile[1], °F (°K)

3.7.1 Nominal OD and ID (see Table A-VI)

3.7.2 2σ OD and ID (see Table A-VII)

3.8 Peak-to-Average Power Ratio - EOL

3.8.1 Nominal - 1.204

3.8.2 2σ - N/A

3.9 Uncertainty Factors for Hot Channel Analysis - LHRFDS Ground Rules

3.10 Neutron Flux Axial Profile[1] (E > 0.1 MeV), n/cm^2-s - EOL

$$\Phi = 3.026 \times 10^{15} \text{ Cos} \left[\frac{2.07\ (x - 18)}{36}\right]$$

where x = distance from bottom of fuel column, in.
Φ = local flux

4.0 RADIAL BLANKET ASSEMBLY DATA - CRBR/LHRFDS Ground Rules

5.0 RADIAL BLANKET PIN DATA - CRBR/LHRFDS Ground Rules

6.0 INTERNAL FERTILE ASSEMBLY DATA - N/A

7.0 INTERNAL FERTILE PIN DATA - N/A

8.0 CONTROL ASSEMBLY DATA - N/A

9.0 CONTROL PIN DATA - N/A

10.0 PERFORMANCE CHARACTERISTICS

10.1 Discharge Exposure by Enrichment Zone, MWd/kg

10.1.1 Peak - 100/98

[1]Design limiting fuel pin.

10.1.2 Average - 77/61

10.2 EOL CDF for Design Limiting Fuel Pin - 0.52

10.3 Plenum Pressure History for Design Limiting Fuel Pin - 2σ

$$P = 1.4227T + 177$$

where P = pressure (psia)
T = full power days

10.4 Core Material Volume Fractions

10.4.1 Fuel - 0.3072/0.3445

10.4.2 Sodium - 0.4239

10.4.2.1 Fraction of Na in Interduct Gap - 0.198

10.4.2.2 Fraction of Na in Assembly Interior - 0.802

10.4.2.3 Fraction of Na in Fuel/Clad Bond - 0.000

10.4.3 Steel - 0.2316

10.4.3.1 Fraction of Steel in Duct - 0.404

10.4.3.2 Fraction of Steel in Wire Wrap - 0.117

10.4.3.3 Fraction of Steel in Cladding - 0.480

10.4.4 Control - 0.0374/0

10.5 Breeding Ratio - 1.17

10.6 Breeding Gain, kg/cycle - 134

10.7 Compound System Doubling Time, yrs - 36.5

10.8 Specific Power, MW/kg-fissile - 0.949

10.9 Fuel Cycle Costs, mills/kW-hr - 3.18

10.10 Assembly Exposure, MWd/assembly (Zone 1/Zone 2)

10.10.1 Peak - 2886/2792

10.10.2 Average - 2567/2038

10.11 Sodium Void Worth, Fresh Core/EOEC - +2.76\$/+4.28\$

10.12 Doppler Coefficient - N/A

10.13 LHRFDS Optimization Function - 3.18

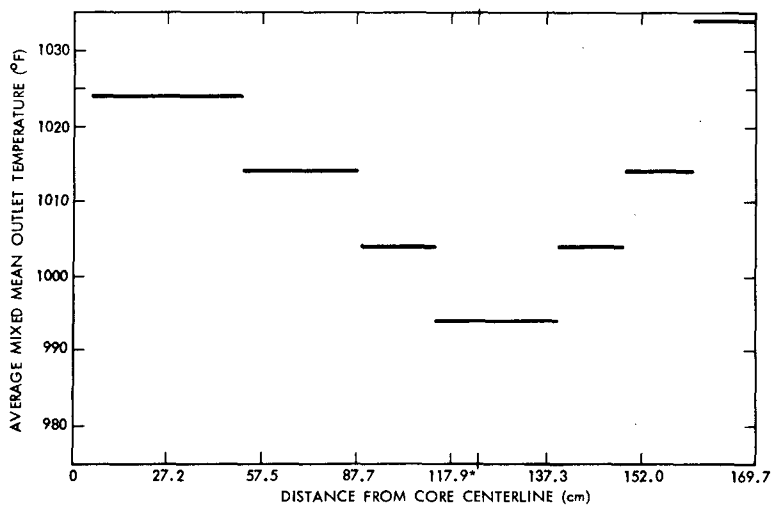
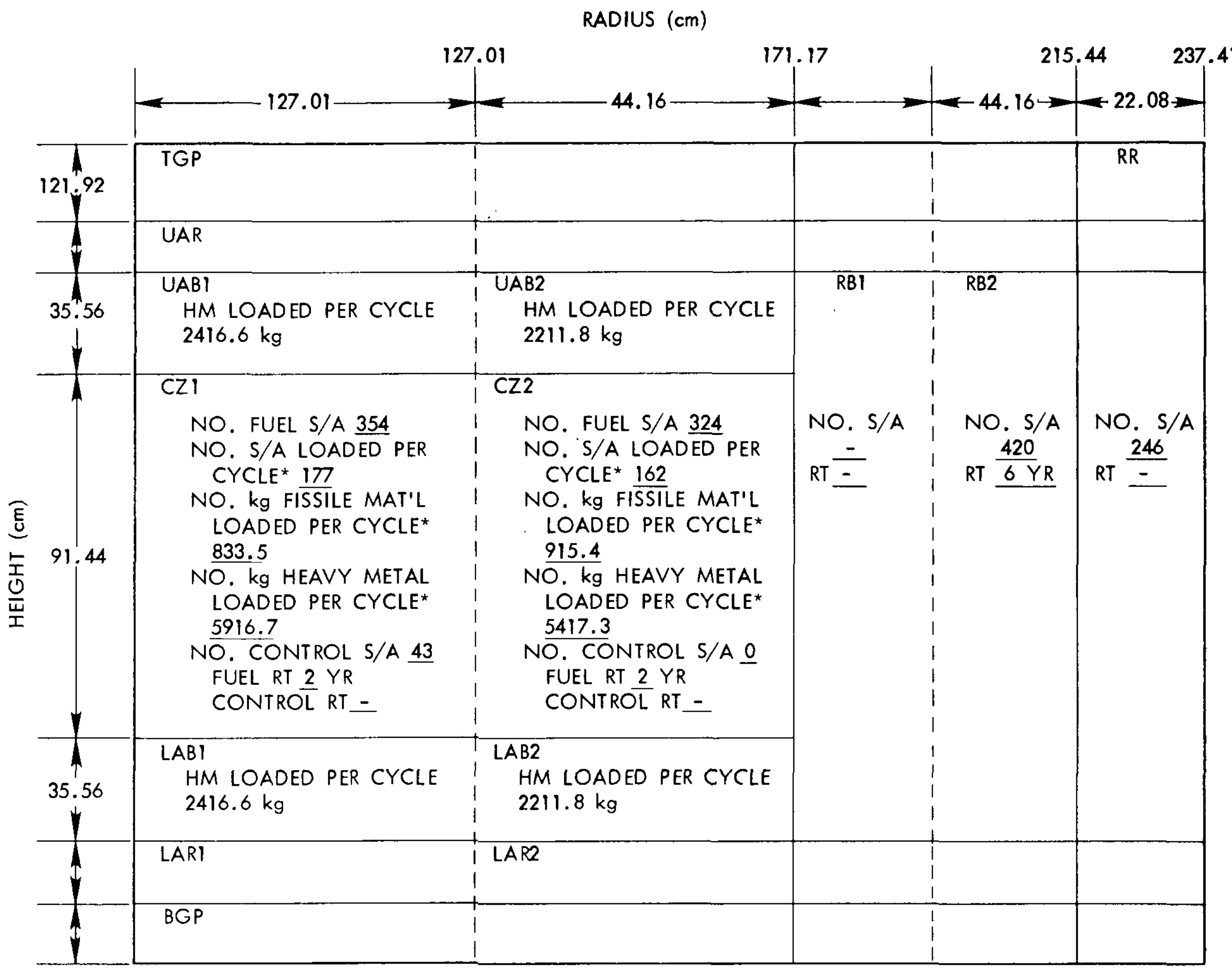


FIGURE A-1. Assembly Coolant Mixed Mean Outlet Temperature (EOL).



TGP = TOP GAS PLENUM
BGP = BOTTOM GAS PLENUM
UAR = UPPER AXIAL REFLECTOR
LAR = LOWER AXIAL REFLECTOR
UAB = UPPER AXIAL BLANKET
LAB = LOWER AXIAL BLANKET
RB1 = ZONE 1 RADIAL BLANKET
RB2 = ZONE 2 RADIAL BLANKET
CZ1 = CORE ZONE 1
CZ2 = CORE ZONE 2

*TO BE SPECIFIED AT START OF EQUILIBRIUM CYCLE.

FIGURE A-2. R-Z Core Diagram for LHRFDS Level I Homogeneous Design.

TABLE A-I

FUEL INVENTORY AT THE BEGINNING OF EQUILIBRIUM CYCLE
(kg)

Isotope	Core Region CZ1	CZ2	UAB1	UAB2	LAB1	LAB2	RB
^{235}U	16.8	15.5	9.1	8.6	9.1	8.6	82.2
^{236}U				Isotope Not Evaluated			
^{238}U	9448	8302	4783	4392	4783	4392	43120
^{239}Pu	1419.8	1537.8	35.7	20.2	35.7	20.2	311.2
^{240}Pu	463.0	494.2	0.86	0.30	0.86	0.30	5.7
^{241}Pu	191.8	217.2	0.024	0.006	0.024	0.006	0.15
^{242}Pu	54.9	58.9	2×10^{-4}	4×10^{-5}	2×10^{-4}	4×10^{-5}	0.002
Fission Products	235.8	168.9	3.8	1.8	3.8	1.8	30.2

TABLE A-II

FUEL INVENTORY AT THE MIDDLE OF EQUILIBRIUM CYCLE
(kg)

Isotope	Core Region CZ1	CZ2	UAB1	UAB2	LAB1	LAB2	RB
^{235}U	14.2	13.9	8.4	8.2	8.4	8.2	80.8
^{236}U				Isotope Not Evaluated			
^{238}U	9242	8182	4744	4371	4744	4371	43040
^{239}Pu	1391.6	1491.2	69.6	39.6	69.6	39.6	386.6
^{240}Pu	489.2	511.0	2.1	0.74	2.1	0.74	8.7
^{241}Pu	168.3	197.3	0.078	0.019	0.078	0.019	0.29
^{242}Pu	57.9	61.1	0.001	2×10^{-4}	0.001	2×10^{-4}	0.004
Fission Products	463.8	334.2	8.3	3.9	8.3	3.9	42.7

TABLE A-III

FUEL INVENTORY AT THE END OF EQUILIBRIUM CYCLE
(kg)

Isotope	Core Region CZ1	CZ2	UAB1	UAB2	LAB1	LAB2	RB
^{235}U	12.0	12.4	7.8	7.8	7.8	7.8	79.2
^{236}U				Isotope Not Evaluated			
^{238}U	9036	8060	4704	4348	4704	4348	42940
^{239}Pu	1363.6	1447.2	101.7	58.7	101.7	58.7	463.4
^{240}Pu	513.6	526.6	4.0	1.5	4.0	1.5	12.4
^{241}Pu	149.7	180.0	0.19	0.05	0.19	0.05	0.48
^{242}Pu	60.1	63.0	0.003	6×10^{-4}	0.003	6×10^{-4}	0.008
Fission Products	687.4	496.8	14.9	6.7	14.9	6.7	57.5

TABLE A-IV

MIDWALL AXIAL TEMPERATURE PROFILE FOR DESIGN LIMITING DUCT
(Nominal, EOL)

Distance Above BOF (ft)	Duct Midwall Temperature (°F)	Duct Midwall Temperature (°K)
0	7.2258E+02	6.5666E+02
2.3077E-01	7.3053E+02	6.6107E+02
4.6154E-01	7.3996E+02	6.6631E+02
6.9231E-01	7.5064E+02	6.7224E+02
9.2308E-01	7.6233E+02	6.7874E+02
1.1538E+00	7.7475E+02	6.8564E+02
1.3845E+00	7.8762E+02	6.9279E+02
1.6154E+00	8.0065E+02	7.0003E+02
1.8462E+00	8.1352E+02	7.0718E+02
2.0769E+00	8.2594E+02	7.1408E+02
2.3077E+00	8.3763E+02	7.2057E+02
2.5385E+00	8.4831E+02	7.2651E+02
2.7692E+00	8.5773E+02	7.3174E+02
3.0000E+00	8.6569E+02	7.3616E+02

TABLE A-V

MIDWALL AXIAL TEMPERATURE PROFILE FOR DESIGN LIMITING DUCT
(2σ, EOL)

Distance Above BOF (ft)	Duct Midwall Temperature (°F)	Duct Midwall Temperature (°K)
0	7.4043E+02	6.6657E+02
2.3077E-01	7.5062E+02	6.7223E+02
4.6154E-01	7.6270E+02	6.7894E+02
6.9231E-01	7.7638E+02	6.8655E+02
9.2308E-01	7.9136E+02	6.9487E+02
1.1538E+00	8.0728E+02	7.0371E+02
1.3845E+00	8.2377E+02	7.1287E+02
1.6154E+00	8.4046E+02	7.2215E+02
1.8462E+00	8.5695E+02	7.3131E+02
2.0769E+00	8.7287E+02	7.4015E+02
2.3077E+00	8.8784E+02	7.4847E+02
2.5385E+00	9.0153E+02	7.5607E+02
2.7692E+00	9.1360E+02	7.6278E+02
3.0000E+00	9.2380E+02	7.6844E+02

TABLE A-VI

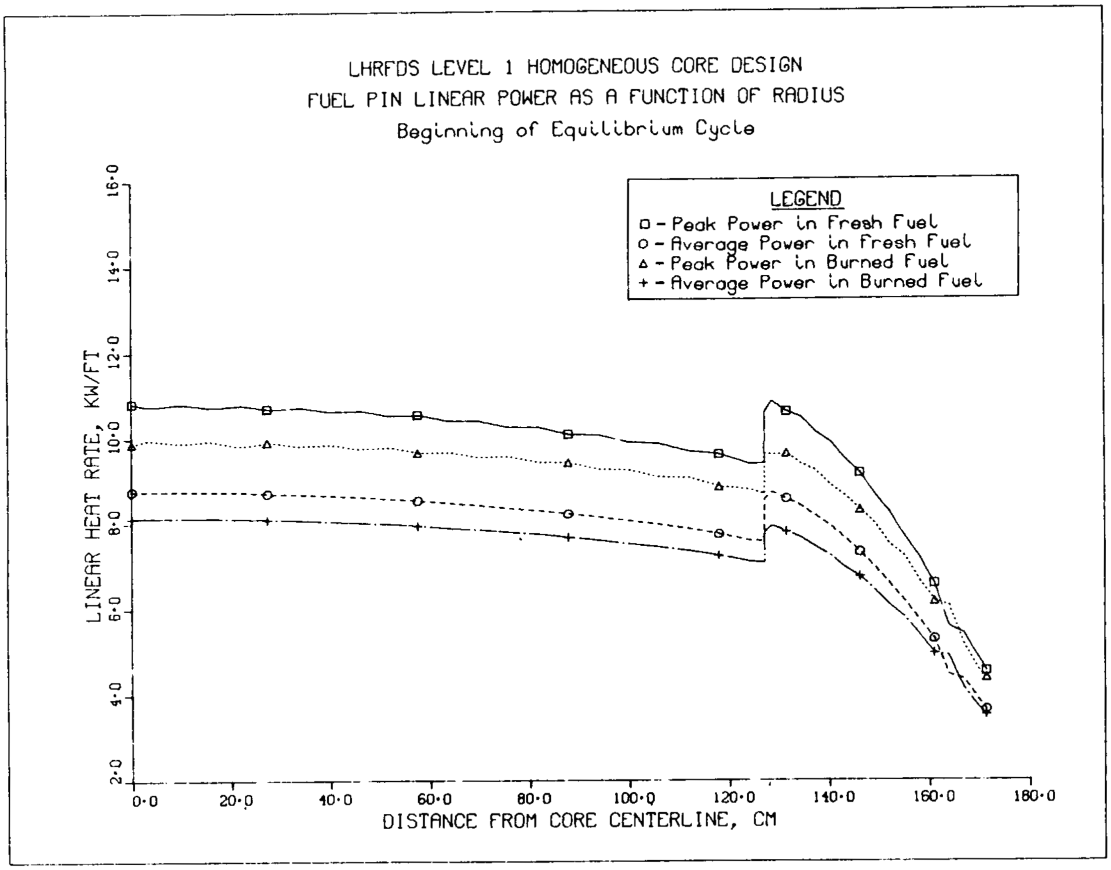
NOMINAL CLADDING TEMPERATURE AXIAL PROFILES FOR DESIGN LIMITING FUEL PIN

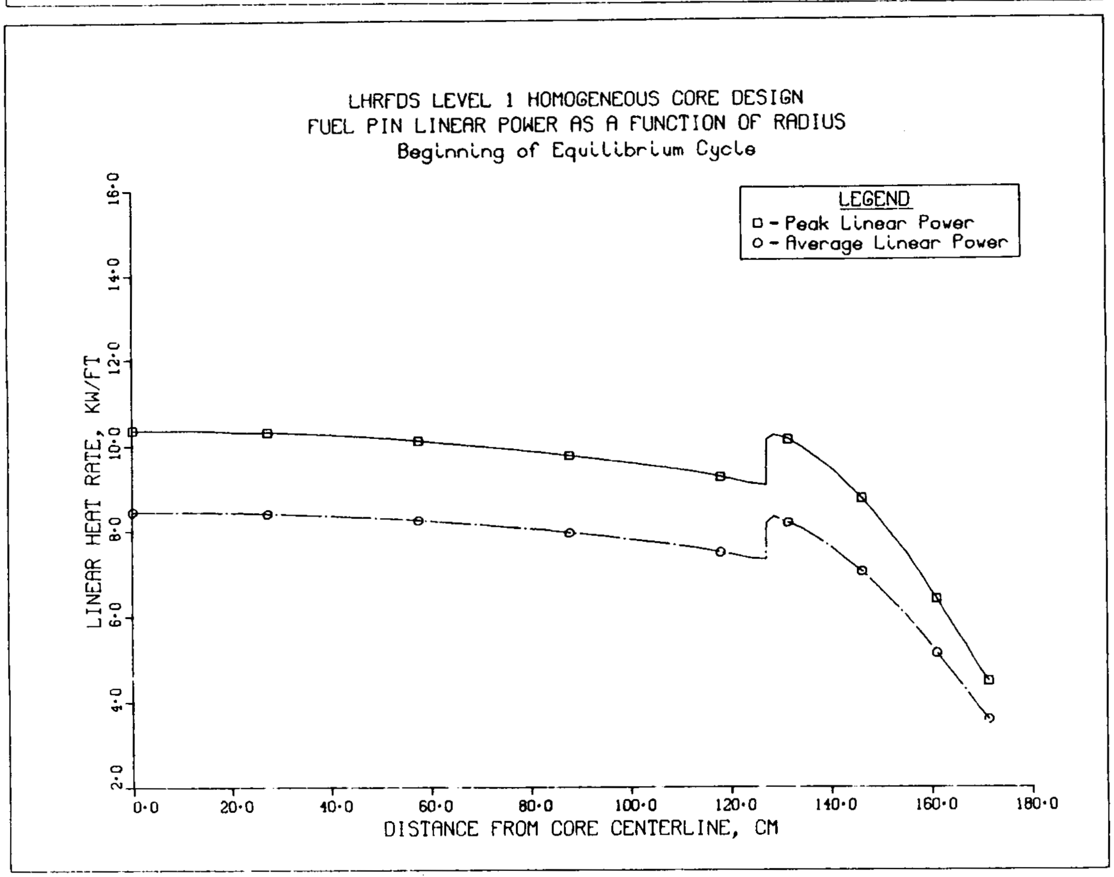
Distance from Bottom of Fuel (in.)	Beginning of Life Clad OD Temperature (°F)	(°K)	Beginning of Life Clad ID Temperature (°F)	(°K)	End of Life Clad OD Temperature (°F)	(°K)	End of Life Clad ID Temperature (°F)	(°K)
1.286	759	677	794	696	754	674	784	691
3.857	788	693	829	716	778	687	815	708
6.429	819	710	866	736	806	703	846	725
9.000	853	729	904	757	835	719	880	744
11.571	889	749	943	779	867	737	913	762
14.143	926	770	981	800	898	754	946	781
16.714	963	790	1018	821	931	772	979	799
19.286	999	810	1053	840	962	790	1009	816
21.857	1034	830	1086	859	992	806	1038	832
24.429	1067	848	1115	875	1020	822	1063	846
27.000	1097	865	1141	889	1045	836	1084	857
29.571	1122	879	1161	900	1067	848	1102	867
32.143	1143	890	1176	909	1085	858	1115	875
34.714	1159	899	1186	914	1099	866	1123	879
36.000	1166	903	1189	916	1105	869	1126	881

