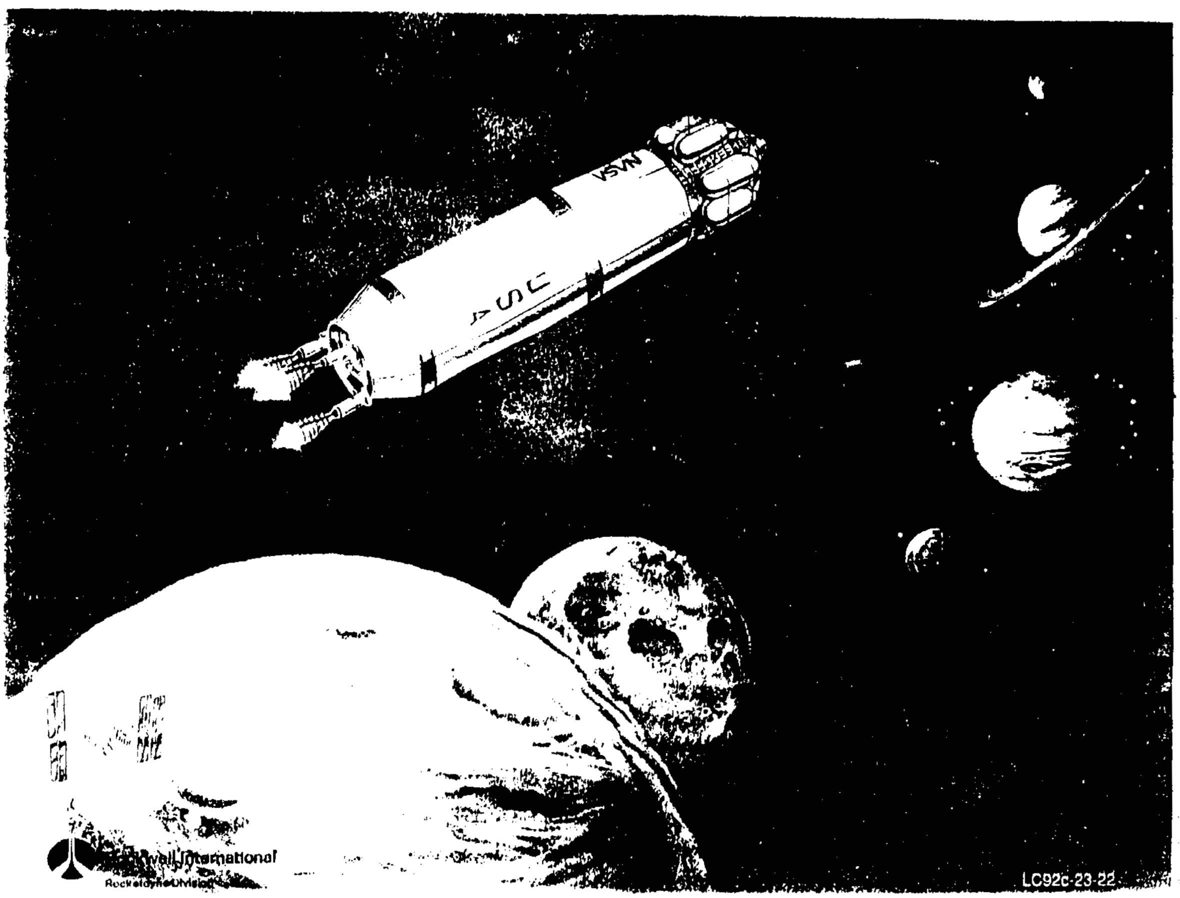
N93-26913

Rover/NERVA-Derived Near-Term Nuclear Propulsion

FY92 Final Review

at

NASA-LeRC
October 22, 1992



Agenda

- Introduction
- Reactor Concept Development
- Engine Conceptual Design
- Key Technology and Streamline Development Plan Assessment

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Introduction

FY92 accomplishments centered on conceptual design and analyses for 25K, 50K, and 75K engines, with emphasis on the 50K engine, to NASA requirements.

During the first period of performance flow and energy balances were prepared for each engine size with single and dual turbopumps. Plan, elevation, and isometric drawings were prepared for each of these configurations, and thrusts-to-weight were estimated. A review of fuel technology and key data from the Rover/NERVA program, established a baseline for proven reactor performance and areas of enhancement to meet near term goals. Studies were performed of the criticality and temperature profiles for probable fuel and moderator loadings for the three engine sizes, with a more detailed analysis of the 50K size.

During the second period of performance, analyses of the 50K engine continued. A chamber/nozzle contour was selected and heat transfer and fatigue analyses were performed for likely materials of construction. Reactor analyses were performed to determine component radiation heating rates, reactor radiation fields, water immersion poisoning requirements, temperature limits for restartability, and a tie tube thermal analysis. In addition reactor safety and reliability were assessed.

Finally, a brief assessment of key enabling technologies was made, with a view toward identifying development issues and identification of the critical path toward achieving engine qualification within 10 years. Our initial appraisal suggests that critical path for the program will be the design, construction, and acceptance testing of engine test facilities.

Requirements

- Rover/NERVA-derived technology
- "Near-Term" man-rated mission
- 4.5 hours qualification test at rated conditions to validate 1.5 hours at rated conditions for manned missions
- Restartable, at least 10 starts
- Launch envelope, 30 m (length) x 10 m (diameter)
- I_{sp} > 850 seconds
- Thrust

  A. Initially--25K, 50K, and 75K
  B. Continued effort--50K

- Thrust/Weight (with internal shield) ≥ 4

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Requirements

Requirements for the FY92 NASA-funded effort derive from the Statement of Work. The basic objective was the assessement of the near-term feasibility of Rover/NERVA-derived nuclear thermal rocket engine technology for meeting piloted missions to Mars. The basic requirements for the engine provided by NASA included size limits, target specific impulse, number of restarts, operating life, and thrust-to-weight lower limit. Initial analyses were to be performed for three engine thrust sizes: 25K, 50K, and 75K. Final concept development was to be performed for the 50K thrust size engine.

Additional Ground Rules

- Pewee fuel element, temperature, and ZrH moderator
  - Chamber temperature 2,550 K
  - Power density 1.18 MW/element
- Tie tubes with expander cycle
- Dual turbopumps/loss of both pumps
- Nozzle expansion ratio, 200/1
- Radiation leakage limits from NERVA
- System requirements of NASA N.P. 002

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Additional Ground Rules

Communication with NASA subsequent to issuance of the Statement of Work provided additional guidance: Pewee operating parameters for chamber temperature and power density, use of tie tubes with the expander cycle, incorporation of dual turbopumps with consideration of pump outages, a nozzle expansion area ratio (200), radiation limits from the NERVA design, and additional system requirements found in NASA N.P. 002, "Nuclear Thermal Rocket Engine Requirements."

NERVA-Derived 50K Engine Schematic

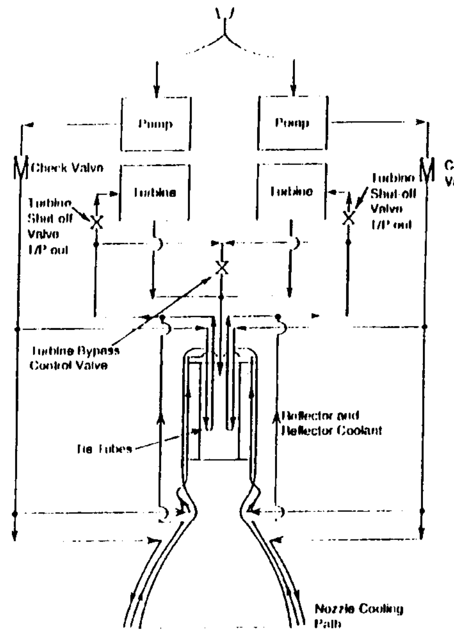
Chamber pressure = 784 psia

Chamber temperature = 2,550 K

Specific impulse = 870 seconds

Nozzle expansion ratio = 200:1

Nozzle bell = 110% length



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NERVA-Derived 50K Engine Schematic

The 50K engine features dual turbopumps supplying liquid hydrogen to the tie tubes, and the chamber and nozzle. Approximately 70% of the flow goes to cool the tie tubes and moderator; the heat pickup provides the energy for the turbines. The propellant flow used to cool the chamber and nozzle also cools the reflector and pressure vessel. The total flow is mixed together, flows through the fuel elements where the temperature is increased to 2,550 K, and is exhausted from the nozzle to produce thrust. The engine is sized and packaged to fit within given geometrical constraints; consequently, chamber pressure and bell nozzle length are selected to maximize specific impulse and thrust-to-weight.