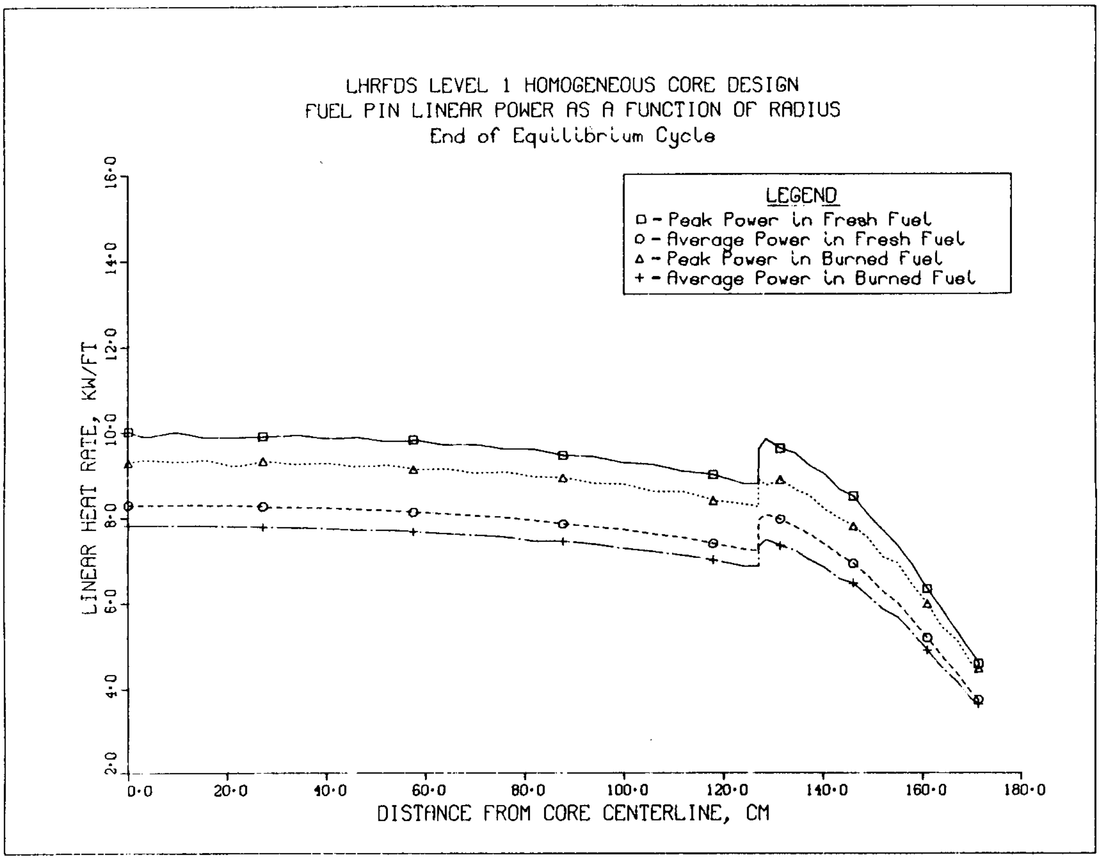
TABLE A-VII

2σ CLADDING TEMPERATURE AXIAL PROFILES FOR DESIGN LIMITING FUEL PIN

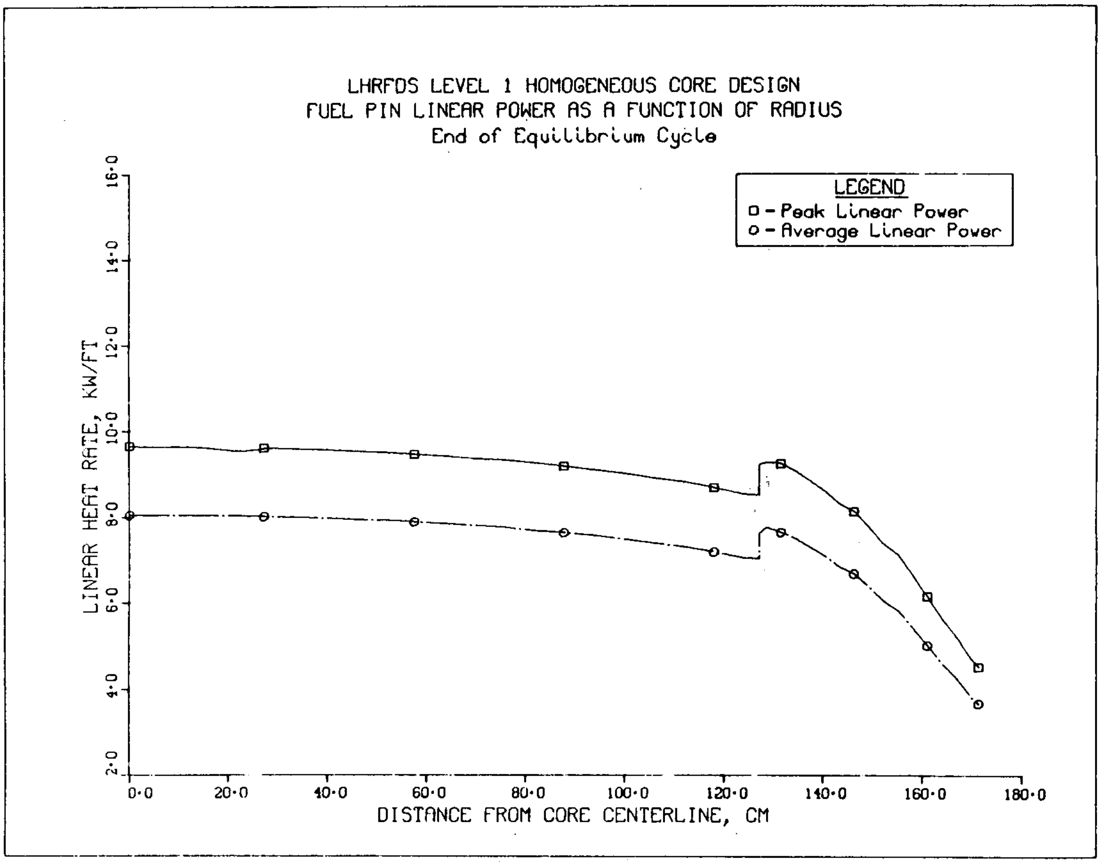
Distance from Bottom of Fuel (in.)	Beginning of Life Clad OD Temperature (°F)	(°K)	Beginning of Life Clad ID Temperature (°F)	(°K)	End of Life Clad OD Temperature (°F)	(°K)	End of Life Clad ID Temperature (°F)	(°K)
1.286	787	692	832	717	781	689	819	710
3.857	824	713	877	742	812	706	859	732
6.429	864	735	924	769	847	726	899	755
9.000	907	759	973	796	885	747	942	779
11.571	954	785	1023	824	925	769	985	802
14.143	1001	811	1071	850	965	791	1027	826
16.714	1049	838	1119	877	1007	815	1069	849
19.286	1095	864	1164	902	1047	837	1108	871
21.857	1140	889	1206	925	1086	859	1144	891
24.429	1182	912	1243	946	1122	879	1176	909
27.000	1220	933	1276	964	1154	896	1204	924
29.571	1252	951	1302	979	1182	912	1226	936
32.143	1279	966	1321	989	1205	925	1243	946
34.714	1300	977	1334	996	1223	935	1254	952
36.000	1308	982	1338	999	1230	939	1257	954


LHRFDS LEVEL 1 HOMOGENEOUS CORE DESIGN
FUEL PIN LINEAR POWER AS A FUNCTION OF RADIUS
Beginning of Equilibrium Cycle
LEGEND
□ - Peak Power in Fresh Fuel
○ - Average Power in Fresh Fuel
△ - Peak Power in Burned Fuel
+ - Average Power in Burned Fuel
LINEAR HEAT RATE, KW/FT
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
0.0
20.0
40.0
60.0
80.0
100.0
120.0
140.0
160.0
180.0
DISTANCE FROM CORE CENTERLINE, CM


LHRFDS LEVEL 1 HOMOGENEOUS CORE DESIGN
FUEL PIN LINEAR POWER AS A FUNCTION OF RADIUS
Beginning of Equilibrium Cycle
LEGEND
□ - Peak Linear Power
○ - Average Linear Power
LINEAR HEAT RATE, KW/FT
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
0.0
20.0
40.0
60.0
80.0
100.0
120.0
140.0
160.0
180.0
DISTANCE FROM CORE CENTERLINE, CM



LHRFDS LEVEL 1 HOMOGENEOUS CORE DESIGN
FUEL PIN LINEAR POWER AS A FUNCTION OF RADIUS
End of Equilibrium Cycle
LEGEND
□ - Peak Power in Fresh Fuel
○ - Average Power in Fresh Fuel
△ - Peak Power in Burned Fuel
+ - Average Power in Burned Fuel
LINEAR HEAT RATE, KW/FT
DISTANCE FROM CORE CENTERLINE, CM
2·0
4·0
6·0
8·0
10·0
12·0
14·0
16·0
0·0
20·0
40·0
60·0
80·0
100·0
120·0
140·0
160·0
180·0



LHRFDS LEVEL 1 HOMOGENEOUS CORE DESIGN
FUEL PIN LINEAR POWER AS A FUNCTION OF RADIUS
End of Equilibrium Cycle
LEGEND
□ - Peak Linear Power
○ - Average Linear Power
LINEAR HEAT RATE, KW/FT
DISTANCE FROM CORE CENTERLINE, CM
2·0
4·0
6·0
8·0
10·0
12·0
14·0
16·0
0·0
20·0
40·0
60·0
80·0
100·0
120·0
140·0
160·0
180·0

1.0 CORE AND REACTOR DATA

1.1 Power Information

1.1.1 Plant Thermal Power, MW_t - 3333

1.1.2 Plant Electric Power, MWe - 1200

1.1.2.1 Net Electric Power, MWe - 1185

1.1.3 Plant Capacity Factor versus Time - 0.70 (constant)

1.1.4 Power Split, Fraction of Total (MOEC)

1.1.4.1 Core Fuel - 0.838

1.1.4.2 Axial Blanket - 0.016

1.1.4.3 Radial Blanket - 0.054

1.1.4.4 Internal Blanket - 0.092

1.1.4.5 Control - 0

1.1.4.6 Radial Shielding - 0

1.1.4.7 Other - 0

1.1.5 Average Linear Power (MOEC)

1.1.5.1 Core Fuel, kW/ft & (W/cm) - 6.243 (204.8)

1.1.5.2 Axial Blanket, kW/ft & (W/cm) - 0.156 (5.1)

1.1.5.3 Radial Blanket, kW/ft & (W/cm) - 1.496 (49.1)

1.1.5.4 Internal Fertile Assembly, kW/ft & (W/cm) - 2.988 (98.0)

1.1.6 Fission Energy and Deposition, MeV/fission - 215, deposited locally

1.2 Temperature Information

1.2.1 Core Inlet Temperature, °F & (°K) - 716 (653)

1.2.2 Core Average Outlet Temperature, °F & (°K) - N/A

1.2.3 Core ΔT, °F & (°K) - N/A

1.2.4 Reactor Inlet Temperature, °F & (°K) - 716 (653)

1.2.5 Reactor Outlet Temperature, °F & (°K) - 965 (791)

1.2.6 Reactor ΔT, °F & (°K) - 249 (138)

1.2.7 Number of Core Orifice Zones - 8 Fuel, 4 Internal Blanket

1.2.8 Radial Profile of Assembly Outlet Temperature - Figure B-1

1.2.9 Core Orificing Criteria - Core orificed to provide 2-year lifetime for all components

1.3 Coolant Information

1.3.1 Peak Power Assembly Pressure Drop, psi & (kPa) - 95 (655)

1.3.2 Reactor Pressure Drop, psi & (kPa) - N/A

1.3.3 Primary System Pressure Drop, psi & (kPa) - 137 (946)

1.3.4 Flow Split, Fraction of Total

1.3.4.1 Core - 68%

1.3.4.2 Internal Fertile Assembly - 12%

1.3.4.3 Radial Blanket

1.3.4.4 Control

1.3.4.5 Radial Shielding

1.3.4.6 Other

20%

1.3.5 Total Coolant Mass Flow Rate, lb_m/hr & (kg/hr) - 1.408×10^8 (6.400×10^7)

1.3.6 Maximum Coolant Velocity, ft/s & (m/s) - 18.73 (5.71)

1.4 Geometric Information (see Figure B-2)

1.4.1 Core Height, in. & (cm) - 36 (91.44)

1.4.2 Axial Blanket Height, in. & (cm) - 14 (35.56)

1.4.3 Radial Blanket Height, in. & (cm) - 64 (162.56)

1.4.4 Axial Shield Height, in. & (cm) - N/A

1.4.5 Number of Core Enrichment Zones - 1

1.4.6 Number of Assemblies, Fuel/IB/RB/Control - 684/313/366/66

1.4.7 Equivalent Diameters[1], in. & (cm) - Figure B-2

1.5 Fuel Management

1.5.1 Refueling Interval, calendar days - 730

[1]Diameter of equivalent volume cylinder.

1.5.2 Fuel Residence Time, full power days (see Figure B-2) - 511

1.5.3 Blanket Residence Time, full power days (see Figure B-2)

1.5.4 Fuel Inventory, kg (see Tables B-I through B-VI)

1.5.5 Fraction of Assemblies Replaced at Each Refueling

1.5.5.1 Fuel Assemblies by Enrichment Zone - 1.0

1.5.5.2 Radial Blanket Assemblies - 0.333

1.5.5.3 Interior Fertile Assemblies - 1.0

2.0 FUEL ASSEMBLY DATA

2.1 Pins per Assembly - 217

2.2 Pin Pitch-to-Diameter Ratio - 1.24

2.3 Spacer Description

2.3.1 Wire Wrap Diameter (or grid spacer thickness & height), in. & (mm) - 0.056 (1.4)

2.3.2 Spacer Pitch, in. & (cm) - 11.9 (30.226)

2.3.3 Edge Ratio - 0.983

2.4 Overall Bundle Length, in. & (cm) - 114 (289.56)

2.5 Lattice Pitch, in. & (cm) - 4.77 (12.12)

2.6 Duct Inside Flat-to-Flat, in. & (cm) - 4.335 (11.011)

2.7 Bundle/Duct Clearance, in. & (mm) - 0.03 (0.765)

2.8 Duct Wall Thickness, in. & (mm) - 0.120 (3.05)

2.9 Interduct Gap, in. & (mm) - 0.195 (4.95)

2.10 Duct Material

2.10.1 Material Type - 316 SS, 20% CW

2.10.2 Swelling Properties - NSMH Rev. 5

2.10.3 Irradiation Creep Properties - NSMH Rev. 3

2.11 Duct Midwall Axial Temperature Profile[1]

2.11.1 Nominal (see Table B-VII)

2.11.2 2σ (see Table B-VIII)

2.12 Duct Wall Pressure Differential Profile[1], psi & (kPa) - EOL

2.12.1 Nominal -

$$P_{psi} = \frac{x}{114} \cdot 32.5 \quad (0 \leq x \leq 114)$$

where x = distance from top of fuel pin bundle (in.)

$$P_{kPa} = \frac{x}{114} \cdot 224 \quad (0 \leq x \leq 114)$$

2.12.2 2σ - N/A

2.13 Neutron Flux Axial Profile[1] (E > 0.1 MeV), n/cm^2-s - EOL

2.13.1 Nominal -

$$\Phi = 2.62 \times 10^{15} \text{ Cos} \left[2.10 \ \frac{(x - 18)}{36}\right] \quad (0 \leq x \leq 36)$$

where x = distance from bottom of fuel column (in.)

2.13.2 2σ - N/A

3.0 FUEL PIN DATA

3.1 Fuel Parameters

3.1.1 Fuel Type (oxide, carbide, nitride) - Oxide

3.1.2 Stoichiometry (O/M, C/M, N/M) - 1.96

3.1.3 Plutonium Content (Pu/Pu + U) - 0.299

3.1.4 Fuel Form (powder or pellet) - Pellet

3.1.4.1 Pellet Diameter, in. & (mm) - 0.1935 (4.9)

3.1.4.2 Pellet Dish and Chamfer Dimensions, in. & (mm) - 0 (0)

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[1]Reported for design limiting duct.

3.1.4.3 Pellet Inside Diameter, in. & (mm) - 0 (0)

3.1.4.4 Pellet Density, g/cm^3 - 10.03

3.1.5 Fuel Smear Density, %TD - 85.5

3.2 Cladding Parameters

3.2.1 Cladding Outside Diameter, in. & (mm) - 0.23 (5.84)

3.2.2 Cladding Wall Thickness, in. & (mm) - 0.015 (0.381)

3.2.3 Diametral Gap, in. & (mm) - 0.0065 (0.1651)

3.2.4 Cladding Material

3.2.4.1 Material Type - 316 SS, 20% CW

3.2.4.2 Swelling Properties - NSMH Rev. 5

3.2.4.3 Irradiation Creep Properties - NSMH Rev. 3

3.2.4.4 Stress-Rupture Properties - LHRFDS Ground Rules

3.3 Stresser Sleeve Parameters - N/A

3.3.1 Sleeve Outside Diameter, in. & (mm)

3.3.2 Sleeve Wall Thickness, in. & (mm)

3.3.3 Fractional Perforation of Sleeve

3.3.4 Sleeve Material

3.4 Equivalent Plenum Volume, in.^{3} & (cc)

3.4.1 Top Plenum - 1.286 (21.082)

3.4.2 Bottom Plenum - N/A

3.5 Bond Type - N/A

3.6 Fuel Pin Linear Power Axial Profile[1], kW/ft - EOL

3.6.1 Nominal -

$$Q = 7.82 \text{ Cos} \left[2.10 \frac{(x - 18)}{36}\right]$$

where x = distance from bottom of fuel column, in.
Q = local linear power, kW/ft

[1]Reported for design limiting fuel pin.

3.6.2 2σ - N/A

3.7 Cladding Temperature Axial Profile, °F (°K)

3.7.1 Nominal OD and ID (see Table B-IX)

3.7.2 2σ OD and ID (see Table B-X)

3.8 Peak-to-Average Pin Linear Power Ratio - EOL

3.8.1 Nominal Axially - 1.21

3.8.2 2σ Axially - N/A

3.9 Uncertainty Factors for Hot Channel Analysis - LHRFDS Ground Rules

3.10 Neutron Flux Axial Profile[1] (E > 0.1 MeV), n/cm^2-s - EOL

$$\Phi = 2.62 \times 10^{15} \cos\left[2.10 \frac{(x - 18)}{36}\right] (0 \le x \le 36)$$

where x = distance from bottom of fuel column, in.

Φ = local flux

4.0 RADIAL BLANKET ASSEMBLY DATA - CRBR/LHRFDS Ground Rules

5.0 RADIAL BLANKET PIN DATA - CRBR/LHRFDS Ground Rules

6.0 INTERNAL FERTILE ASSEMBLY DATA

6.1 Pins per Assembly - 61

6.2 Pin Pitch-to-Diameter Ratio - 1.07

6.3 Spacer Description

6.3.1 Wire Wrap Diameter (or grid spacer thickness and height), in. & (mm) - 0.036 (0.914)

6.3.2 Spacer Pitch, in. & (cm) - N/A

6.3.3 Edge Ratio - N/A

6.4 Overall Assembly Length, in. & (cm) - 114 (289.56)

6.5 Lattice Pitch, in. & (cm) - 4.77 (12.14)

[1] Reported for design limiting fuel pin.

6.6 Duct Inside Flat-to-Flat, in. & (cm) - 4.335 (11.011)

6.7 Bundle/Duct Clearance, in. & (mm) - N/A

6.8 Duct Wall Thickness, in. & (mm) - 0.120 (3.05)

6.9 Interduct Gap, in. & (mm) - 0.195 (0.495)

6.10 Duct Material

6.10.1 Material Type - 316 SS, 20% CW

6.10.2 Swelling Properties - NSMH Rev. 5

6.10.3 Irradiation Creep Properties - NSMH Rev. 3

6.11 Duct Midwall Axial Temperature Profile[1], °F (°K)

6.11.1 Nominal (see Table B-XI)

6.11.2 2σ (see Table B-XII)

6.12 Duct Wall Pressure Differential Profile[1], psi & (kPa) - EOL

6.12.1 Nominal -

$$P_{psi} = \frac{x}{114} \cdot 12.5 \quad (0 \le x \le 114)$$

where x = distance from top of fuel pin bundle (in.)

$$P_{kPa} = \frac{x}{114} \cdot 86 \quad (0 \le x \le 114)$$

6.12.2 2σ - N/A

6.13 Neutron Flux Axial Profile[1] (E > 0.1 MeV), n/cm^2-s - EOL

6.13.1 Nominal -

$$\Phi = 2.08 \times 10^{15} \operatorname{Cos}\left[2.82 \frac{(x - 18)}{36}\right] \quad (0 \le x \le 36)$$

where x = distance from bottom of fuel column (in.)

6.13.2 2σ - N/A

[1]Reported for design limiting duct.