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Key Features/Attributes

- Proven technology, low risk approach
    - Nozzle technology flying with Space Shuttle
    - Existing turbopump designs applicable
    - Rover/NERVA-derived reactor
    - Minimum development time/money
    - Supports 10-year qualification goal
- I_{sp} > 150 seconds better than NERVA-XE'
- MCNP permits fuel loading for flat profile
- Tie-tube support approach facilitates
    - Expander cycle turbine for improved I_{sp}
    - Incorporation of ZrH to minimize reactor size
- Optimized packaging and flow balancing
- Can accept evolutionary improvements

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Key Features/Attributes

The Rocketdyne-Westinghouse nuclear thermal rocket engine benefits from a combination of the technology proven in the Rover/NERVA program and modern rocket engine man-rated components. The goal of producing a qualified engine within 10-years can be achieved with minimal development, based on the current state of the art. Cooled chamber and nozzle technology from the SSME is directly applicable, and turbopumps from the J-2S, Rover, and SSME bracket current requirements. Studies were initiated to examine pump-out performance with boost pumps and multiple turbopumps; however, meaningful results were not achieved within the allocable funding limitations.

Easily achievable enhancements provide improvements in I_{sp} over the last NERVA engine tested, NRX-XE'. Incorporation of tie tubes and the expander cycle, increase of the expansion ratio from 10 to 200, regenerative cooling throughout, and increase of the chamber temperature to the Pewee conditions adds over 150 seconds of specific impulse. A further increase of chamber temperature to 2,700 K by use of composite elements would add another 30 seconds to bring the total to 900 seconds.

The preliminary configuration has the turbopumps at the side of the chamber to shorten the overall length of the engine assembly. Within that configuration flow and energy balances are optimized to minimize pressure which directly affects ducting wall thickness.

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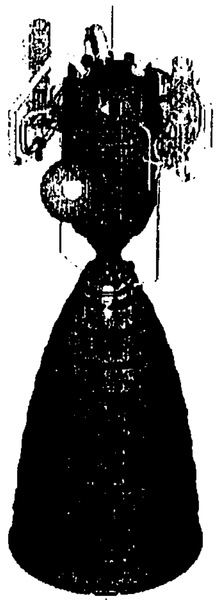
NERVA-Derived 50K Engine Isometric



Reactor exit temperature = 2,550 K

Dual turbopumps

Split-flow expander cycle

Nozzle expansion ratio = 200:1

Nozzle bell = 110% length

Specific impulse = 870 seconds

Thrust/weight = 5.3 (with shield)

Engine length = 7.6 m

Exit diameter = 2.4 m

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NERVA-Derived 50K Engine Isometric

A key feature of the engine includes compact packaging, with turbopumps mounted to the side of the reactor vessel to reduce the overall height and permit a higher expansion within the geometrical constraints. Another feature has the tubular nozzle attaching to the chamber at a low expansion ratio to save weight and to facilitate ground testing. An area for evolutionary change in this design would be the substitution of uncooled composite ceramic materials for the tubular nozzle for a potential weight saving and some increase in specific impulse.

Technology Assessment Results

- Technology available for most issues

  Rover/NERVA, SSME, Rocketdyne state of the art, SP-100, terrestrial advanced reactors, state-of-the-art electronics and computers

- Unresolved system design issues

  Loss of turbopumps, lifetime, intact reentry--water subcriticality (or total dispersal), decay heat removal, engine-out cooling during operations, fuel midband corrosion

- Critical path is engine test facility

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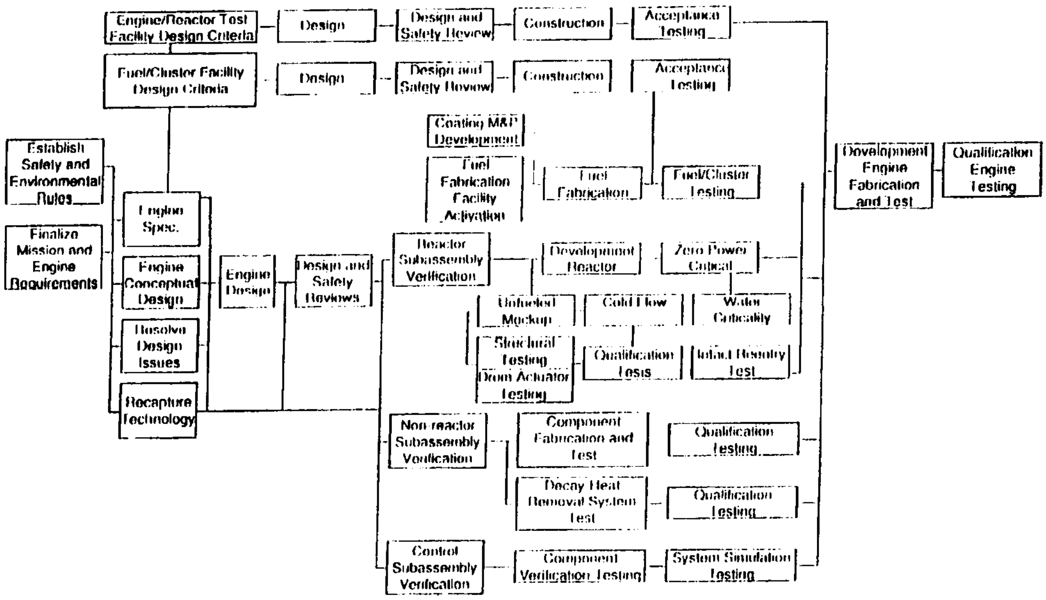
Technology Assessment Results

The assessment of key technologies led to the conclusions that (1) existing technology in reactors and engine systems is applicable to most design areas, (2) there are issues requiring attention early in the program to assure satisfactory resolution, and (3) the assured early availability of an engine/reactor test facility is critical to meet, successfully meet the 10-year engine qualification goal.

NTR Streamline Development Logic



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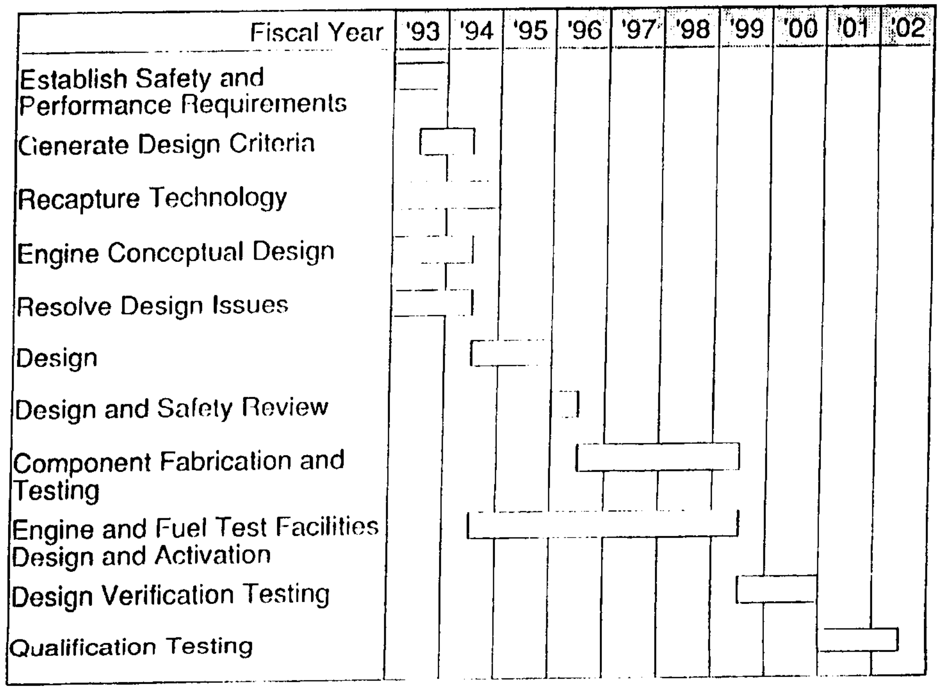
NTR Streamline Development Logic

A development logic diagram can include many layers of detail and be organized in many different ways. This high-level diagram shows many necessary tasks in setting requirements, recapturing technology, resolution of key design issues, facility design, construction, and activation, and testing of components and systems. The most important message is that the program must start with well-defined requirements and design criteria, and that the availability of key test facilities will drive the rate of achievement of the 10-year goals. Near-term activities of conceptual design, technology recovery, and resolution of design issues will provide a sound basis for proceeding quickly as substantial funding becomes available.