7.0 INTERNAL FERTILE PIN DATA

7.1 Fuel Parameters

7.1.1 Fuel Type (oxide, carbide, nitride) - Oxide

7.1.2 Stoichiometry (O/M, C/M, N/M) - 1.96

7.1.3 Plutonium Content (Pu/Pu + U) - 0

7.1.4 Fuel Form (powder or pellet) - Pellet

7.1.4.1 Pellet Diameter, in. & (mm) - 0.472 (12.0)

7.1.4.2 Pellet Dish and Chamfer Dimensions, in. & (mm) - 0 (0)

7.1.4.3 Pellet Inside Diameter, in. & (mm) - 0 (0)

7.1.4.4 Pellet Density, g/cm^3 - 10.3

7.1.5 Fuel Smear Density, %TD - 93.7

7.2 Cladding Parameters

7.2.1 Cladding Outside Diameter, in. & (mm) - 0.506 (12.9)

7.2.2 Cladding Wall Thickness, in. & (mm) - 0.015 (0.381)

7.2.3 Diametral Gap, in. & (mm) - 0.004 (0.101)

7.2.4 Cladding Material

7.2.4.1 Material Type - 316 SS, 20% CW

7.2.4.2 Swelling Properties - NSMH Rev. 5

7.2.4.3 Irradiation Creep Properties - NSMH Rev. 3

7.2.4.4 Stress-Rupture Properties - LHRFDS Ground Rules

7.3 Stresser Sleeve Parameters - N/A

7.3.1 Sleeve Outside Diameter, in. & (mm)

7.3.2 Sleeve Wall Thickness, in. & (mm)

7.3.3 Fractional Perforation of Sleeve

7.3.4 Sleeve Material

7.4 Equivalent Plenum Volume, in.^{3} & (cc)

7.4.1 Top Plenum - 7.29 (119.4)

7.4.2 Bottom Plenum - N/A

7.5 Bond Type (sodium or helium) - N/A

7.6 Fuel Pin Linear Power Axial Profile[1], kW/ft - EOL

7.6.1 Nominal -

$$Q = 13.96 \text{ Cos} \left[2.82 \frac{(x - 18)}{36}\right]$$

where x = distance from bottom of fuel column, in.
Q = local linear power, kW/ft

7.6.2 2σ - N/A

7.7 Cladding Temperature Axial Profile, °F & (°K)

7.7.1 Nominal OD and ID (see Table B-XIII)

7.7.2 2σ OD and ID (see Table B-XIV)

7.8 Peak-to-Average Pin Linear Power Ratio - EOL

7.8.1 Nominal Axially - 1.39

7.8.2 2σ Axially - N/A

7.9 Uncertainty Factors for Hot Channel Analysis - LHRFDS Ground Rules

7.10 Neutron Flux Axial Profile[1] (E > 0.1 MeV), n/cm^2-s - EOL

$$\Phi = 2.08 \times 10^{15} \text{ Cos} \left[2.82 \frac{(x - 18)}{36}\right] (0 \leq x \leq 36)$$

where x = distance from bottom of fuel column, in.
Φ = local flux

8.0 CONTROL ASSEMBLY DATA - N/A

9.0 CONTROL PIN DATA - N/A

10.0 PERFORMANCE CHARACTERISTICS

10.1 Discharge Exposure, MWd/kg

10.1.1 Peak - 88

10.1.2 Average - 62

[1]Reported for design limiting pin.

10.2 EOL CDF for Design Limiting Fuel Pin - 0.65

10.3 Plenum Pressure History for Design Limiting Fuel Pin - 2σ

$$P = 1.5235T + 177$$

where P = pressure (psia)
T = full power days

10.4 Core Material Volume Fractions

10.4.1 Fuel - 0.5509/0.3075/0.5509/0.2940/0.5509/0.3154/0.5590/0.3236

10.4.2 Sodium (Blankets/Fuel) - 0.2835/0.4215

10.4.2.1 Fraction of Na in Interduct Gap - 0.2825/0.190

10.4.2.2 Fraction of Na in Assembly Interior - 0.7175/0.8100

10.4.2.3 Fraction of Na in Fuel/Clad Bond - 0.000

10.4.3 Steel - 0.1656/0.2326

10.4.3.1 Fraction of Steel in Duct - 0.5569/0.404

10.4.3.2 Fraction of Steel in Wire Wrap - 0.0187/0.117

10.4.3.3 Fraction of Steel in Cladding - 0.4244/0.480

10.5 Breeding Ratio - 1.29

10.6 Breeding Gain, kg/cycle - 452

10.7 Compound System Doubling Time, yrs - 31.8

10.8 Specific Power, MW/kg-fissile - 0.63

10.9 Fuel Cycle Costs, mills/kW-hr - 5.53

10.10 Assembly Exposure, MWd/assembly

10.10.1 Peak - 2418

10.10.2 Average - 2112

10.11 Sodium Void Worth* (BOEC/MOEC/Estimated EOEC), $ - -0.11/+0.75/+1.6

10.12 Doppler Coefficient - N/A

10.13 LHRFDS Optimization Function - 3.87

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*Worth of voiding flowing sodium, active core length only. No voiding of control, axial or radial blankets, nor interduct gaps.

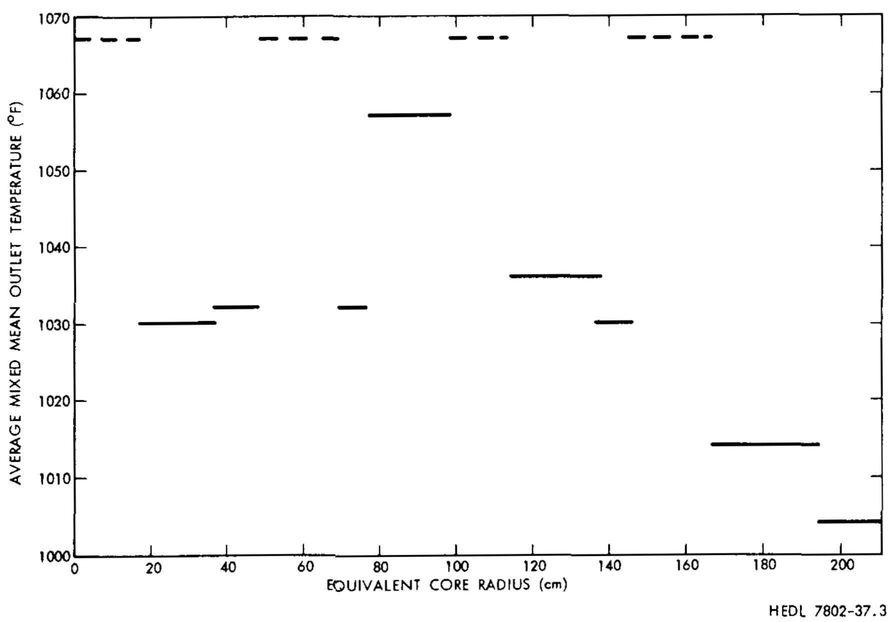
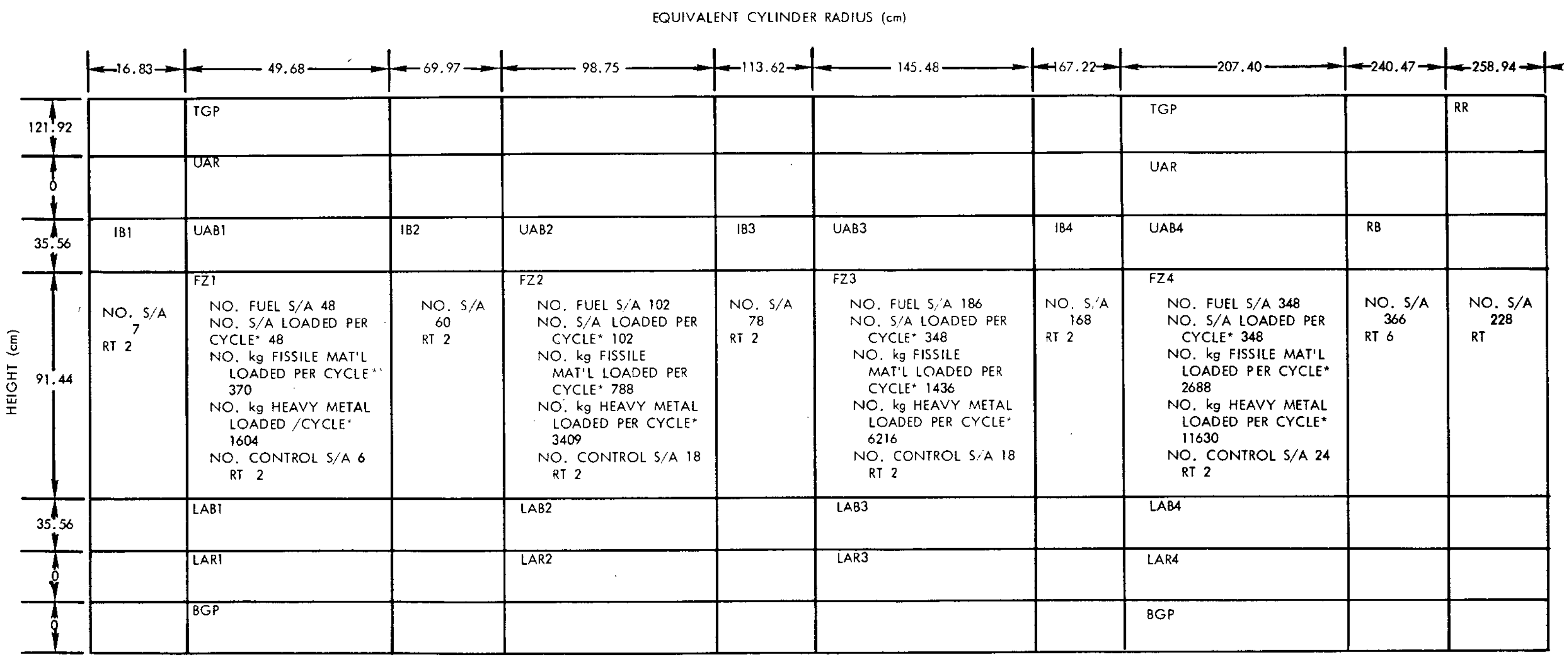


FIGURE B-1. Assembly Coolant Mixed Mean Outlet Temperature (EOL).



TGP = TOP GAS PLENUM
BGP = BOTTOM GAS PLENUM
UAR = UPPER AXIAL REFLECTOR
LAR = LOWER AXIAL REFLECTOR

UAB = UPPER AXIAL BLANKET
LAB = LOWER AXIAL BLANKET
RB = RADIAL BLANKET

FZ1 = FUEL ZONE 1
FZ2 = FUEL ZONE 2
FZ3 = FUEL ZONE 3
FZ4 = FUEL ZONE 4

IB1 = INTERIOR BLANKET 1
IB2 = INTERIOR BLANKET 2
IB3 = INTERIOR BLANKET 3
IB4 = INTERIOR BLANKET 4

*SPECIFIED FOR START OF EQUILIBRIUM CYCLE

FIGURE B-2. R-Z Core Diagram for LHRFDS Level I Heterogeneous Design.

TABLE B-I

FUEL INVENTORY AT THE BEGINNING OF EQUILIBRIUM CYCLE
(kg)

Isotope	Core Region							
	FZ1	FZ2	FZ3	FZ4	AB1*	AB2	AB3	AB4
^{235}U	2.28	4.86	8.86	16.6	2.66	5.65	10.3	19.3
^{236}U				Isotope Not Evaluated				
^{238}U	1123	2388	4354	8146	1307	2778	5068	9480
^{239}Pu	322	685	1248	2336	0	0	0	0
^{240}Pu	96.7	206	375	702	0	0	0	0
^{241}Pu	48.5	103	188	352	0	0	0	0
^{242}Pu	11.5	24.4	44.5	83.2	0	0	0	0
Fission Products	0	0	0	0	0	0	0	0

*Note - Number shown is for combined upper and lower axial blankets (model had midplane symmetry).

TABLE B-II

FUEL INVENTORY AT THE BEGINNING OF EQUILIBRIUM CYCLE
(kg)

Isotope	Core Region				
	IB1	IB2	IB3	IB4	RB
^{235}U	1.47	12.6	16.4	35.4	69.8
^{236}U			Isotope Not Evaluated		
^{238}U	724	6208	8076	17390	37480.0
^{239}Pu	0	0	0	0	362.6
^{240}Pu	0	0	0	0	7.25
^{241}Pu	0	0	0	0	0.192
^{242}Pu	0	0	0	0	2×10^{-3}
Fission Products	0	0	0	0	40.4

TABLE B-III

FUEL INVENTORY AT THE MIDDLE OF EQUILIBRIUM CYCLE
(kg)

Isotope	Core Region FZ1	FZ2	FZ3	FZ4	AB1*	AB2	AB3	AB4
^{235}U	2.05	4.31	7.57	13.7	2.54	5.37	9.67	17.8
^{236}U	Isotope Not Evaluated							
^{238}U	1107	2350	4264	7936	1301	2764	5034	9404
^{239}Pu	305	645	1153	2112	5.74	14.0	31.9	70.1
^{240}Pu	100	214	394	739	3.88×10^{-2}	0.109	0.311	0.795
^{241}Pu	43.1	90.8	161	292	3.11×10^{-4}	1.01×10^{-3}	3.60×10^{-3}	1.06×10^{-2}
^{242}Pu	12.0	25.6	47.2	89.0	8.14×10^{-7}	3.06×10^{-6}	1.38×10^{-5}	4.82×10^{-5}
Fission Products	33.0	77.5	187	441	0.411	0.990	2.51	6.27

*Note - Number shown is for combined upper and lower axial blankets (model had midplane symmetry).

TABLE B-IV

FUEL INVENTORY AT THE MIDDLE OF EQUILIBRIUM CYCLE
(kg)

Isotope	Core Region IB1	IB2	IB3	IB4	RB
^{235}U	1.34	11.6	14.6	30.8	65.9
^{236}U	Isotope Not Evaluated				
^{238}U	717	6152	7972	17122	37220
^{239}Pu	6.55	52.1	92.3	232	562
^{240}Pu	0.104	0.778	1.87	5.73	17.7
^{241}Pu	2.00×10^{-3}	1.42×10^{-2}	4.53×10^{-2}	0.172	0.716
^{242}Pu	1.40×10^{-5}	9.21×10^{-5}	4.06×10^{-4}	1.87×10^{-3}	1.20×10^{-3}
Fission Products	0.773	5.50	12.4	30.5	82.9

TABLE B-V

FUEL INVENTORY AT THE END OF EQUILIBRIUM CYCLE
(kg)

Isotope	Core Region: FZ1	FZ2	FZ3	FZ4	AB1*	AB2	AB3	AB4
^{235}U	1.81	3.73	6.36	11.4	2.41	5.05	9.00	16.6
^{236}U	Isotope Not Evaluated							
^{238}U	1089	2304	4164	7746	1294	2746	4994	9332
^{239}Pu	287	602	1060	1937	12.4	29.9	65.5	132
^{240}Pu	104	223	411	767	0.183	0.509	1.35	2.89
^{241}Pu	38.1	79.5	138	250	3.16×10^{-3}	1.01×10^{-2}	3.21×10^{-2}	7.30×10^{-2}
^{242}Pu	12.5	26.8	49.5	92.9	1.82×10^{-5}	6.66×10^{-4}	2.60×10^{-4}	6.40×10^{-4}
Fission Products	69.8	164	379	807	1.13	2.81	6.99	15.5

*Note - Number shown is for combined upper and lower axial blankets (model had midplane symmetry).

TABLE B-VI

FUEL INVENTORY AT THE END OF EQUILIBRIUM CYCLE
(kg)

Isotope	Core Region: IB1	IB2	IB3	IB4	RB
^{235}U	1.21	10.5	12.8	26.9	62.4
^{236}U	Isotope Not Evaluated				
^{238}U	709	6084	7854	16856	37000
^{239}Pu	13.3	107.0	182.0	427.0	734.0
^{240}Pu	0.449	3.43	7.64	20.5	30.5
^{241}Pu	1.76×10^{-2}	0.129	0.368	1.13	1.58
^{242}Pu	2.66×10^{-4}	1.83×10^{-3}	7.05×10^{-3}	2.48×10^{-2}	3.56×10^{-2}
Fission Products	2.322	16.9	37.4	89.2	133.0

TABLE B-VII

MIDWALL AXIAL TEMPERATURE PROFILE FOR DESIGN LIMITING FUEL DUCT
(Nominal, EOL)

Distance Above BOF (ft)	Duct Midwall Temperature (°F)	Duct Midwall Temperature (°K)
0	7.1856E+02	654
2.3077E-01	7.2696E+02	659
4.6154E-01	7.3722E+02	665
6.9231E-01	7.4907E+02	671
9.2308E-01	7.6226E+02	679
1.1538E+00	7.7628E+02	686
1.3845E+00	7.9093E+02	695
1.6154E+00	8.0578E+02	703
1.8462E+00	8.2043E+02	711
2.0769E+00	8.3450E+02	719
2.3077E+00	8.4764E+02	726
2.5385E+00	8.5949E+02	733
2.7692E+00	8.6975E+02	738
3.0000E+00	8.7815E+02	743

TABLE B-VIII

MIDWALL AXIAL TEMPERATURE PROFILE FOR DESIGN LIMITING FUEL DUCT
(2σ, EOL)

Distance Above BOF (ft)	Duct Midwall Temperature (°F)	Duct Midwall Temperature (°K)
0	7.3528E+02	664
2.3077E-01	7.4604E+02	670
4.6154E-01	7.5919E+02	677
6.9231E-01	7.7437E+02	685
9.2308E-01	7.9127E+02	695
1.1538E+00	8.0924E+02	705
1.3845E+00	8.2801E+02	715
1.6154E+00	8.4704E+02	726
1.8462E+00	8.6581E+02	736
2.0769E+00	8.8383E+02	746
2.3077E+00	9.0067E+02	756
2.5385E+00	9.1585E+02	764
2.7692E+00	9.2900E+02	771
3.0000E+00	9.3976E+02	777

TABLE B-IX

NOMINAL CLADDING TEMPERATURE AXIAL PROFILES FOR DESIGN LIMITING FUEL PIN

Distance from Bottom of Fuel (in.)	Beginning of Life: Clad OD Temperature (°F)	Beginning of Life: Clad OD Temperature (°K)	Beginning of Life: Clad ID Temperature (°F)	Beginning of Life: Clad ID Temperature (°K)	End of Life: Clad OD Temperature (°F)	End of Life: Clad OD Temperature (°K)	End of Life: Clad ID Temperature (°F)	End of Life: Clad ID Temperature (°K)
1.286	742	667	779	688	739	666	770	683
3.857	767	681	812	706	760	677	797	698
6.429	796	697	846	725	784	691	826	714
9.000	828	715	883	746	810	705	857	731
11.571	863	735	922	767	839	721	889	749
14.143	899	755	959	788	869	783	921	767
16.714	936	775	997	809	900	755	952	784
19.286	973	796	1033	829	931	772	983	801
21.857	1009	816	1067	848	962	790	1011	817
24.429	1044	835	1099	866	991	806	1038	832
27.000	1076	853	1127	881	1018	821	1061	845
29.571	1105	869	1150	894	1042	834	1080	855
32.143	1130	883	1169	905	1063	846	1096	864
34.714	1151	895	1183	912	1080	855	1108	871
36.000	1160	900	1188	915	1088	860	1112	873

TABLE B-X

2σ CLADDING TEMPERATURE AXIAL PROFILES FOR DESIGN LIMITING FUEL PIN

Distance from Bottom of Fuel (in.)	Beginning of Life: Clad OD Temperature (°F)	Beginning of Life: Clad OD Temperature (°K)	Beginning of Life: Clad ID Temperature (°F)	Beginning of Life: Clad ID Temperature (°K)	End of Life: Clad OD Temperature (°F)	End of Life: Clad OD Temperature (°K)	End of Life: Clad ID Temperature (°F)	End of Life: Clad ID Temperature (°K)
1.286	789	694	839	721	781	689	822	712
3.857	827	715	886	747	812	706	862	734
6.429	869	738	935	775	847	726	903	757
9.000	914	763	985	802	885	747	945	780
11.571	961	789	1036	831	924	769	989	805
14.143	1009	816	1086	859	965	791	1030	827
16.714	1058	843	1134	885	1005	814	1071	850
19.286	1105	869	1180	911	1045	836	1109	871
21.857	1151	895	1222	934	1083	857	1145	891
24.429	1193	918	1260	955	1118	876	1176	909
27.000	1232	940	1293	974	1150	894	1204	924
29.571	1265	958	1319	988	1178	910	1225	936
32.143	1292	973	1339	999	1201	922	1242	945
34.714	1314	985	1352	1006	1219	932	1252	951
36.000	1322	990	1356	1009	1226	936	1256	953

TABLE B-XI

MIDWALL AXIAL TEMPERATURE PROFILE FOR DESIGN LIMITING INTERNAL FERTILE DUCT
(Nominal, EOL)

Distance Above BOF (ft)	Duct Midwall Temperature (°F)	Duct Midwall Temperature (°K)
0	7.2547E+02	6.5826E+02
2.3077E-01	7.3060E+02	6.6111E+02
4.6154E-01	7.3976E+02	6.6620E+02
6.9231E-01	7.5250E+02	6.7328E+02
9.2308E-01	7.6823E+02	6.8202E+02
1.1538E+00	7.8620E+02	6.9200E+02
1.3845E+00	8.0555E+02	7.0275E+02
1.6154E+00	8.2538E+02	7.1377E+02
1.8462E+00	8.4473E+02	7.2452E+02
2.0769E+00	8.6270E+02	7.3450E+02
2.3077E+00	8.7842E+02	7.4323E+02
2.5385E+00	8.9117E+02	7.5032E+02
2.7692E+00	9.0032E+02	7.5540E+02
3.0000E+00	9.0546E+02	7.5826E+02

TABLE B-XII

MIDWALL AXIAL TEMPERATURE PROFILE FOR DESIGN LIMITING INTERNAL FERTILE DUCT
(2σ, EOL)

Distance Above BOF (ft)	Duct Midwall Temperature (°F)	Duct Midwall Temperature (°K)
0	7.4413E+02	6.6863E+02
2.3077E-01	7.5071E+02	6.7228E+02
4.6154E-01	7.6244E+02	6.7880E+02
6.9231E-01	7.7877E+02	6.8787E+02
9.2308E-01	7.9892E+02	6.9907E+02
1.1538E+00	8.2195E+02	7.1186E+02
1.3845E+00	8.4674E+02	7.2563E+02
1.6154E+00	8.7215E+02	7.3975E+02
1.8462E+00	8.9694E+02	7.5352E+02
2.0769E+00	9.1997E+02	7.6632E+02
2.3077E+00	9.4011E+02	7.7751E+02
2.5385E+00	9.5645E+02	7.8658E+02
2.7692E+00	9.6817E+02	7.9309E+02
3.0000E+00	9.7476E+02	7.9676E+02