NTR Streamline Development Plan Summary



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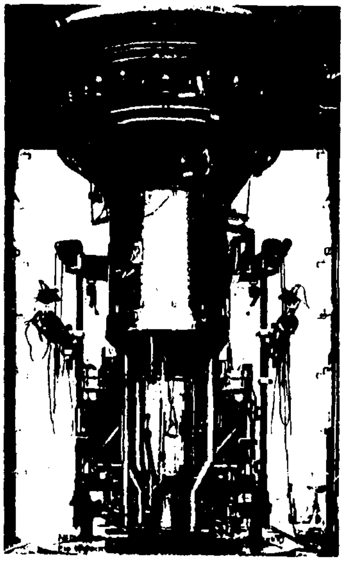
NTR Streamline Development Plan Summary

The time-phasing of key groups of activities from the development logic diagram shows that several tasks should be emphasized at the start: setting requirements, technology recapture, and establishing design criteria. Test facility design, construction and activation must also begin promptly to assure that the 10-year schedule can be met.

REACTOR CONCEPT DEVELOPMENT



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- NERVA Derivative Reactor Concept Design
- NTR Nuclear Parameter Study
- Analysis of Reactor for 50K/lb$_f$ Engine
- Assessment of Fuel Technology
- Assessment of Nuclear Safety Issues
- Summary and Conclusions

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Development of a nuclear thermal rocket design concept for Fast Track studies is based on the NERVA/Rover technology database. Design analyses to provide NTR designs to meet program requirements are developed with current design methodologies benchmarked to NERVA/Rover technology. The NERVA derivative reactor concept design is based on NERVA R-1 reactor design with design features upgraded to include the demonstrated capabilities of the NERVA/Rover program. A historical summary of the completed tests of the NERVA/Rover program and the NTR performance demonstrated by test results are summarized in the following pages.

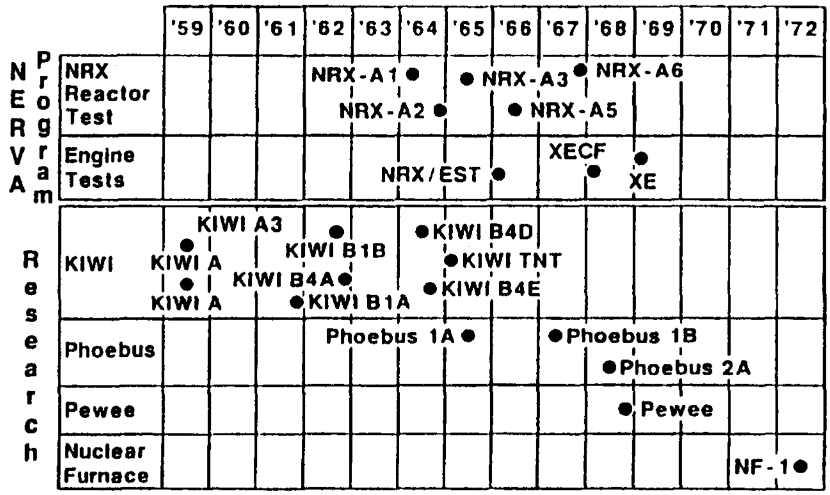
Based on a set of NASA directives, parametric analyses of the size and performance characteristics of NTR reactors which provide performance consistent with 25K, 50K, and 75K lb$_f$ engines was completed. Later discussions show the results of more detailed studies on the reactor design for the 50K lb$_f$ engine.

Based on a review of the NERVA/Rover technology database, a current assessment of the fuel technology and nuclear safety issues for the application of the NERVA derivative reactor in the NTR program is discussed.

In summary, the lessons learned during the conduct of the work tasks are discussed.

NERVA/Rover REACTOR SYSTEM TEST SEQUENCE

The fast track engine draws upon the existing 1.4 billion dollar technology base developed by Los Alamos National Laboratory and Westinghouse during the NERVA/Rover Nuclear Rocket Engine Program.



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The extent of the NERVA/Rover technology is demonstrated by the number of reactor and engine tests completed over the 1959-1972 time frame. The reactor tests completed in the KIWI/PHOEBUS/PEEWEE series demonstrated the wide range in reactor size and power capability provided by the technology. The NERVA test series culminating in the NRX-A6 and XE-Prime tests demonstrated lifetime and performance capabilities of the NERVA/Rover-based NTR's. The NERVA program successfully completed the preliminary design of the R-1 reactor design and the Fast Track reactor designs developed in the current work tasks are derived from the extensive technology database of the NERVA/Rover programs.