TABLE B-XIII

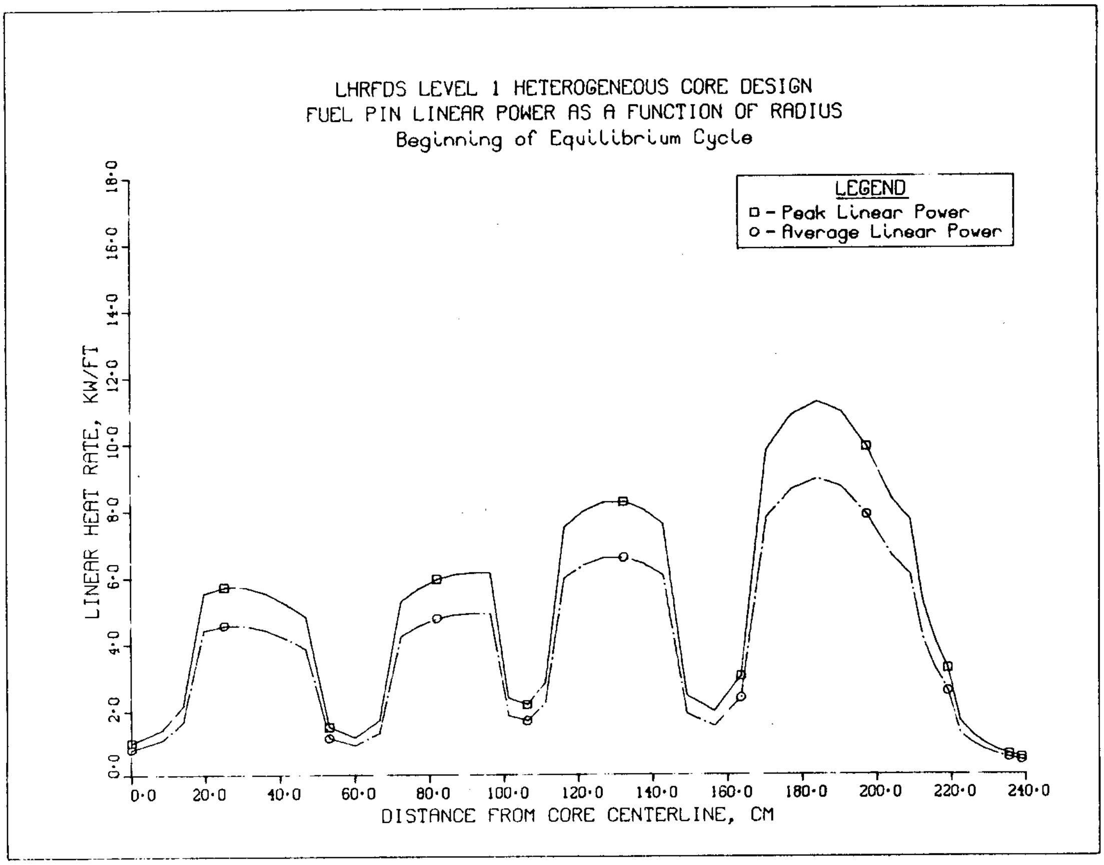
NOMINAL CLADDING TEMPERATURE AXIAL PROFILES FOR DESIGN LIMITING INTERNAL FERTILE PIN

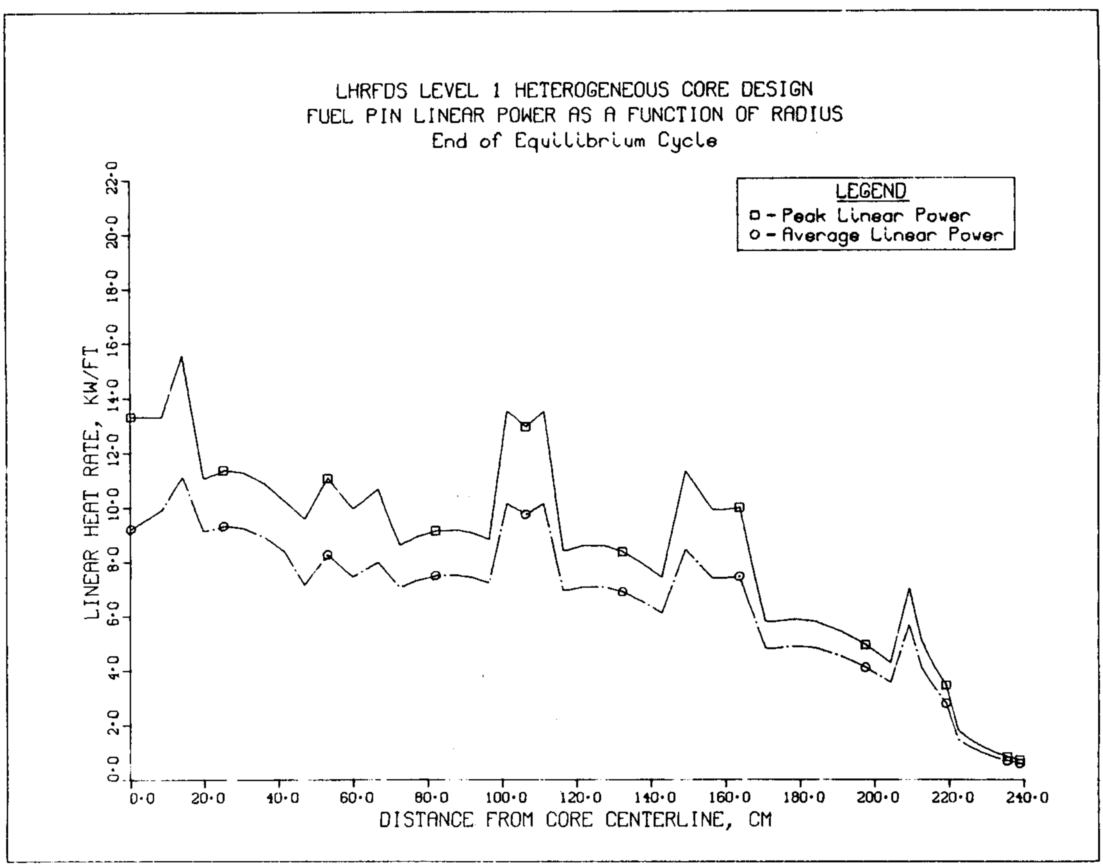
Distance from Bottom of Fuel (in.)	Beginning of Life Clad OD Temperature (°F)	(°K)	Beginning of Life Clad ID Temperature (°F)	(°K)	End of Life Clad OD Temperature (°F)	(°K)	End of Life Clad ID Temperature (°F)	(°K)
1.286	741	667	744	669	746	670	756	675
3.857	746	670	751	672	762	679	780	689
6.429	753	674	760	677	785	691	809	705
9.000	761	678	770	683	815	708	843	724
11.571	771	684	781	689	849	727	882	745
14.143	782	690	793	696	886	747	922	767
16.714	794	696	805	702	926	770	962	790
19.286	806	703	817	709	966	792	1002	812
21.857	817	709	828	715	1005	814	1039	832
24.429	827	715	837	720	1041	834	1071	850
27.000	837	720	845	725	1072	851	1099	866
29.571	844	724	851	728	1098	865	1119	877
32.143	850	727	855	730	1118	876	1133	885
34.714	854	730	856	731	1130	883	1138	887
36.000	854	730	856	731	1133	885	1138	887

TABLE B-XIV

2σ CLADDING TEMPERATURE AXIAL PROFILES FOR DESIGN LIMITING INTERNAL FERTILE PIN

Distance from Bottom of Fuel (in.)	Beginning of Life Clad OD Temperature (°F)	(°K)	Beginning of Life Clad ID Temperature (°F)	(°K)	End of Life Clad OD Temperature (°F)	(°K)	End of Life Clad ID Temperature (°F)	(°K)
1.286	760	677	764	680	771	684	785	691
3.857	768	682	775	686	798	699	821	711
6.429	778	687	787	692	832	717	863	735
9.000	790	694	801	700	873	740	911	761
11.571	804	702	817	709	921	767	963	790
14.143	818	710	833	718	971	795	1015	819
16.714	834	719	848	726	1023	824	1068	849
19.286	849	727	863	735	1074	852	1118	876
21.857	863	735	877	742	1122	879	1164	902
24.429	876	742	889	749	1166	903	1204	924
27.000	887	748	898	754	1204	924	1236	942
29.571	895	752	904	757	1233	940	1258	954
32.143	901	756	908	760	1253	951	1271	961
34.714	904	757	908	760	1263	957	1274	963
36.000	905	758	907	759	1265	958	1271	961


LHRFDS LEVEL 1 HETEROGENEOUS CORE DESIGN
FUEL PIN LINEAR POWER AS A FUNCTION OF RADIUS
Beginning of Equilibrium Cycle
LEGEND
□ - Peak Linear Power
○ - Average Linear Power
LINEAR HEAT RATE, KW/FT
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
0.0
20.0
40.0
60.0
80.0
100.0
120.0
140.0
160.0
180.0
200.0
220.0
240.0
DISTANCE FROM CORE CENTERLINE, CM


LHRFDS LEVEL 1 HETEROGENEOUS CORE DESIGN
FUEL PIN LINEAR POWER AS A FUNCTION OF RADIUS
End of Equilibrium Cycle
LEGEND
□ - Peak Linear Power
○ - Average Linear Power
LINEAR HEAT RATE, KW/FT
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
22.0
0.0
20.0
40.0
60.0
80.0
100.0
120.0
140.0
160.0
180.0
200.0
220.0
240.0
DISTANCE FROM CORE CENTERLINE, CM

APPENDIX B

SPECIFICATION OF LHRFDS LEVEL I HETEROGENEOUS CORE DESIGN IN THE ALSDAWG FORMAT

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March 1977
(Revised May 1977)

APPENDIX C

SPECIFICATION OF LHRFDS LEVEL II HOMOGENEOUS CORE DESIGN IN THE ALSDAWG FORMAT

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1.0 CORE AND REACTOR DATA

1.1 Power Information

1.1.1 Plant Thermal Power, MW_t - 3734

1.1.2 Plant Electric Power, MWe - 1200

1.1.2.1 Net Electric Power, MWe 1185

1.1.3 Plant Capacity Factor versus Time - 0.70 (constant)

1.1.4 Power Split, Fraction of Total (BOEC)

1.1.4.1 Core Fuel - 0.9329

1.1.4.2 Axial Blanket - 0.0428

1.1.4.3 Radial Blanket - 0.0243

1.1.4.4 Internal Blanket - 0

1.1.4.5 Control - 0

1.1.4.6 Radial Shielding - 0

1.1.4.7 Other - 0

1.1.5 Average Linear Power (BOEC)

1.1.5.1 Core Fuel, kW/ft & (W/cm) - 8.75 (287.1)

1.1.5.2 Axial Blanket, kW/ft & (W/cm) - 0.216 (7.098)

1.1.5.3 Radial Blanket, kW/ft & (W/cm) - 0.0007 (0.025)

1.1.5.4 Internal Fertile Assembly, kW/ft & (W/cm) - N/A

1.1.6 Fission Energy and Deposition, MeV/fission - 215, deposited locally

1.2 Temperature Information

1.2.1 Core Inlet Temperature, °F & (°K) - 595 (589)

1.2.2 Core Average Outlet Temperature, °F & (°K) - N/A

1.2.3 Core ΔT, °F & (°K) - N/A

1.2.4 Reactor Inlet Temperature, °F & (°K) - 595 (589)

1.2.5 Reactor Outlet Temperature, °F & (°K) - 895 (752)

1.2.6 Reactor ΔT, °F & (°K) - 300 (167)

1.2.7 Number of Core Orifice Zones - 5

1.2.8 Radial Profile of Assembly Outlet Temperature - Figure C-1

1.2.9 Core Orificing Criteria - Core orificed to provide acceptable fuel pin and duct lifetimes.

1.3 Coolant Information

1.3.1 Peak Power Assembly Pressure Drop, psi & (kPa) - 43 (296)

1.3.2 Reactor Pressure Drop, psi & (kPa) - N/A

1.3.3 Primary System Pressure Drop, psi & (kPa) - ∿ 100 (690)

1.3.4 Flow Split, Fraction of Total

1.3.4.1 Core - 85%

1.3.4.2 Radial Blanket

1.3.4.3 Internal Fertile Assembly

1.3.4.4 Control

1.3.4.5 Radial Shielding

1.3.4.6 Other

(1.3.4.2 – 1.3.4.6) 15%

1.3.5 Total Coolant Mass Flow Rate, lb_m/hr & (kg/hr) - 1.427×10^8 (6.486×10^7)

1.3.6 Maximum Coolant Velocity, ft/s & (m/s) - 22.30 (6.80)

1.4 Geometric Information (see Figure C-2)

1.4.1 Core Height, in. & (cm) - 24 (60.96)

1.4.2 Axial Blanket Height, in. & (cm) - 14 (35.56)

1.4.3 Radial Blanket Height, in. & (cm) - 52 (132.08)

1.4.4 Axial Shield Height, in. & (cm) - N/A

1.4.5 Number of Core Enrichment Zones - 2

1.4.6 Number of Assemblies, CZ1, CZ2, RB - 547/270/444

1.4.7 Equivalent Diameters[1], in. & (cm) - 119.7 (304.1)/146.3 (371.6)/181.8 (461.7)

1.5 Fuel Management

1.5.1 Refueling Interval, calendar days - 365

[1]Diameter of equivalent volume cylinder.

1.5.2 Fuel Residence Time, full power days (see Figure C-2) - 511

1.5.3 Blanket Residence Time, full power days (see Figure C-2) - 1278

1.5.4 Fuel Inventory, kg (see Tables C-I through C-III)

1.5.5 Fraction of Assemblies Replaced at Each Refueling

1.5.5.1 Fuel Assemblies by Enrichment Zone - 0.5/0.5

1.5.5.2 Radial Blanket Assemblies - 0.2

1.5.5.3 Interior Fertile Assemblies - N/A

2.0 FUEL ASSEMBLY DATA

2.1 Pins per Assembly - 271

2.2 Pin Pitch-to-Diameter Ratio - 1.17

2.3 Spacer Description

2.3.1 Wire Wrap Diameter (or grid spacer thickness & height), in. & (mm) - 0.039 (0.99)

2.3.2 Spacer Pitch, in. & (cm) - 11.9 (30.226)

2.3.3 Edge Ratio - 0.98

2.4 Overall Bundle Length, in. & (cm) - 84 (213.36)

2.5 Lattice Pitch, in. & (cm) - 4.875 (12.38)

2.6 Duct Inside Flat-to-Flat, in. & (cm) - 4.535 (11.52)

2.7 Bundle/Duct Clearance, in. & (mm) - 0.03 (0.765)

2.8 Duct Wall Thickness, in. & (mm) - 0.101 (2.57)

2.9 Interduct Gap, in. & (mm) - 0.138 (3.51)

2.10 Duct Material

2.10.1 Material Type - 316 SS, 20% CW

2.10.2 Swelling Properties - NSMH Rev. 5

2.10.3 Irradiation Creep Properties - NSMH Rev. 3

2.11 Duct Midwall Axial Temperature Profile[1]

2.11.1 Nominal (see Table C-IV)

2.11.2 2σ (see Table C-V)

2.12 Duct Wall Pressure Differential Axial Profile[1], psi & (kPa) - EOL

2.12.1 Nominal -

$$P_{psi} = \frac{x}{84} \cdot 43 \ (0 \leq x \leq 84)$$

where x = distance from top of fuel pin bundle (in.)

$$P_{kPa} = \frac{x}{84} \cdot 296 \ (0 \leq x \leq 84)$$

2.12.2 2σ - N/A

2.13 Neutron Flux Axial Profile[1] (E > 0.1 MeV), n/cm^2-s - EOL

2.13.1 Nominal -

$$\Phi = 4.57 \times 10^{15} \text{ Cos} \left[1.82 \frac{(x - 12)}{24}\right] (0 \leq x \leq 24)$$

where x = distance from bottom of fuel (in.)

Φ = flux

2.13.2 2σ - N/A

3.0 FUEL PIN DATA

3.1 Fuel Parameters

3.1.1 Fuel Type (oxide, carbide, nitride) - Oxide

3.1.2 Stoichiometry (O/M, C/M, N/M) - 1.94

3.1.3 Plutonium Content (Pu/Pu + U) - 0.1929/0.2339

3.1.4 Fuel Form (powder or pellet) - Pellet

3.1.4.1 Pellet Diameter, in. & (mm) - 0.1935 (4.9)

3.1.4.2 Pellet Dish and Chamfer Dimensions, in. & (mm) - 0 (0)

---

[1]Reported for design limiting duct.

3.1.4.3 Pellet Inside Diameter, in. & (mm) - 0 (0)

3.1.4.4 Pellet Density, g/cm^3 - 10.03

3.1.5 Fuel Smear Density, %TD - 85.5

3.2 Cladding Parameters

3.2.1 Cladding Outside Diameter, in. & (mm) - 0.23 (5.84)

3.2.2 Cladding Wall Thickness, in. & (mm) - 0.015 (0.381)

3.2.3 Diametral Gap, in. & (mm) - 0.0065 (0.16510)

3.2.4 Cladding Material

3.2.4.1 Material Type - 316 SS, 20% CW

3.2.4.2 Swelling Properties - NSMH Rev. 5

3.2.4.3 Irradiation Creep Properties - NSMH Rev. 3

3.2.4.4 Stress-Rupture Properties - LHRFDS Ground Rules

3.3 Stresser Sleeve Parameters - N/A

3.3.1 Sleeve Outside Diameter, in. & (mm)

3.3.2 Sleeve Wall Thickness, in. & (mm)

3.3.3 Fractional Perforation of Sleeve

3.3.4 Sleeve Material

3.4 Equivalent Plenum Volume, in.^{3} & (cc)

3.4.1 Top Plenum - 0.804 (13.179)

3.4.2 Bottom Plenum - N/A

3.5 Bond Type - N/A

3.6 Fuel Pin Linear Power Axial Profile[1], kW/ft - EOL

3.6.1 Nominal -

$$Q = 9.8 \text{ Cos} \left[\frac{1.82\ (x - 12)}{24}\right]$$

where x = distance from bottom of fuel column, in.

Q = local linear power, kW/ft

[1]Design limiting fuel pin.

3.6.2 2σ - N/A

3.7 Cladding Temperature Axial Profile[1], °F (°K)

3.7.1 Nominal OD and ID (see Table C-VI)

3.7.2 2σ OD and ID (see Table C-VII)

3.8 Peak-to-Average Power Ratio - EOL

3.8.1 Nominal - 1.14

3.8.2 2σ - N/A

3.9 Uncertainty Factors for Hot Channel Analysis - LHRFDS Ground Rules

3.10 Neutron Flux Axial Profile[2] (E > 0.1 MeV), n/cm^2-s - EOL

$$\Phi = 3.57 \times 10^{15} \text{ Cos} \left[\frac{1.82\ (x - 12)}{24}\right]$$

where x = distance from bottom of fuel column, in.
Φ = local flux

4.0 RADIAL BLANKET ASSEMBLY DATA

4.1 Pins per Assembly - 61

4.2 Pin Pitch-to-Diameter Ratio - 1.07

4.3 Spacer Description

4.3.1 Wire Wrap Diameter (or grid spacer thickness & height), in. & (mm) - 0.037 (0.94)

4.3.2 Spacer Pitch, in. & (cm) - 11.9 (30.226)

4.3.3 Edge Ratio - 0.98

4.4 Overall Bundle Length, in. & (cm) - 84 (213.36)

4.5 Lattice Pitch, in. & (cm) - 4.875 (12.38)

4.6 Duct Inside Flat-to-Flat, in. & (cm) - 4.535 (11.52)

4.7 Bundle/Duct Clearance, in. & (mm) - 0.03 (0.765)

---

[1]Design limiting fuel pin.

[2]Reported for design limiting duct.

4.8 Duct Wall Thickness, in. & (mm) - 0.101 (2.57)

4.9 Interduct Gap, in. & (mm) - 0.138 (3.51)

4.10 Duct Material

4.10.1 Material Type - 316 SS, 20% CW

4.10.2 Swelling Properties - NSMH Rev. 5

4.10.3 Irradiation Creep Properties - NSMH Rev. 3

4.11 Duct Midwall Axial Temperature Profile[1] - N/A

4.11.1 Nominal

4.11.2 2σ

4.12 Duct Wall Pressure Differential Axial Profile[2], psi & (kPa) - EOL

4.12.1 Nominal - N/A

4.12.2 2σ - N/A

4.13 Neutron Flux Axial Profile[2] (E > 0.1 MeV), n/cm^2-s - EOL

4.13.1 Nominal - N/A

4.13.2 2σ - N/A

5.0 RADIAL BLANKET PIN DATA

5.1 Fuel Parameters

5.1.1 Fuel Type (oxide, carbide, nitride) - Oxide

5.1.2 Stoichiometry (O/M, C/M, N/M) - 1.94

5.1.3 Plutonium Content (Pu/Pu + U) - 0

5.1.4 Fuel Form (powder or pellet) - pellet

5.1.4.1 Pellet Diameter, in. & (mm) - 0.4830 (12.2)

5.1.4.2 Pellet Dish and Chamfer Dimensions, in. & (mm) - 0 (0)

5.1.4.3 Pellet Inside Diameter, in. & (mm) - 0

5.1.4.4 Pellet Density, g/cm^3 - 10.39

---

[1]Reported for design limiting duct.

[2]Design limiting fuel pin.

5.1.5 Fuel Smear Density, %TD - 90

5.2 Cladding Parameters

5.2.1 Cladding Outside Diameter, in. & (mm) - 0.526 (13.36)

5.2.2 Cladding Wall Thickness, in. & (mm) - 0.015 (0.381)

5.2.3 Diametral Gap, in. & (mm) - 0.0065 (0.16510)

5.2.4 Cladding Material

5.2.4.1 Material Type - 316 SS, 20% CW

5.2.4.2 Swelling Properties - NSMH Rev. 5

5.2.4.3 Irradiation Creep Properties - NSMH Rev. 3

5.2.4.4 Stress-Rupture Properties - LHRFDS Ground Rules

5.3 Stresser Sleeve Parameters - N/A

5.3.1 Sleeve Outside Diameter, in. & (mm)

5.3.2 Sleeve Wall Thickness, in. & (mm)

5.3.3 Fractional Perforation of Sleeve

5.3.4 Sleeve Material

5.4 Equivalent Plenum Volume, in.^{3} & (cc)

5.4.1 Top Plenum - 4.94 (81.033)

5.4.2 Bottom Plenum - N/A

5.5 Bond Type - N/A

5.6 Fuel Pin Linear Power Axial Profile[1], kW/ft - EOL

5.6.1 Nominal - N/A

5.6.2 2σ - N/A

5.7 Cladding Temperature Axial Profile[1], °F & (°K)

5.7.1 Nominal OD and ID - N/A

5.7.2 2σ OD and ID - N/A

[1]Design limiting fuel pin.

5.8 Peak-to-Average Power Ratio - EOL

5.8.1 Nominal - 1.20

5.8.2 2σ - N/A

5.9 Uncertainty Factors for Hot Channel Analysis - LHRFDS Ground Rules

5.10 Neutron Flux Axial Profile[1] ($E > 0.1$ MeV), n/cm^2-s - EOL - N/A

6.0 INTERNAL FERTILE ASSEMBLY DATA - N/A

7.0 INTERNAL FERTILE PIN DATA - N/A

8.0 CONTROL ASSEMBLY DATA - N/A

9.0 CONTROL PIN DATA - N/A

10.0 PERFORMANCE CHARACTERISTICS

10.1 Discharge Exposure by Enrichment Zone, MWd/kg

10.1.1 Peak - 107/108

10.1.2 Average - 85/69

10.2 EOL CDF for Design Limiting Fuel Pin - 0.06

10.3 Plenum Pressure History for Design Limiting Fuel Pin - 2σ

$$P = 1.965T + 173$$

where P = pressure (psia)
T = full power days

10.4 Core Material Volume Fractions

10.4.1 Fuel - 0.3763/0.4138

10.4.2 Sodium - 0.3581

10.4.2.1 Fraction of Na in Interduct Gap - 0.141

---

[1]Design limiting fuel pin.

10.4.2.2 Fraction of Na in Assembly Interior - 0.859

10.4.2.3 Fraction of Na in Fuel/Clad Bond - 0.000

10.4.3 Steel - 0.2281

10.4.3.1 Fraction of Steel in Duct - 0.345

10.4.3.2 Fraction of Steel in Wire Wrap - 0.069

10.4.3.3 Fraction of Steel in Cladding - 0.585

10.4.4 Control - 0.0375

10.5 Breeding Ratio - 1.29

10.6 Breeding Gain, kg/cycle - 261

10.7 Compound System Doubling Time, yrs - 17.2

10.8 Specific Power, MW/kg-fissile - 1.06

10.9 Fuel Cycle Costs, mills/kW-hr - 4.43

10.10 Assembly Exposure, MWd/assembly (Zone 1/Zone 2)

10.10.1 Peak - 2761/2700

10.10.2 Average - 1923/1068

10.11 Sodium Void Worth, Fresh Core/EOEC ($) - 1.27/3.16

10.12 Doppler Coefficient - N/A

10.13 LHRFDS Optimization Function - 2.73

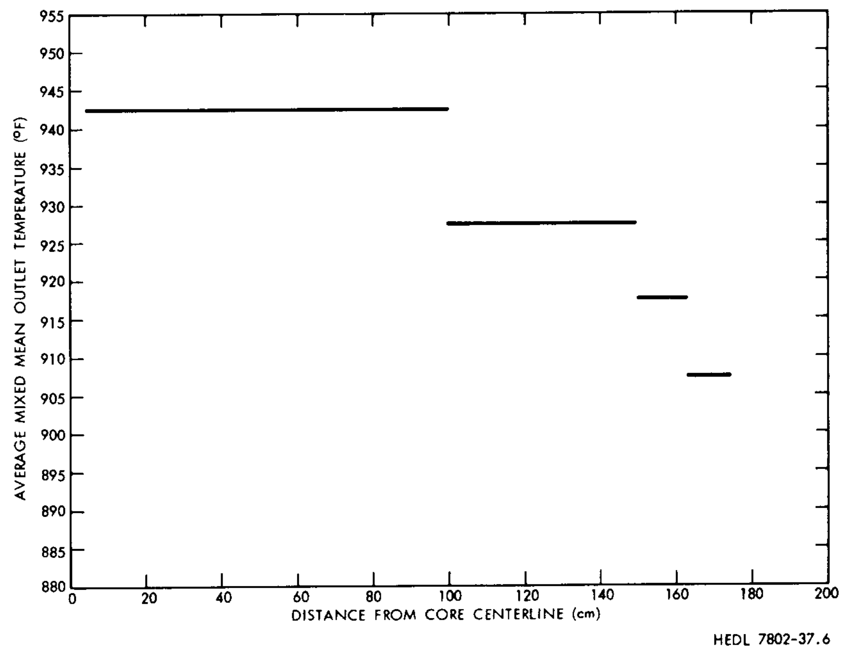
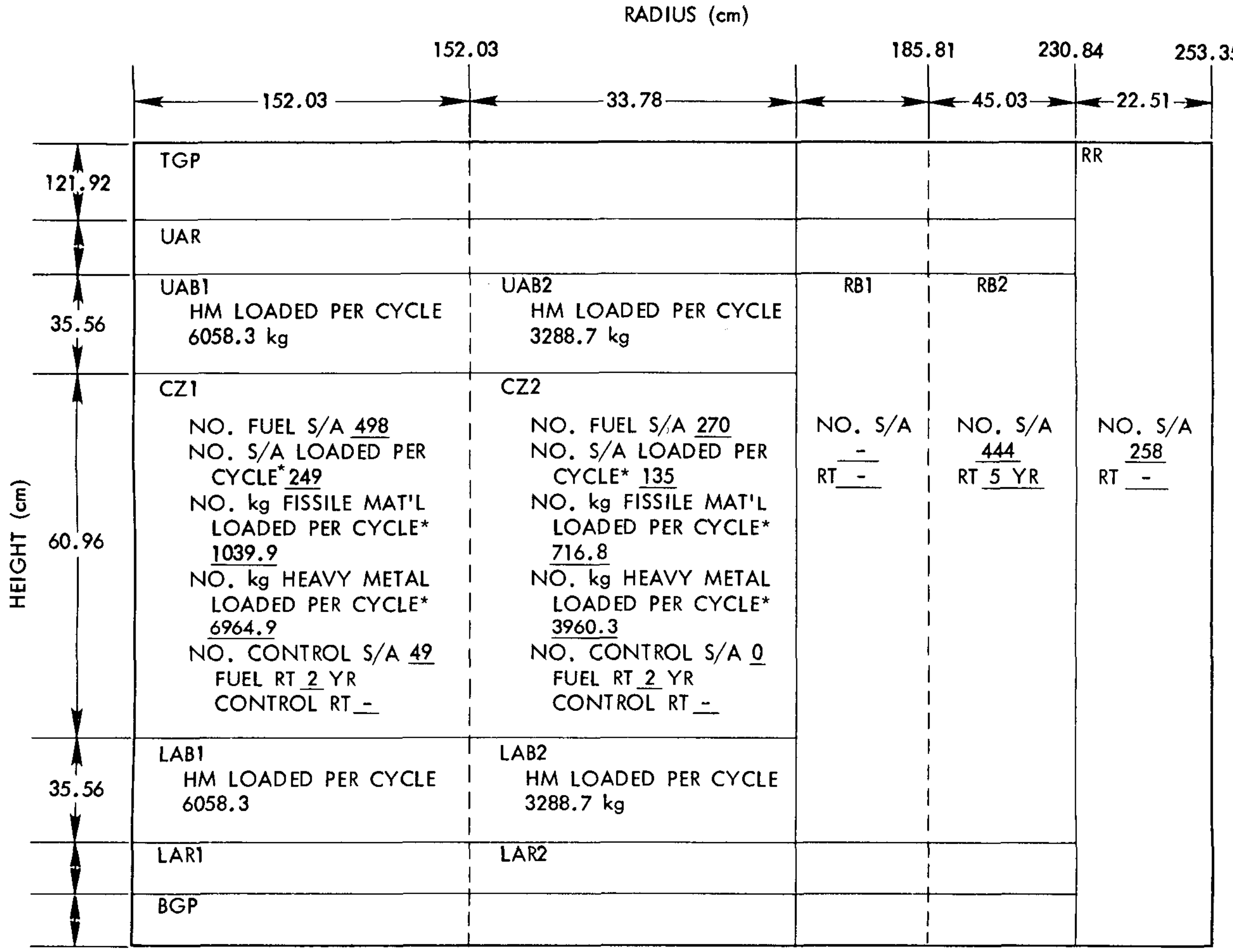


FIGURE C-1. Assembly Coolant Mixed Mean Outlet Temperature (EOL).



TGP = TOP GAS PLENUM
BGP = BOTTOM GAS PLENUM
UAR = UPPER AXIAL REFLECTOR
LAR = LOWER AXIAL REFLECTOR
UAB = UPPER AXIAL BLANKET
LAB = LOWER AXIAL BLANKET
RB1 = ZONE 1 RADIAL BLANKET
RB2 = ZONE 2 RADIAL BLANKE
CZ1 = CORE ZONE 1
CZ2 = CORE ZONE 2

*TO BE SPECIFIED AT START OF EQUILIBRIUM CYCLE.

FIGURE C-2. R-Z Core Diagram for LHRFDS Level II Homogeneous Design.

TABLE C-I

FUEL INVENTORY AT THE BEGINNING OF EQUILIBRIUM CYCLE
(kg)

Isotope	Core Region CZ1	CZ2	UAB1	UAB2	LAB1	LAB2	RB
^{235}U	19.46	11.07	23.1	12.9	23.1	12.9	93.88
^{236}U	Isotope Not Evaluated						
^{238}U	10966	5906	12010	6536	12010	6536	48800
^{239}Pu	1750.6	1197.0	77.75	25.86	77.75	25.86	306.2
^{240}Pu	574.2	386.6	1.68	0.35	1.68	0.35	5.35
^{241}Pu	237.8	169.7	0.05	0.007	0.05	0.007	0.151
^{242}Pu	68.4	46.2	5×10^{-4}	5×10^{-5}	5×10^{-4}	5×10^{-5}	0.002
Fission Products	309.2	142.0	8.4	2.4	8.4	2.4	30.7

TABLE C-II

FUEL INVENTORY AT THE MIDDLE OF EQUILIBRIUM CYCLE
(kg)

Isotope	Core Region CZ1	CZ2	UAB1	UAB2	LAB1	LAB2	RB
^{235}U	16.4	9.89	21.6	12.35	21.6	12.35	92.4
^{236}U	Isotope Not Evaluated						
^{238}U	10720	5876	11920	6508	11920	6508	48700
^{239}Pu	1696.2	1150.2	150.6	50.54	150.6	50.54	379.8
^{240}Pu	603	398	3.97	0.84	3.97	0.84	8.2
^{241}Pu	207.4	153.2	0.14	0.02	0.14	0.02	0.28
^{242}Pu	71.9	47.9	0.002	2×10^{-4}	0.002	2×10^{-4}	0.004
Fission Products	606.8	280.4	18.0	5.06	18.0	5.06	43.0

TABLE C-III

FUEL INVENTORY AT THE END OF EQUILIBRIUM CYCLE
(kg)

Isotope	Core Region CZ1	CZ2	UAB1	UAB2	LAB1	LAB2	RB
^{235}U	13.8	8.8	20.2	11.9	20.2	11.9	90.9
^{236}U	Isotope Not Evaluated						
^{238}U	10476	5786	11840	6480	11840	6480	48620
^{239}Pu	1645.2	1106.2	220.6	75.1	220.6	75.1	455.2
^{240}Pu	629.2	408.4	7.7	1.7	7.7	1.7	11.7
^{241}Pu	183.2	139.0	0.35	0.05	0.35	0.05	0.47
^{242}Pu	74.5	49.2	0.007	7×10^{-4}	0.007	7×10^{-4}	0.015
Fission Products	895.6	415.8	31.5	8.6	31.5	8.6	57.6

TABLE C-IV

MIDWALL AXIAL TEMPERATURE PROFILE FOR DESIGN LIMITING DUCT
(Nominal, EOL)

Distance Above BOF (ft)	Duct Midwall Temperature (°F)	(°K)
0	602	590
0.15385	612	595
0.30769	622	601
0.46154	634	607
0.61538	646	614
0.76923	659	621
0.92308	673	629
1.0769	687	637
1.2308	700	644
1.3846	713	651
1.5385	726	659
1.6923	738	665
1.8462	748	671
2.0000	757	676

TABLE C-V

MIDWALL AXIAL TEMPERATURE PROFILE FOR DESIGN LIMITING DUCT
(2σ, EOL)

Distance Above BOF (ft)	Duct Midwall Temperature (°F)	(°K)
0	620	600
0.15385	632	606
0.30769	646	614
0.46154	660	622
0.61538	676	631
0.76923	693	640
0.92308	711	650
1.0769	728	660
1.2308	746	670
1.3846	763	679
1.5385	779	688
1.6923	794	696
1.8462	807	704
2.0000	819	710

TABLE C-VI

NOMINAL CLADDING TEMPERATURE AXIAL PROFILES FOR DESIGN LIMITING FUEL PIN

Distance from Bottom of Fuel (in.)	Beginning of Life: Clad OD Temperature (°F)	(°K)	Clad ID Temperature (°F)	(°K)	End of Life: Clad OD Temperature (°F)	(°K)	Clad ID Temperature (°F)	(°K)
0.857	638	610	695	641	634	607	682	634
2.571	664	624	729	660	656	620	710	650
4.286	693	640	763	679	680	633	739	666
6.000	724	657	799	699	706	647	769	682
7.714	757	676	835	719	734	663	800	700
9.429	791	695	871	739	762	679	829	716
11.143	826	714	905	758	791	695	859	732
12.857	860	733	939	777	820	711	887	748
14.571	894	752	970	794	848	726	913	762
16.286	926	770	999	810	875	741	938	776
18.000	957	787	1026	825	901	756	960	789
19.714	985	802	1048	837	925	769	978	799
21.429	1011	817	1067	848	946	781	994	807
23.143	1033	829	1083	857	965	791	1007	815
24.000	1043	835	1089	860	973	796	1012	817

TABLE C-VII

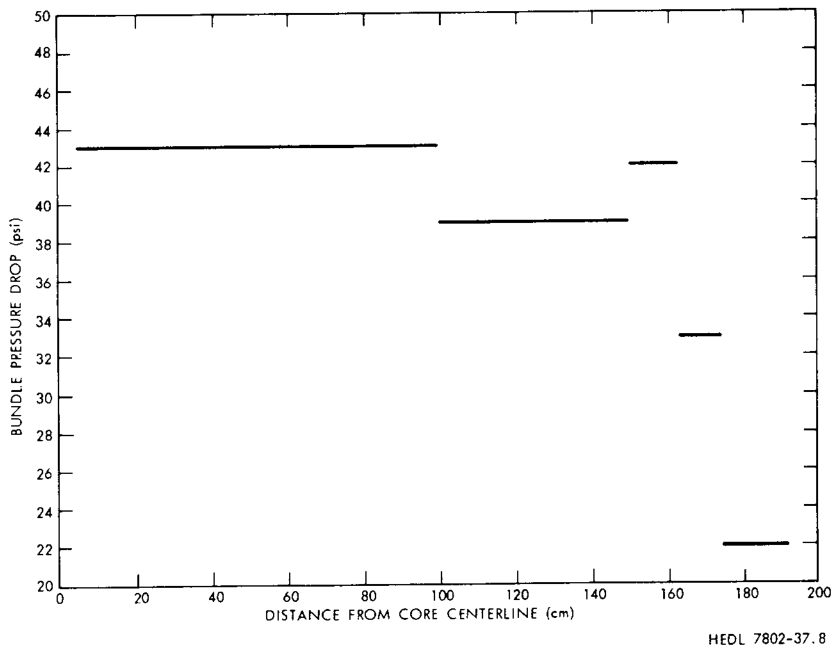
2σ CLADDING TEMPERATURE AXIAL PROFILES FOR DESIGN LIMITING FUEL PIN

Distance from Bottom of Fuel (in.)	Beginning of Life: Clad OD Temperature (°F)	(°K)	Clad ID Temperature (°F)	(°K)	End of Life: Clad OD Temperature (°F)	(°K)	Clad ID Temperature (°F)	(°K)
0.857	699	644	776	686	688	637	752	673
2.571	739	666	824	713	721	656	793	696
4.286	780	689	872	740	755	675	834	719
6.000	824	713	921	767	792	695	875	741
7.714	869	738	969	794	830	716	916	764
9.429	914	763	1015	819	868	737	954	785
11.143	960	789	1060	844	906	759	992	806
12.857	1004	813	1102	867	942	779	1027	826
14.571	1046	836	1140	889	978	799	1060	844
16.286	1086	859	1175	908	1011	817	1089	860
18.000	1122	879	1206	925	1042	834	1115	875
19.714	1155	897	1231	939	1069	849	1136	886
21.429	1183	912	1251	950	1092	862	1152	895
23.143	1206	925	1265	958	1112	873	1164	902
24.000	1216	931	1271	961	1120	877	1169	905

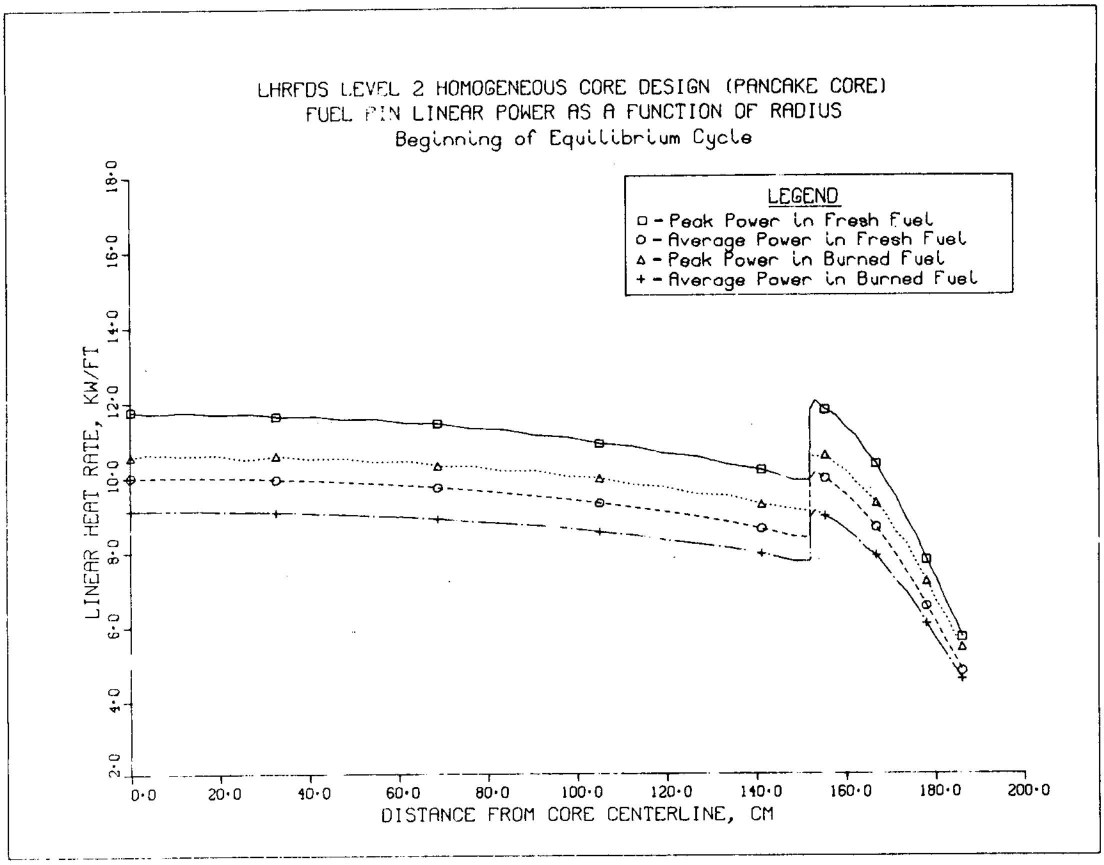
TABLE C-VIII

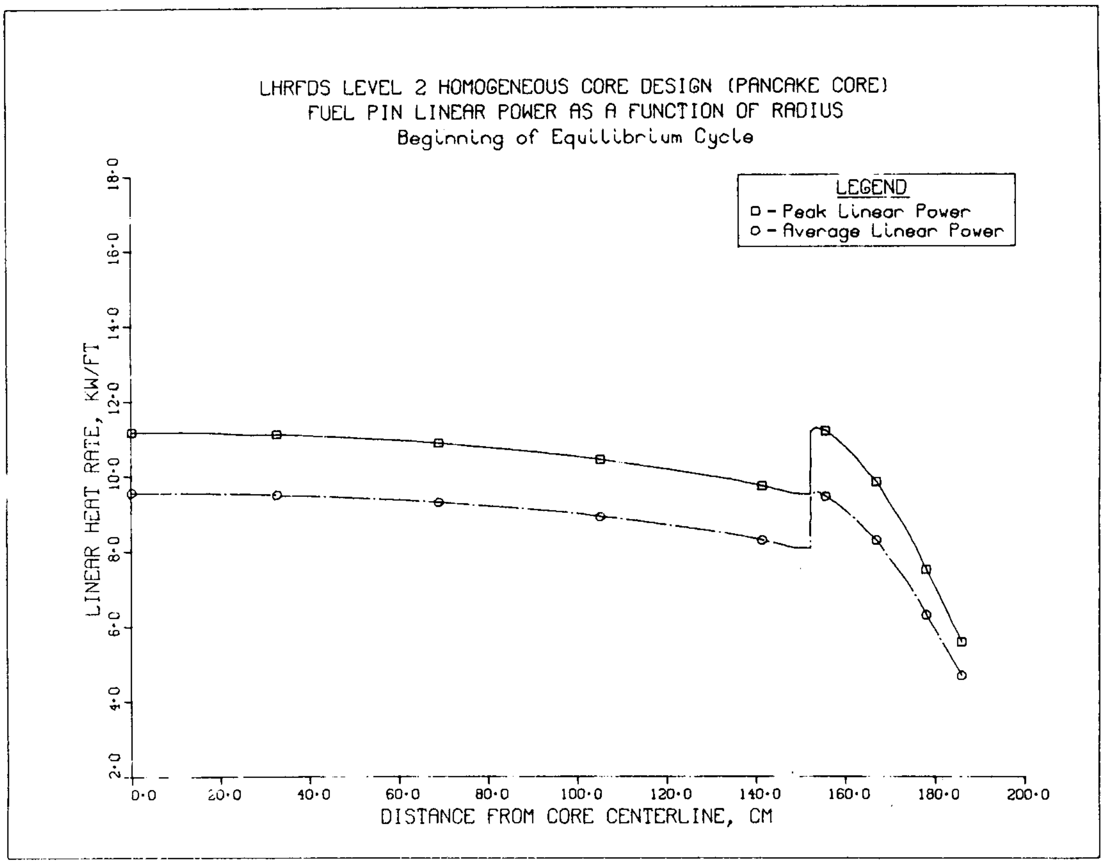
HEAVY METAL AND FISSILE Pu CHARGE AND DISCHARGE INFORMATION
FOR THE HEDL LEVEL II HOMOGENEOUS DESIGN

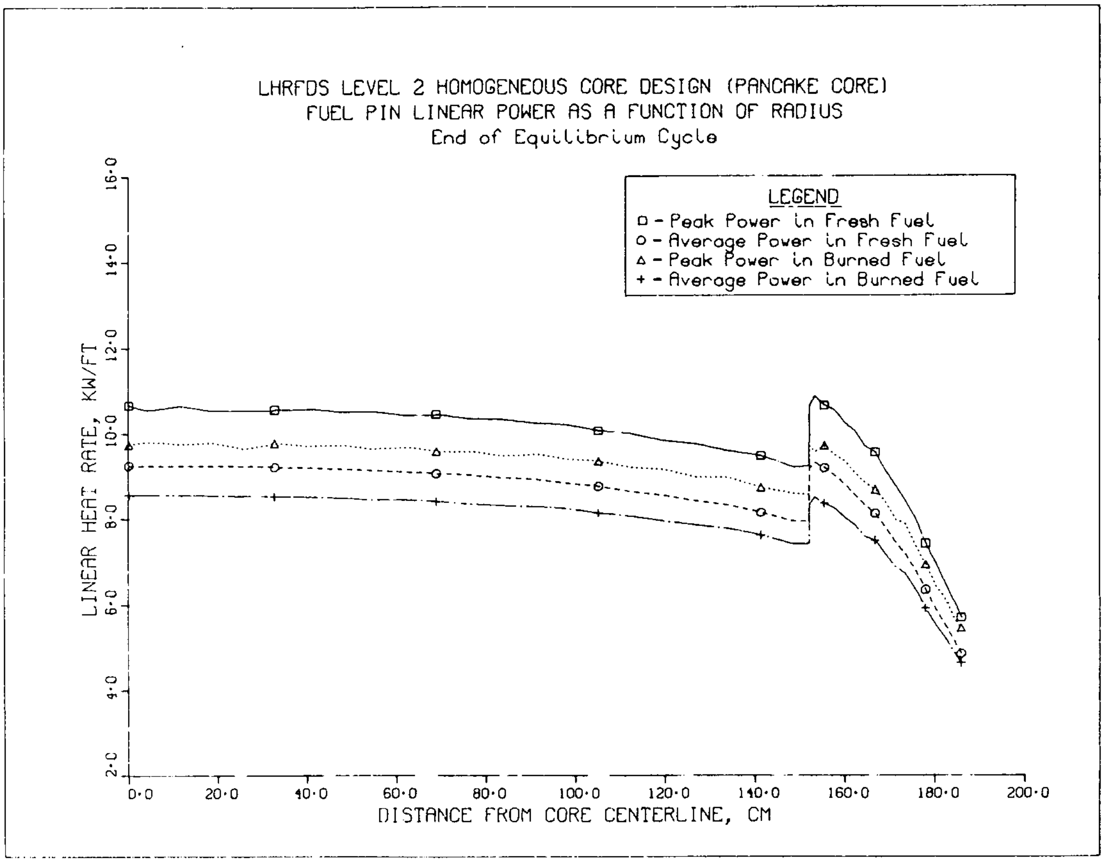
		Heavy Metal		Fissile Pu		
					Discharge	
Year	Capacity Factor	Core	Blanket	Core Feed	Core	Blanket
1	0.700	21855	86536	3512	1600	231
2		10927	28542	1756	1475	481
3						534
4						
5						
6						
.						
.						
16						
17						
18						
19						
20						
21						
22						
23						
24						
25						
26						
27						
28						
29						
30					3074	1045

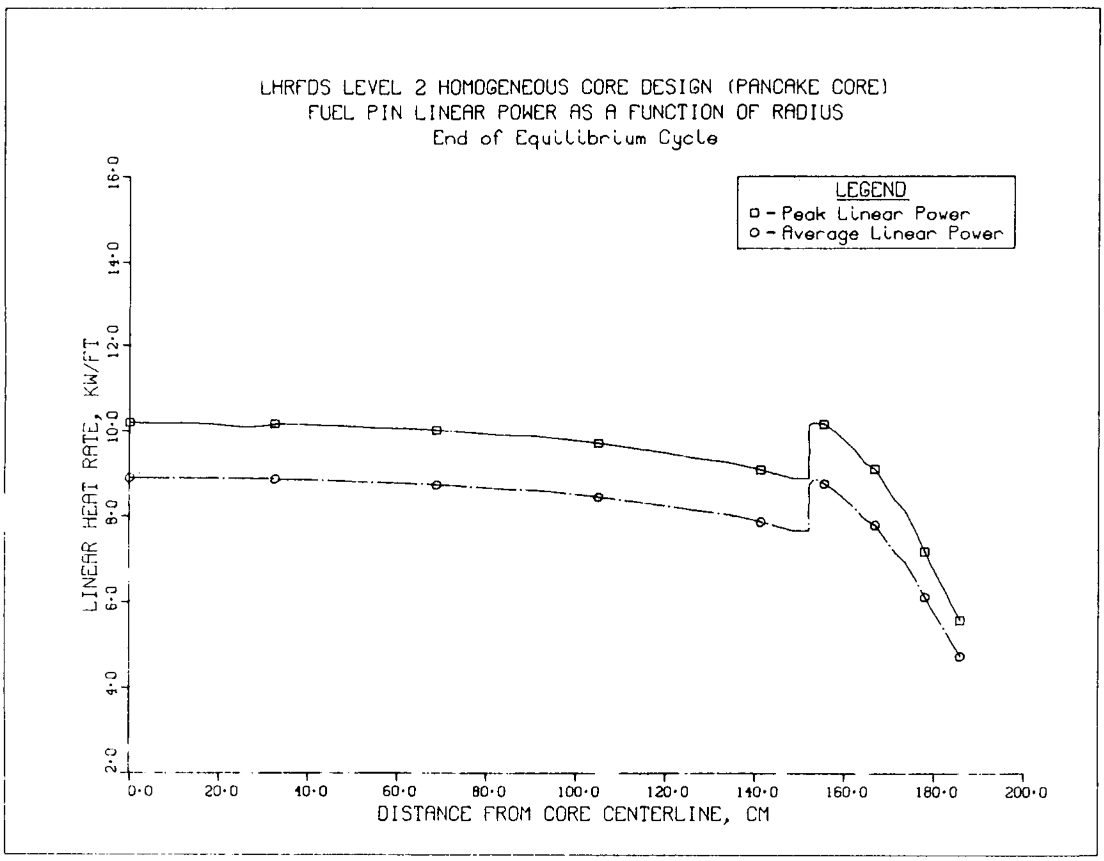


BUNDLE PRESSURE DROP


LHRFDS LEVEL 2 HOMOGENEOUS CORE DESIGN (PANCAKE CORE)
FUEL PIN LINEAR POWER AS A FUNCTION OF RADIUS
Beginning of Equilibrium Cycle
LEGEND
□ - Peak Power in Fresh Fuel
○ - Average Power in Fresh Fuel
△ - Peak Power in Burned Fuel
+ - Average Power in Burned Fuel
LINEAR HEAT RATE, KW/FT
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
0.0
20.0
40.0
60.0
80.0
100.0
120.0
140.0
160.0
180.0
200.0
DISTANCE FROM CORE CENTERLINE, CM


LHRFDS LEVEL 2 HOMOGENEOUS CORE DESIGN (PANCAKE CORE)
FUEL PIN LINEAR POWER AS A FUNCTION OF RADIUS
Beginning of Equilibrium Cycle
LEGEND
□ - Peak Linear Power
○ - Average Linear Power
LINEAR HEAT RATE, KW/FT
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
0.0
20.0
40.0
60.0
80.0
100.0
120.0
140.0
160.0
180.0
200.0
DISTANCE FROM CORE CENTERLINE, CM


LHRFDS LEVEL 2 HOMOGENEOUS CORE DESIGN (PANCAKE CORE)
FUEL PIN LINEAR POWER AS A FUNCTION OF RADIUS
End of Equilibrium Cycle
LEGEND
□ - Peak Power in Fresh Fuel
○ - Average Power in Fresh Fuel
△ - Peak Power in Burned Fuel
+ - Average Power in Burned Fuel
LINEAR HEAT RATE, KW/FT
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
0.0
20.0
40.0
60.0
80.0
100.0
120.0
140.0
160.0
180.0
200.0
DISTANCE FROM CORE CENTERLINE, CM


LHRFDS LEVEL 2 HOMOGENEOUS CORE DESIGN (PANCAKE CORE)
FUEL PIN LINEAR POWER AS A FUNCTION OF RADIUS
End of Equilibrium Cycle
LEGEND
□ - Peak Linear Power
○ - Average Linear Power
LINEAR HEAT RATE, KW/FT
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
0.0
20.0
40.0
60.0
80.0
100.0
120.0
140.0
160.0
180.0
200.0
DISTANCE FROM CORE CENTERLINE, CM

APPENDIX D

SPECIFICATION OF LHRFDS LEVEL II HETEROGENEOUS CORE DESIGN IN THE ALSDAWG FORMAT

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May 1977

1.0 CORE AND REACTOR DATA

1.1 Power Information

1.1.1 Plant Thermal Power, MW_t - 3750

1.1.2 Plant Electric Power, MWe - 1200

1.1.2.1 Net Electric Power, MWe - 1185

1.1.3 Plant Capacity Factor versus Time - 0.70 (constant)

1.1.4 Power Split, Fraction of Total (MOEC)

1.1.4.1 Core Fuel - 0.758

1.1.4.2 Axial Blanket - 0.012

1.1.4.3 Radial Blanket - 0.061

1.1.4.4 Internal Blanket - 0.169

1.1.4.5 Control - 0

1.1.4.6 Radial Shielding - 0

1.1.4.7 Other - 0

1.1.5 Average Linear Power (MOEC)

1.1.5.1 Core Fuel, kW/ft & (W/cm) - 6.911 (226.7)

1.1.5.2 Axial Blanket, kW/ft & (W/cm) - 0.188 (6.2)

1.1.5.3 Radial Blanket, kW/ft & (W/cm) - 0.984 (32.3)

1.1.5.4 Internal Fertile Assembly, kW/ft & (W/cm) - 3.617 (118.7)

1.1.6 Fission Energy and Deposition, MeV/fission - 215, deposited locally

1.2 Temperature Information

1.2.1 Core Inlet Temperature, °F & (°K) - 595 (589)

1.2.2 Core Average Outlet Temperature, °F & (°K) - N/A

1.2.3 Core ΔT, °F & (°K) - N/A

1.2.4 Reactor Inlet Temperature, °F & (°K) - 595 (589)

1.2.5 Reactor Outlet Temperature, °F & (°K) - 895 (752)

1.2.6 Reactor ΔT, °F & (°K) - 300 (167)

1.2.7 Number of Core Orifice Zones - 7 Fuel, 6 Internal Blanket

1.2.8 Radial Profile of Assembly Outlet Temperature - Figure D-1

1.2.9 Core Orificing Criteria - Core orificed to provide 2-1/2 year lifetime for all components

1.3 Coolant Information

1.3.1 Peak Power Assembly Pressure Drop, psi & (kPa) - ∿ 120 (828)

1.3.2 Reactor Pressure Drop, psi & (kPa) - N/A

1.3.3 Primary System Pressure Drop, psi & (kPa) - ∿ 170 (1173)

1.3.4 Flow Split, Fraction of Total

1.3.4.1 Core - 59%

1.3.4.2 Internal Fertile Assembly - 26%

1.3.4.3 Radial Blanket

1.3.4.4 Control 15%

1.3.4.5 Radial Shielding

1.3.4.6 Other

1.3.5 Total Coolant Mass Flow Rate, lb_m/hr & (kg/hr) - 1.399×10^8 (6.358×10^7)

1.3.6 Maximum Coolant Velocity, ft/s & (m/s) - 30.60 (9.33)

1.4 Geometric Information (see Figure D-2)

1.4.1 Core Height, in. & (cm) - 48 (121.9)

1.4.2 Axial Blanket Height, in. & (cm) - 14 (35.56)

1.4.3 Radial Blanket Height, in. & (cm) - 76 (193.0)

1.4.4 Axial Shield Height, in. & (cm) - N/A

1.4.5 Number of Core Enrichment Zones - 2

1.4.6 Number of Assemblies, Fuel/IB/RB/Control - 378/217/288/36

1.4.7 Equivalent Diameters[1], in. & (cm) - Figure D-2

1.5 Fuel Management

1.5.1 Refueling Interval, calendar days - 456

[1]Diameter of equivalent volume cylinder.

1.5.2 Fuel Residence Time, full power days (see Figure D-2) - 639

1.5.3 Blanket Residence Time, full power days

1.5.3.1 Radial Blankets - 1597

1.5.3.2 Internal Blankets - 639

1.5.4 Fuel Inventory, kg (see Tables D-I through D-VI)

1.5.5 Fraction of Assemblies Replaced at Each Refueling

1.5.5.1 Fuel Assemblies by Enrichment Zone - 0.5

1.5.5.2 Radial Blanket Assemblies - 0.2

1.5.5.3 Interior Fertile Assemblies - 0.5

2.0 FUEL ASSEMBLY DATA

2.1 Pins per Assembly - 271

2.2 Pin Pitch-to-Diameter Ratio - 1.207

2.3 Spacer Description

2.3.1 Wire Wrap Diameter (or grid spacer thickness & height), in. & (mm) - 0.048 (1.2)

2.3.2 Spacer Pitch, in. & (cm) - 11.9 (30.226)

2.3.3 Edge Ratio - 0.985

2.4 Overall Bundle Length, in. & (cm) - 140 (355.6)

2.5 Lattice Pitch, in. & (cm) - 5.095 (12.94)

2.6 Duct Inside Flat-to-Flat, in. & (cm) - 4.685 (11.90)

2.7 Bundle/Duct Clearance, in. & (mm) - 0.03 (0.765)

2.8 Duct Wall Thickness, in. & (mm) - 0.102 (2.59)

2.9 Interduct Gap, in. & (mm) - 0.206 (5.23)

2.10 Duct Material

2.10.1 Material Type - 316 SS, 20% CW

2.10.2 Swelling Properties - NSMH Rev. 5

2.10.3 Irradiation Creep Properties - NSMH Rev. 3

2.11 Duct Midwall Axial Temperature Profile[1]

2.11.1 Nominal (see Table D-VII)

2.11.2 2σ (see Table D-VIII)

2.12 Duct Wall Pressure Differential Profile[1], psi & (kPa) - EOL

2.12.1 Nominal -

$$P_{psi} = \frac{x}{140} \cdot 90 \quad (0 \leq x \leq 140)$$

where x = distance from top of fuel pin bundle (in.)

$$P_{kPa} = \frac{x}{140} \cdot 620 \quad (0 \leq x \leq 140)$$

2.12.2 2σ - N/A

2.13 Neutron Flux Axial Profile[1] (E > 0.1 MeV), n/cm^2-s - EOL

2.13.1 Nominal -

$$\Phi = 3.44 \times 10^{15} \text{ Cos } \left[2.36 \frac{(x - 24)}{48}\right] \quad (0 \leq x \leq 48)$$

where x = distance from bottom of fuel column (in.)

2.13.2 2σ - N/A

3.0 FUEL PIN DATA

3.1 Fuel Parameters

3.1.1 Fuel Type (oxide, carbide, nitride) - Oxide

3.1.2 Stoichiometry (O/M, C/M, N/M) - 1.96

3.1.3 Plutonium Content (Pu/Pu + U) - 0.252/0,293

3.1.4 Fuel Form (powder or pellet) - Pellet

3.1.4.1 Pellet Diameter, in. & (mm) - 0.1935 (4.9)

3.1.4.2 Pellet Dish and Chamfer Dimensions, in. & (mm) - 0 (0)

[1]Reported for design limiting duct.

3.1.4.3 Pellet Inside Diameter, in. & (mm) - 0 (0)

3.1.4.4 Pellet Density, g/cm^3 - 10.03

3.1.5 Fuel Smear Density, %TD - 85.5

3.2 Cladding Parameters

3.2.1 Cladding Outside Diameter, in. & (mm) - 0.23 (5.84)

3.2.2 Cladding Wall Thickness, in. & (mm) - 0.015 (0.381)

3.2.3 Diametral Gap, in. & (mm) - 0.0065 (0.1651)

3.2.4 Cladding Material

3.2.4.1 Material Type - 316 SS, 20% CW

3.2.4.2 Swelling Properties - NSMH Rev. 5

3.2.4.3 Irradiation Creep Properties - NSMH Rev. 3

3.2.4.4 Stress-Rupture Properties - LHRFDS Ground Rules

3.3 Stresser Sleeve Parameters - N/A

3.3.1 Sleeve Outside Diameter, in. & (mm)

3.3.2 Sleeve Wall Thickness, in. & (mm)

3.3.3 Fractional Perforation of Sleeve

3.3.4 Sleeve Material

3.4 Equivalent Plenum Volume, $in.^3$ & (cc)

3.4.1 Top Plenum - 1.71 (28.01)

3.4.2 Bottom Plenum - N/A

3.5 Bond Type - N/A

3.6 Fuel Pin Linear Power Axial Profile[1], kW/ft - EOL

3.6.1 Nominal -

$$Q = 6.40 \text{ Cos} \left[2.18 \frac{(x - 24)}{48}\right]$$

where x = distance from bottom of fuel column, in.

Q = local linear power, kW/ft

[1]Reported for design limiting fuel pin.

3.6.2 2σ - N/A

3.7 Cladding Temperature Axial Profile, °F (°K)

3.7.1 Nominal OD and ID (see Table D-IX)

3.7.2 2σ OD and ID (see Table D-X)

3.8 Peak-to-Average Pin Linear Power Ratio - EOL

3.8.1 Nominal Axially -1.23

3.8.2 2σ Axially - N/A

3.9 Uncertainty Factors for Hot Channel Analysis - LHRFDS Ground Rules

3.10 Neutron Flux Axial Profile[1] (E > 0.1 MeV), n/cm^2-s - EOL

$$\Phi = 2.19 \times 10^{15} \text{ Cos} \left[2.18 \frac{(x - 24)}{48}\right] \quad (0 \le x \le 48)$$

where x = distance from bottom of fuel column, in.

Φ = local flux

4.0 RADIAL BLANKET ASSEMBLY DATA - Same as Internal Fertile Assembly (see Section 6.0)

5.0 RADIAL BLANKET PIN DATA - Same as Internal Fertile Pin (see Section 7.0)

6.0 INTERNAL FERTILE ASSEMBLY DATA

6.1 Pins per Assembly - 127

6.2 Pin Pitch-to-Diameter Ratio - 1.11

6.3 Spacer Description

6.3.1 Wire Wrap Diameter (or grid spacer thickness and height), in. & (mm) - 0.040 (1.02)

6.3.2 Spacer Pitch, in. & (cm) - 12 (30.5)

6.3.3 Edge Ratio - 0.967

6.4 Overall Assembly Length, in. & (cm) - 140 (355.6)

6.5 Lattice Pitch, in. & (cm) - 5.095 (12.94)

[1]Reported for design limiting fuel pin.

6.6 Duct Inside Flat-to-Flat, in. & (cm) - 4.685 (11.90)

6.7 Bundle/Duct Clearance, in. & (mm) - 0.03 (0.765)

6.8 Duct Wall Thickness, in. & (mm) - 0.102 (2.59)

6.9 Interduct Gap, in. & (mm) - 0.206 (5.23)

6.10 Duct Material

6.10.1 Material Type - 316 SS, 20% CW

6.10.2 Swelling Properties - NSMH Rev. 5

6.10.3 Irradiation Creep Properties - NSMH Rev. 3

6.11 Duct Midwall Axial Temperature Profile[1], °F (°K)

6.11.1 Nominal (see Table D-XI)

6.11.2 2σ (see Table D-XII)

6.12 Duct Wall Pressure Differential Profile[1], psi & (kPa) - EOL

6.12.1 Nominal -

$$P_{psi} = \frac{x}{140} \cdot 92.0 \quad (0 \le x \le 140)$$

where x = distance from top of fuel pin bundle (in.)

$$P_{kPa} = \frac{x}{140} \cdot 633 \quad (0 \le x \le 140)$$

6.12.2 2σ - N/A

6.13 Neutron Flux Axial Profile[1] (E > 0.1 MeV), n/cm^2-s - EOL

6.13.1 Nominal -

$$\Phi = 3.26 \times 10^{15} \text{ Cos} \left[3.06 \frac{(x - 24)}{48}\right] \quad (0 \le x \le 48)$$

where x = distance from bottom of fuel column (in.)

6.13.2 2σ - N/A

[1]Reported for design limiting duct.

7.0 INTERNAL FERTILE PIN DATA

7.1 Fuel Parameters

7.1.1 Fuel Type (oxide, carbide, nitride) - Oxide

7.1.2 Stoichiometry (O/M, C/M, N/M) - 1.96

7.1.3 Plutonium Content (Pu/Pu + U) - 0

7.1.4 Fuel Form (powder or pellet) - Pellet

7.1.4.1 Pellet Diameter, in. & (mm) - 0.330 (8.4)

7.1.4.2 Pellet Dish and Chamfer Dimensions, in. & (mm) - 0 (0)

7.1.4.3 Pellet Inside Diameter, in. & (mm) - 0 (0)

7.1.4.4 Pellet Density, g/cm^3 - 10.56

7.1.5 Fuel Smear Density, %TD - 93.7

7.2 Cladding Parameters

7.2.1 Cladding Outside Diameter, in. & (mm) - 0.365 (9.27)

7.2.2 Cladding Wall Thickness, in. & (mm) - 0.015 (0.381)

7.2.3 Diametral Gap, in. & (mm) - 0.005 (0.127)

7.2.4 Cladding Material

7.2.4.1 Material Type - 316 SS, 20% CW

7.2.4.2 Swelling Properties - NSMH Rev. 5

7.2.4.3 Irradiation Creep Properties - NSMH Rev. 3

7.2.4.4 Stress-Rupture Properties - LHRFDS Ground Rules

7.3 Stresser Sleeve Parameters - N/A

7.3.1 Sleeve Outside Diameter, in. & (mm)

7.3.2 Sleeve Wall Thickness, in. & (mm)

7.3.3 Fractional Perforation of Sleeve

7.3.4 Sleeve Material

7.4 Equivalent Plenum Volume, in.^{3} & (cc)

7.4.1 Top Plenum - 4.79 (78.6)

7.4.2 Bottom Plenum - N/A

7.5 Bond Type (sodium or helium) - N/A

7.6 Fuel Pin Linear Power Axial Profile[1], kW/ft - EOL

7.6.1 Nominal -

$$Q = 14.77 \text{ Cos} \left[3.06 \frac{(x - 24)}{48}\right]$$

where x = distance from bottom of fuel column (in.)
Q = local linear power (kW/ft)

7.6.2 2σ - N/A

7.7 Cladding Temperature Axial Profile, °F & (°K)

7.7.1 Nominal OD and ID (see Table D-XIII)

7.7.2 2σ OD and ID (see Table D-XIV)

7.8 Peak-to-Average Pin Linear Power Ratio - EOL

7.8.1 Nominal Axially - 1.45

7.8.2 2σ Axially - N/A

7.9 Uncertainty Factors for Hot Channel Analysis - LHRFDS Ground Rules

7.10 Neutron Flux Axial Profile[1] (E > 0.1 MeV), n/cm^2-s - EOL

$$\Phi = 3.17 \times 10^{15} \text{ Cos} \left[3.06 \frac{(x - 24)}{48}\right] \quad (0 \le x \le 48)$$

where x = distance from bottom of fuel column (in.)
Φ = local flux

8.0 CONTROL ASSEMBLY DATA - N/A

9.0 CONTROL PIN DATA - N/A

10.0 PERFORMANCE CHARACTERISTICS

10.1 Discharge Exposure, MWd/kg

10.1.1 Peak - 122
10.1.2 Average - 86

---

[1]Reported for design limiting pin.

10.2 EOL CDF for Design Limiting Fuel Pin - 0.25

10.3 Plenum Pressure History for Design Limiting Fuel Pin - 2σ

$$P = 167 + 1.038T$$

where P = pressure (psia)

T = full power days

10.4 Core Material Volume Fractions

10.4.1 Fuel - 0.4979/0.3786/0.4979/0.3245/0.4979/0.3786/0.4979/0.3118/0.4979/0.3786/0.4979/0.3506

10.4.2 Sodium (Blankets/Fuel) - 0.3266/0.4025

10.4.2.1 Fraction of Na in Interduct Gap - 0.2425/0.1968

10.4.2.2 Fraction of Na in Assembly Interior - 0.7575/0.8032

10.4.2.3 Fraction of Na in Fuel/Clad Bond - 0.000

10.4.3 Steel - 0.1755/0.2188

10.4.3.1 Fraction of Steel in Duct - 0.4287/0.3438

10.4.3.2 Fraction of Steel in Wire Wrap - 0.0404/0.0980

10.4.3.3 Fraction of Steel in Cladding - 0.5309/0.5581

10.5 Breeding Ratio - 1.31

10.6 Breeding Gain, kg/cycle - 362

10.7 Compound System Doubling Time, yrs - 19.0

10.8 Specific Power, MW/kg-fissile - 0.83

10.9 Fuel Cycle Costs, mills/kW-hr - 2.95

10.10 Assembly Exposure, MWd/assembly

10.10.1 Peak - 5375

10.10.2 Average - 4870

10.11 Sodium Void Worth* (BOEC/EOEC), $ - +1.81/+2.53

10.12 Doppler Coefficient - N/A

10.13 LHRFDS Optimization Function - 2.24

*Worth of voiding flowing sodium, active core height only. No voiding of control, axial or radial blankets, nor interduct gaps.

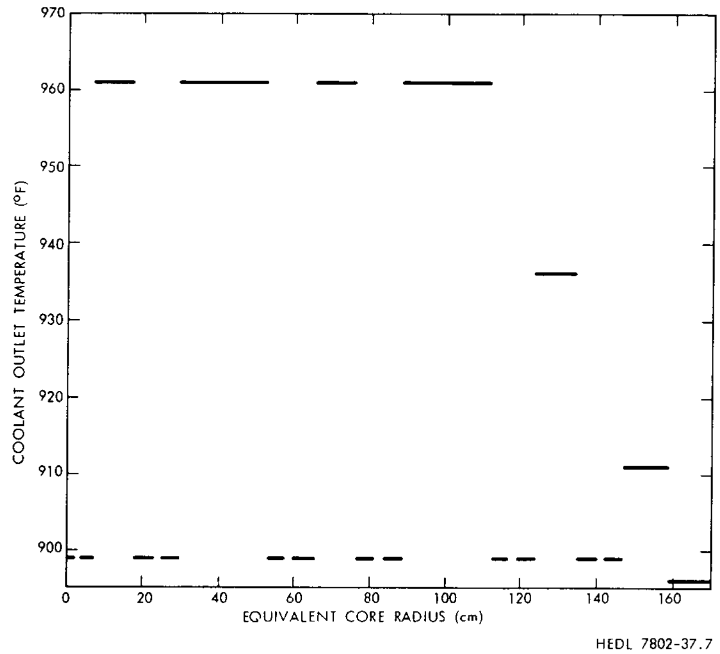
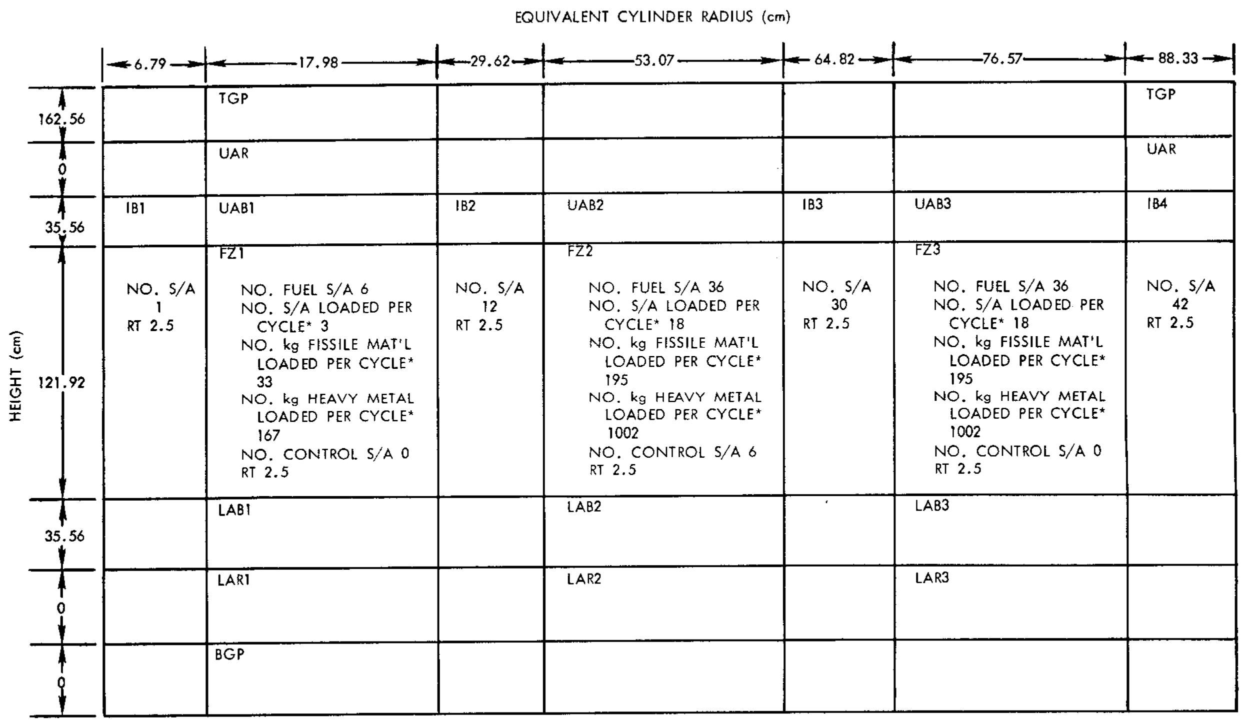


FIGURE D-1. Assembly Coolant Mixed Mean Outlet Temperature (EOL).



TGP = TOP GAS PLENUM
BGP = BOTTOM GAS PLENUM
UAR = UPPER AXIAL REFLECTOR
LAR = LOWER AXIAL REFLECTOR

UAB = UPPER AXIAL BLANKET
LAB = LOWER AXIAL BLANKET
RB = RADIAL BLANKET

IB1 = INTERIOR BLANKET 1
IB2 = INTERIOR BLANKET 2
IB3 = INTERIOR BLANKET 3
IB4 = INTERIOR BLANKET 4
IB5 = INTERIOR BLANKET 5
IB6 = INTERIOR BLANKET 6

FZ1 = FUEL ZONE 1
FZ2 = FUEL ZONE 2
FZ3 = FUEL ZONE 3
FZ4 = FUEL ZONE 4
FZ5 = FUEL ZONE 5
FZ6 = FUEL ZONE 6

*SPECIFIED FOR START OF EQUILIBRIUM CYCLE.

FIGURE D-2. R-Z Core Diagram for LHRFDS Level II Heterogeneous Design.

EQUIVALENT CYLINDER RADIUS (cm)

111.85	123.62	135.38	147.15	170.68	205.98	217.75
				TGP		RR
				UAR		
UAB4	IB5	UAB5	IB6	UAB6	RB	
FZ4 NO. FUEL S/A 84 NO. S/A LOADED PER CYCLE* 42 NO. kg FISSILE MAT'L LOADED PER CYCLE* 456 NO. kg HEAVY METAL LOADED PER CYCLE* 2337 NO. CONTROL S/A 18 RT 2.5	NO. S/A 60 RT 2.5	FZ5 NO. FUEL S/A 66 NO. S/A LOADED PER CYCLE* 33 NO. kg FISSILE MAT'L LOADED PER CYCLE* 417 NO. kg HEAVY METAL LOADED PER CYCLE* 1836 NO. CONTROL S/A 0 RT 2.5	NO. S/A 72 RT 2.5	FZ6 NO. FUEL S/A 150 NO. S/A LOADED PER CYCLE* 75 NO. kg FISSILE MAT'L LOADED PER CYCLE* 948 NO. kg HEAVY METAL LOADED PER CYCLE* 4173 NO. CONTROL S/A 12 RT 2.5	NO. S/A 288 RT 6.25	NO. S/A 108 RT
LAB4		LAB5		LAB6		
LAR4		LAR5		LAR6		
BGP				BGP		

Figure D-2. (Continued.)

TABLE D-I

FUEL INVENTORY AT THE BEGINNING OF EQUILIBRIUM CYCLE
(kg)

Isotope	Core Region											
	FZ1	FZ2	FZ3	FZ4	FZ5	FZ6	AB1*	AB2	AB3	AB4	AB5	AB6
^{235}U	0.44	2.64	2.67	6.18	4.61	10.7	0.39	2.37	2.38	5.53	4.36	10.0
^{236}U	Isotope Not Evaluated											
^{238}U	244.4	1466	1467	3421	2540	5796	203.1	1218	1218	2842	2233	5083
^{239}Pu	53.7	321.6	323.1	750.9	679.0	1558	1.0	6.2	5.5	14.3	10.1	18.4
^{240}Pu	17.9	107.3	107.2	250.3	227.9	512.0	0.02	0.10	0.09	0.24	0.15	0.22
^{241}Pu	7.5	45.2	45.5	105.7	96.9	223.7	--	0.002	0.002	0.005	0.003	0.003
^{242}Pu	2.1	12.8	12.8	29.8	27.2	61.2	--	--	--	--	--	--
Fission Products	7.8	47.2	43.2	108.5	93.3	181.1	0.009	0.6	0.5	1.4	0.9	1.7

*Note - Number shown is for combined upper and lower axial blankets (model had midplane symmetry).

TABLE D-II

FUEL INVENTORY AT THE BEGINNING OF EQUILIBRIUM CYCLE
(kg)

Isotope	Core Region						
	IB1	IB2	IB3	IB4	IB5	IB6	RB
^{235}U	0.22	2.70	6.81	9.57	13.7	16.7	64.0
^{236}U	Isotope Not Evaluated						
^{238}U	124.4	1495	3744	5243	7496	9008	35900
^{239}Pu	1.63	19.8	46.2	63.3	88.1	95.5	512.8
^{240}Pu	0.08	0.98	2.11	2.81	3.77	3.64	16.7
^{241}Pu	0.004	0.05	0.11	0.14	0.18	0.15	0.7
^{242}Pu	--	0.001	0.002	0.003	0.003	0.003	0.01
Fission Products	0.44	5.1	11.2	15.0	21.0	22.0	0.79

TABLE D-III

FUEL INVENTORY AT THE MIDDLE OF EQUILIBRIUM CYCLE
(kg)

Isotope	Core Region											
	FZ1	FZ2	FZ3	FZ4	FZ5	FZ6	AB1*	AB2	AB3	AB4	AB5	AB6
^{235}U	0.37	2.25	2.29	5.30	4.00	9.58	0.37	2.24	2.26	5.24	4.18	9.67
^{236}U	Isotope Not Evaluated											
^{238}U	239.2	1434	1438	3350	2492	5707	202.0	1211	1211	2826	2223	5064
^{239}Pu	50.7	303.1	306.1	709.7	636.0	1474	2.0	12.5	11.5	28.4	19.4	35.8
^{240}Pu	18.8	112.3	112.4	261.8	237.2	527.4	0.043	0.286	0.242	0.616	0.360	0.526
^{241}Pu	6.6	39.6	40.2	92.5	85.8	201.2	0.001	0.009	0.007	0.018	0.009	0.011
^{242}Pu	2.2	13.4	13.4	31.3	28.5	63.4	--	--	--	--	--	--
Fission Products	15.7	96.8	38.7	218.3	183.2	355.1	0.2	1.4	1.2	3.1	1.9	3.5

*Note - Number shown is for combined upper and lower axial blankets (model had midplane symmetry).

TABLE D-IV

FUEL INVENTORY AT THE MIDDLE OF EQUILIBRIUM CYCLE
(kg)

Isotope	Core Region						
	IB1	IB2	IB3	IB4	IB5	IB6	RB
^{235}U	0.20	2.39	6.07	8.50	12.1	14.8	0.62
^{236}U	Isotope Not Evaluated						
^{238}U	122.7	1474	3693	5171	7390	8886	35730
^{239}Pu	2.88	35.1	83.5	116.8	166.8	187.2	636.5
^{240}Pu	0.16	2.00	4.43	6.10	8.64	8.82	26.1
^{241}Pu	0.01	0.14	0.29	0.40	0.55	0.51	1.34
^{242}Pu	--	0.004	0.008	0.011	0.016	0.013	0.03
Fission Products	0.86	10.0	22.2	30.7	44.8	49.1	115.3

TABLE D-V

FUEL INVENTORY AT THE END OF EQUILIBRIUM CYCLE
(kg)

Isotope	Core Region											
	FZ1	FZ2	FZ3	FZ4	FZ5	FZ6	AB1*	AB2	AB3	AB4	AB5	AB6
^{235}U	0.32	1.93	1.98	4.56	3.48	8.59	0.36	2.12	2.15	4.98	4.00	9.33
^{236}U	Isotope Not Evaluated											
^{238}U	234.2	1404	1409	3283	2446	5623	200.9	1204	1205	2810	2212	5045
^{239}Pu	48.1	287.1	291.3	673.9	598.1	1400	3.0	18.5	16.9	41.8	28.6	52.5
^{240}Pu	19.6	116.5	116.6	271.3	244.7	540.2	0.08	0.55	0.47	1.19	0.70	1.03
^{241}Pu	5.9	35.2	36.0	82.9	76.9	182.7	0.003	0.021	0.017	0.044	0.023	0.026
^{242}Pu	2.3	13.9	13.9	32.4	29.4	65.2	--	--	--	--	--	--
Fission Products	23.2	141.9	130.4	318.9	266.3	513.9	0.4	2.4	2.0	5.3	3.3	5.8

*Note - Number shown is for combined upper and lower axial blankets (model had midplane symmetry).

TABLE D-VI

FUEL INVENTORY AT THE END OF EQUILIBRIUM CYCLE
(kg)

Isotope	Core Region						
	IB1	IB2	IB3	IB4	IB5	IB6	RB
^{235}U	0.18	2.11	5.38	7.54	10.8	13.3	59.4
^{236}U	Isotope Not Evaluated						
^{238}U	120.8	1452	3640	5097	7286	8772	35580
^{239}Pu	4.04	49.2	117.7	164.2	232.6	260.9	743.4
^{240}Pu	0.29	3.61	8.11	11.19	15.73	16.13	36.0
^{241}Pu	0.02	0.31	0.65	0.88	1.23	1.15	2.14
^{242}Pu	--	0.01	0.02	0.03	0.04	0.04	0.056
Fission Products	1.47	17.2	38.5	53.1	76.6	83.3	153.6

TABLE D-VII

MIDWALL AXIAL TEMPERATURE PROFILE FOR DESIGN LIMITING FUEL DUCT
(Nominal, EOL)

Distance Above BOF (ft)	Duct Midwall Temperature (°F)	Duct Midwall Temperature (°K)
0	597.93	587
3.0769E-01	605.92	592
6.1538E-01	616.59	598
9.2308E-01	629.59	605
1.2308E+00	644.49	613
1.5385E+00	660.78	622
1.8462E+00	677.94	632
2.1538E+00	695.38	641
2.4615E+00	712.53	651
2.7692E+00	728.83	660
3.0769E+00	743.72	669
3.3846E+00	756.72	676
3.6923E+00	767.40	681
4.0000E+00	775.39	686

TABLE D-VIII

MIDWALL AXIAL TEMPERATURE PROFILE FOR DESIGN LIMITING FUEL DUCT
(2σ, EOL)

Distance Above BOF (ft)	Duct Midwall Temperature (°F)	Duct Midwall Temperature (°K)
0	614.75	597
3.0769E-01	624.99	602
6.1538E-01	638.66	610
9.2308E-01	655.32	619
1.2308E+00	674.41	630
1.5385E+00	695.28	641
1.8462E+00	717.27	654
2.1538E+00	739.62	666
2.4615E+00	761.59	678
2.7692E+00	782.48	690
3.0769E+00	801.55	701
3.3846E+00	818.21	710
3.6923E+00	831.90	717
4.0000E+00	842.13	723

TABLE D-IX

NOMINAL CLADDING TEMPERATURE AXIAL PROFILES FOR DESIGN LIMITING FUEL PIN

Distance from Bottom of Fuel (in.)	Beginning of Life: Clad OD Temperature (°F)	Beginning of Life: Clad OD Temperature (°K)	Beginning of Life: Clad ID Temperature (°F)	Beginning of Life: Clad ID Temperature (°K)	End of Life: Clad OD Temperature (°F)	End of Life: Clad OD Temperature (°K)	End of Life: Clad ID Temperature (°F)	End of Life: Clad ID Temperature (°K)
1.714	618	599	653	618	611	595	633	607
5.143	648	615	694	641	630	605	659	621
8.571	685	636	739	666	653	618	688	637
12.000	728	660	789	694	680	633	719	655
15.429	775	686	842	723	710	650	753	674
18.857	826	714	895	752	741	667	786	692
22.286	878	743	948	782	774	685	820	711
25.714	930	772	999	810	807	704	852	729
29.143	981	800	1047	837	839	721	883	746
32.571	1030	827	1090	861	869	738	910	761
36.000	1074	852	1128	882	897	754	934	774
39.429	1112	873	1159	899	921	767	952	784
42.857	1144	891	1181	911	941	778	966	792
46.286	1168	904	1196	920	956	786	975	797
48.000	1177	909	1200	922	962	790	977	798

TABLE D-X

2σ CLADDING TEMPERATURE AXIAL PROFILES FOR DESIGN LIMITING FUEL PIN

Distance from Bottom of Fuel (in.)	Beginning of Life: Clad OD Temperature (°F)	Beginning of Life: Clad OD Temperature (°K)	Beginning of Life: Clad ID Temperature (°F)	Beginning of Life: Clad ID Temperature (°K)	End of Life: Clad OD Temperature (°F)	End of Life: Clad OD Temperature (°K)	End of Life: Clad ID Temperature (°F)	End of Life: Clad ID Temperature (°K)
1.714	663	624	709	649	645	614	675	630
5.143	709	649	770	683	674	630	713	651
8.571	763	679	834	719	708	649	754	674
12.000	823	712	903	757	745	669	798	699
15.429	888	749	974	796	787	692	844	724
18.857	955	786	1043	835	829	716	888	749
22.286	1024	824	1111	872	872	740	931	772
25.714	1091	861	1175	908	914	763	972	795
29.143	1155	897	1234	941	954	785	1010	816
32.571	1214	930	1287	970	991	806	1043	835
36.000	1266	959	1331	995	1024	824	1070	850
39.429	1310	983	1364	1013	1051	839	1090	861
42.857	1343	1001	1387	1026	1073	851	1104	869
46.286	1367	1015	1399	1032	1087	859	1111	872
48.000	1374	1019	1401	1034	1092	862	1112	873

TABLE D-XI

MIDWALL AXIAL TEMPERATURE PROFILE FOR DESIGN LIMITING INTERNAL FERTILE DUCT (Nominal, EOL)

Distance Above BOF (ft)	Duct Midwall Temperature (°F)	Duct Midwall Temperature (°K)
0	597.64	587.24
3.0769E-01	605.77	591.76
6.1538E-01	616.52	597.73
9.2308E-01	629.56	604.98
1.2308E+00	644.44	613.24
1.5385E+00	660.69	622.27
1.8462E+00	677.78	631.77
2.1538E+00	695.15	641.42
2.4615E+00	712.24	650.91
2.7692E+00	728.48	659.93
3.0769E+00	743.37	668.21
3.3846E+00	756.40	675.44
3.6923E+00	767.15	681.42
4.0000E+00	775.29	685.94

TABLE D-XII

MIDWALL AXIAL TEMPERATURE PROFILE FOR DESIGN LIMITING INTERNAL FERTILE DUCT (2σ, EOL)

Distance Above BOF (ft)	Duct Midwall Temperature (°F)	Duct Midwall Temperature (°K)
0	614.38	596.54
3.0769E-01	624.80	602.33
6.1538E-01	638.57	609.98
9.2308E-01	655.28	619.27
1.2308E+00	674.35	629.86
1.5385E+00	695.17	641.43
1.8462E+00	717.07	653.59
2.1538E+00	739.32	665.96
2.4615E+00	761.22	678.12
2.7692E+00	782.03	689.68
3.0769E+00	801.11	700.28
3.3846E+00	817.80	709.56
3.6923E+00	831.58	717.21
4.0000E+00	842.01	723.01

TABLE D-XIII

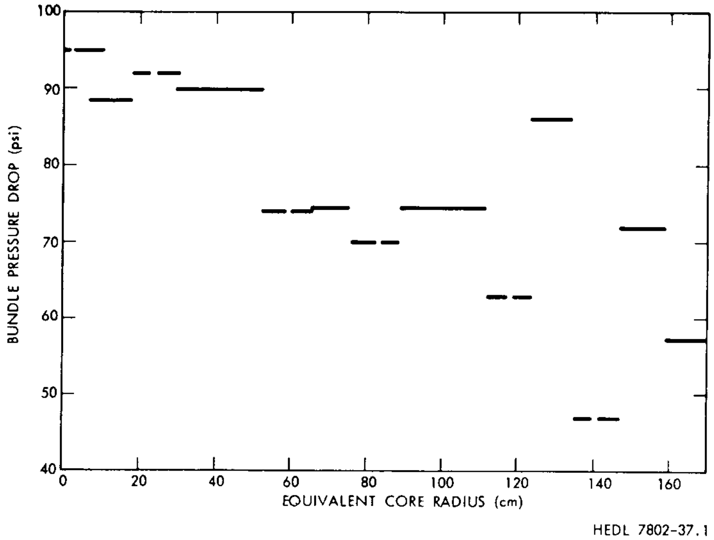
NOMINAL CLADDING TEMPERATURE AXIAL PROFILES FOR DESIGN LIMITING INTERNAL FERTILE PIN

Distance from Bottom of Fuel	Beginning of Life				End of Life			
	Clad OD Temperature		Clad ID Temperature		Clad OD Temperature		Clad ID Temperature	
(in.)	(°F)	(°K)	(°F)	(°K)	(°F)	(°K)	(°F)	(°K)
1.714	608	593	613	596	618	599	643	612
5.143	612	595	618	599	636	609	668	626
8.571	616	597	623	601	657	620	696	642
12.000	620	600	629	605	681	634	726	659
15.429	625	602	635	608	707	648	757	676
18.857	631	606	641	611	736	664	787	692
22.286	636	609	647	615	765	680	817	709
25.714	642	612	652	617	794	696	846	725
29.143	647	615	657	620	822	712	872	740
32.571	652	617	662	623	849	727	895	752
36.000	657	620	665	625	873	740	915	764
39.429	661	622	668	626	894	752	930	772
42.857	664	624	670	627	911	761	940	777
46.286	666	625	671	628	924	769	946	781
48.000	667	626	671	628	929	771	947	781

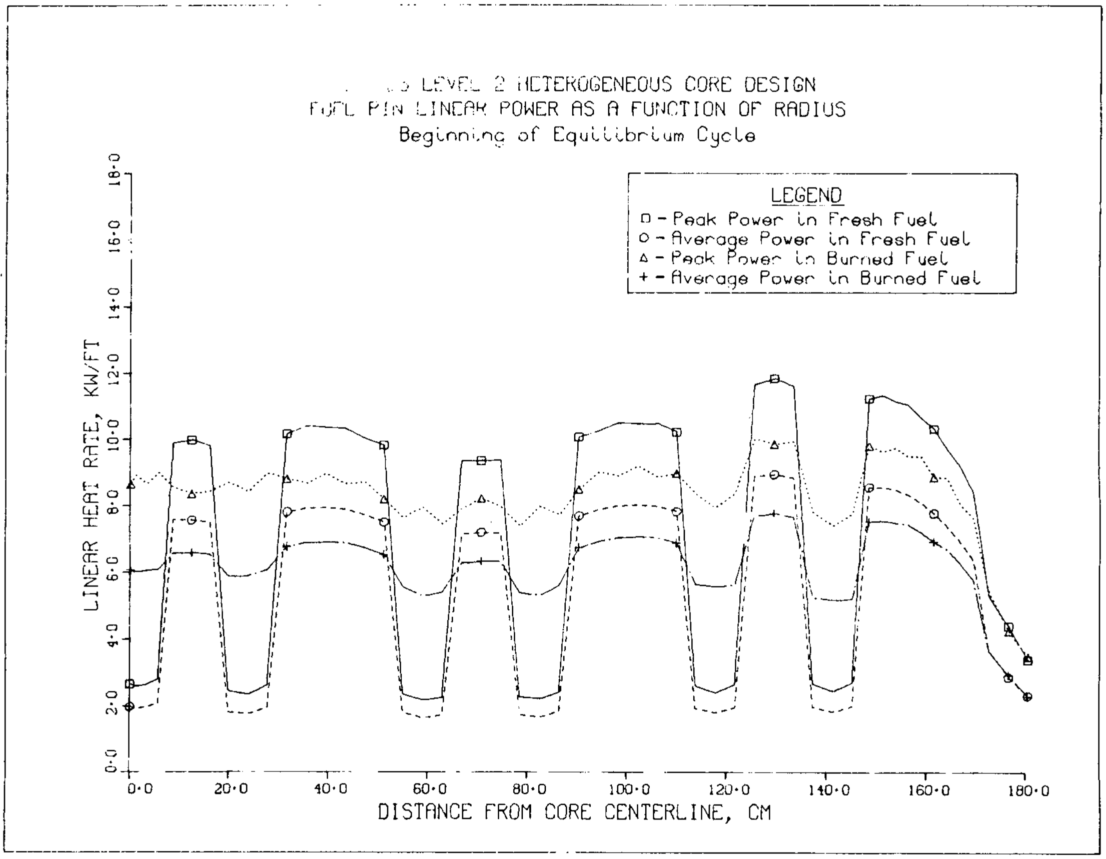
TABLE D-XIV

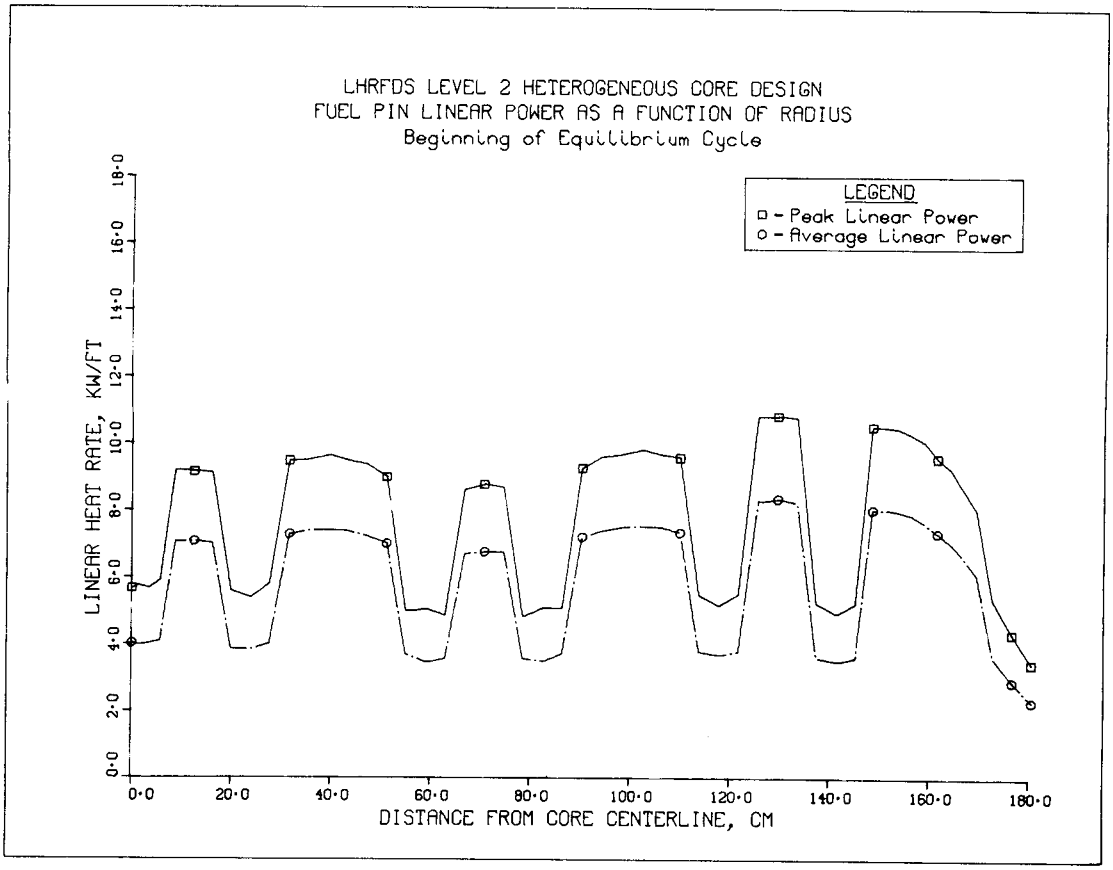
2σ CLADDING TEMPERATURE AXIAL PROFILES FOR DESIGN LIMITING INTERNAL FERTILE PIN

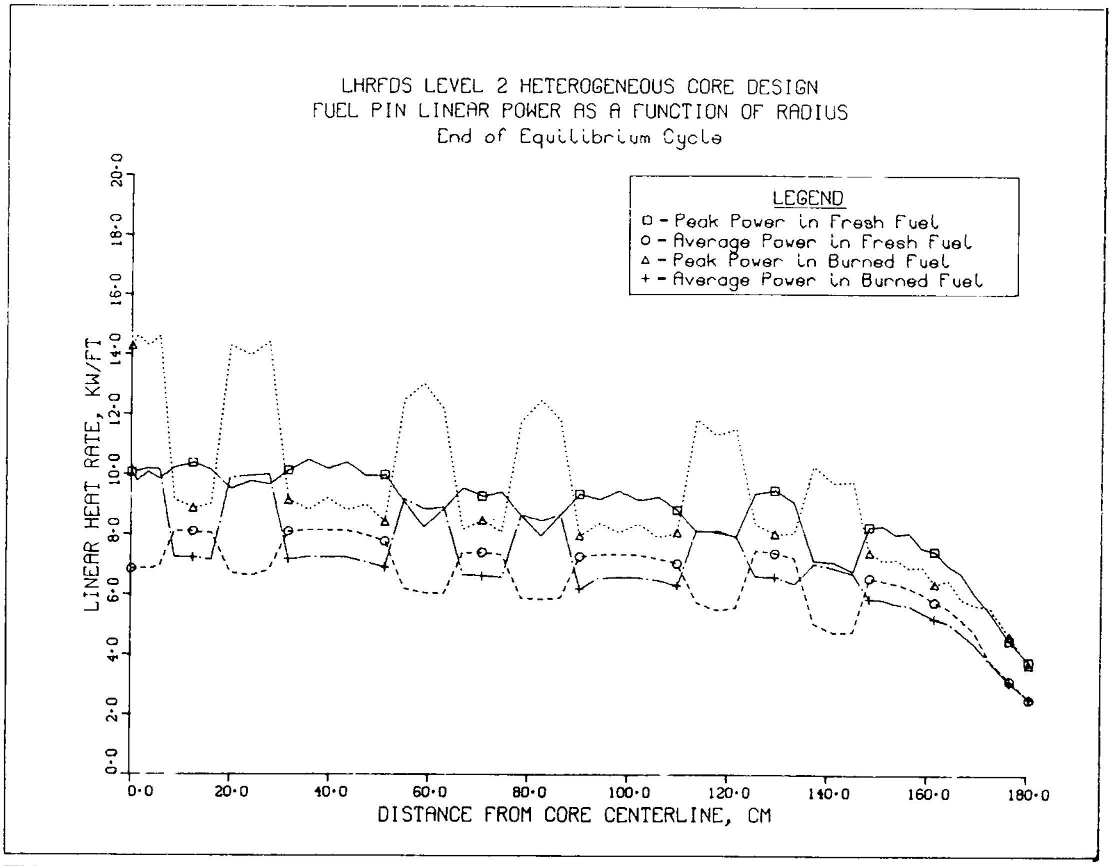
Distance from Bottom of Fuel	Beginning of Life				End of Life			
	Clad OD Temperature		Clad ID Temperature		Clad OD Temperature		Clad ID Temperature	
(in.)	(°F)	(°K)	(°F)	(°K)	(°F)	(°K)	(°F)	(°K)
1.714	628	604	635	608	655	619	688	637
5.143	634	607	642	612	682	634	727	659
8.571	640	611	650	616	714	652	766	681
12.000	646	614	658	621	749	671	808	704
15.429	653	618	667	626	787	692	851	728
18.857	661	622	675	630	825	714	891	750
22.286	668	626	682	634	863	735	931	772
25.714	675	630	689	638	901	756	967	792
29.143	682	634	696	642	936	775	999	810
32.571	688	637	701	645	968	793	1026	825
36.000	693	640	705	647	996	809	1048	837
39.429	697	642	708	649	1019	821	1063	846
42.857	701	645	709	649	1036	831	1072	851
46.286	703	646	709	649	1046	836	1073	851
48.000	703	646	709	649	1050	839	1072	851

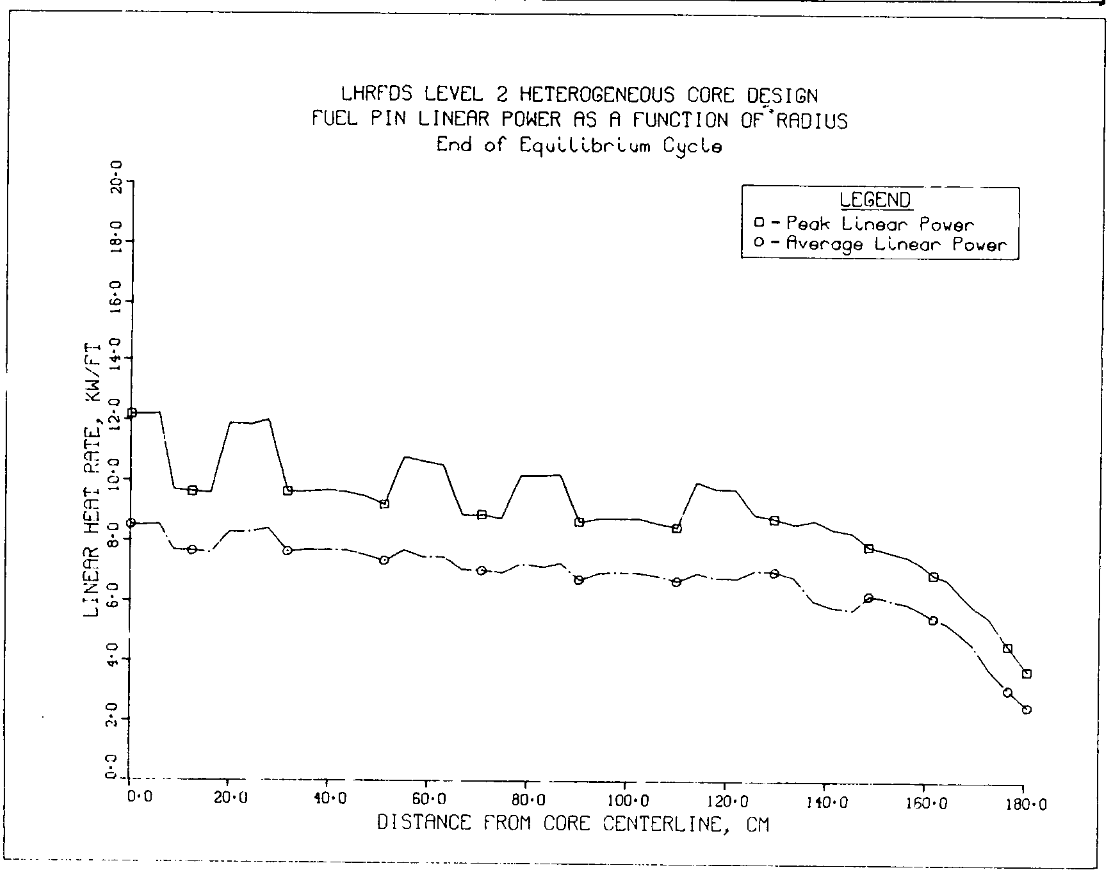


BUNDLE PRESSURE DROP


LEVEL 2 HETEROGENEOUS CORE DESIGN
POWER AS A FUNCTION OF RADIUS
Beginning of Equilibrium Cycle
LEGEND
□ - Peak Power in Fresh Fuel
○ - Average Power in Fresh Fuel
△ - Peak Power in Burned Fuel
+ - Average Power in Burned Fuel
LINEAR HEAT RATE, KW/FT
0·0
2·0
4·0
6·0
8·0
10·0
12·0
14·0
16·0
18·0
0·0
20·0
40·0
60·0
80·0
100·0
120·0
140·0
160·0
180·0
DISTANCE FROM CORE CENTERLINE, CM


LHRFDS LEVEL 2 HETEROGENEOUS CORE DESIGN
FUEL PIN LINEAR POWER AS A FUNCTION OF RADIUS
Beginning of Equilibrium Cycle
LEGEND
□ - Peak Linear Power
○ - Average Linear Power
LINEAR HEAT RATE, KW/FT
0·0
2·0
4·0
6·0
8·0
10·0
12·0
14·0
16·0
18·0
0·0
20·0
40·0
60·0
80·0
100·0
120·0
140·0
160·0
180·0
DISTANCE FROM CORE CENTERLINE, CM


LHRFDS LEVEL 2 HETEROGENEOUS CORE DESIGN
FUEL PIN LINEAR POWER AS A FUNCTION OF RADIUS
End of Equilibrium Cycle
LEGEND
□ - Peak Power in Fresh Fuel
○ - Average Power in Fresh Fuel
△ - Peak Power in Burned Fuel
+ - Average Power in Burned Fuel
LINEAR HEAT RATE, KW/FT
DISTANCE FROM CORE CENTERLINE, CM


LHRFDS LEVEL 2 HETEROGENEOUS CORE DESIGN
FUEL PIN LINEAR POWER AS A FUNCTION OF RADIUS
End of Equilibrium Cycle
LEGEND
□ - Peak Linear Power
○ - Average Linear Power
LINEAR HEAT RATE, KW/FT
DISTANCE FROM CORE CENTERLINE, CM

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