Demonstrated Technology

Rover/NERVA Test History

Reactor ID	Chamber Temp. (K)	Fuel Exit Temp. (K)	Space Equiv. ISP (sec)	Time at Full Power (min)	Fuel Type	MWt/ Thrust (kN)
KIWI-B4D	1890-2130	2222	780	1	UO2/Graphite	914/204
KIWI-B4E	1890-2100	2389	820	82.5	UC2/Graphite	914/204
NRX-A2	2090	>2200	775	3.4	UC2/Graphite	1100/245
NRX-A3	2244	>2400	820	18.3	UC2/Graphite	1100/245
PHOEBUS-1A	2366	2478	835	10.5	UC2/Graphite	1340/298
NRX-A4(NRX-EST)	2264-2290	>2400	820	28.6	UC2/Graphite	1100/245
NRX-A5	2280-2333	>2400	820	29.8	UC2/Graphite	1100/245
PHOEBUS-1B	2222-2290	2445	828	30	UC2/Graphite	1340/298
NRX-A6	2300-2405	2556	847	62.7	UC2/Graphite	1100/245
PEWEE-1	1835	2550 2750	845 890	43	UC2/Graphite UC2/Graphite	500/111
XE-PRIME	2278	>2400	820	7.8	UC2/Graphite	1100/245
NF-1	—	2450	830	109	Composite/ Carbide	
PHOEBUS-2A TESTED	2256	2308	805	12.5	UC2/Graphite	4100/913
DESIGN	2500	2550	840		UC2/Graphite	5000/1113

The demonstrated capabilities of NERVA/Rover based NTR's is summarized in the following table. The performance levels reached in each of the key tests completed as shown. As shown, the NERVA/Rover technology provides reactor performance capabilities similar to the requirements of the Fast Track program and later discussions show the capability of NERVA/Rover based design concepts to meet the Fast Track program needs.

NERVA Derivative Reactor Concept Design for 50K lbf Thrust Engine

- Layout drawing
- Solid models

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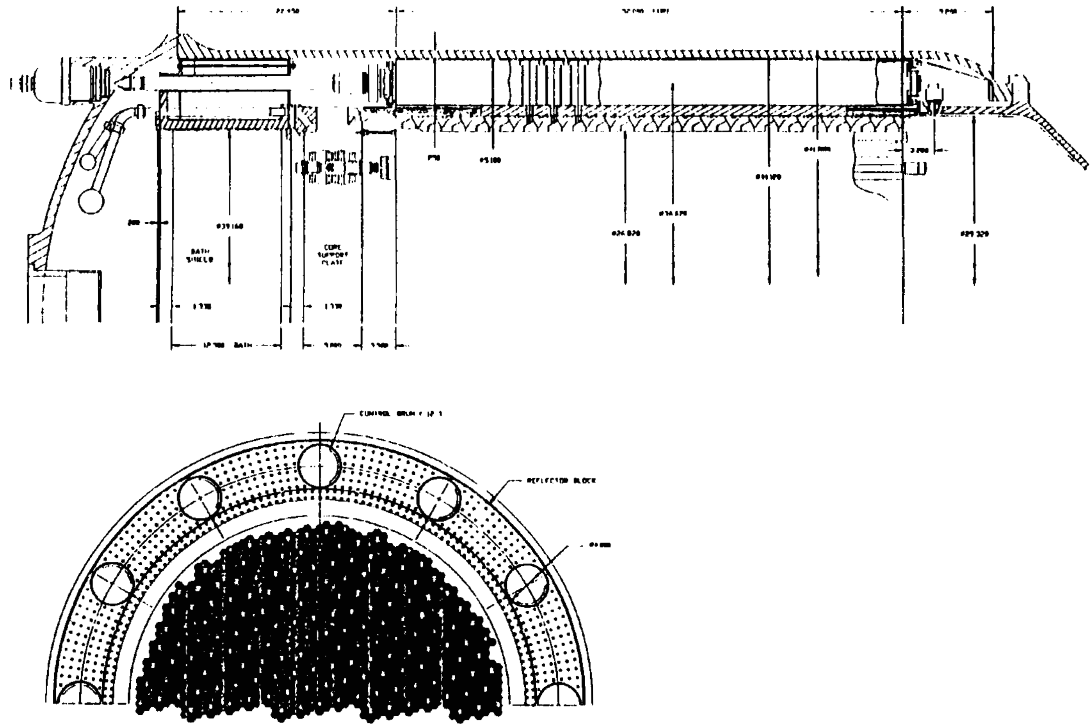
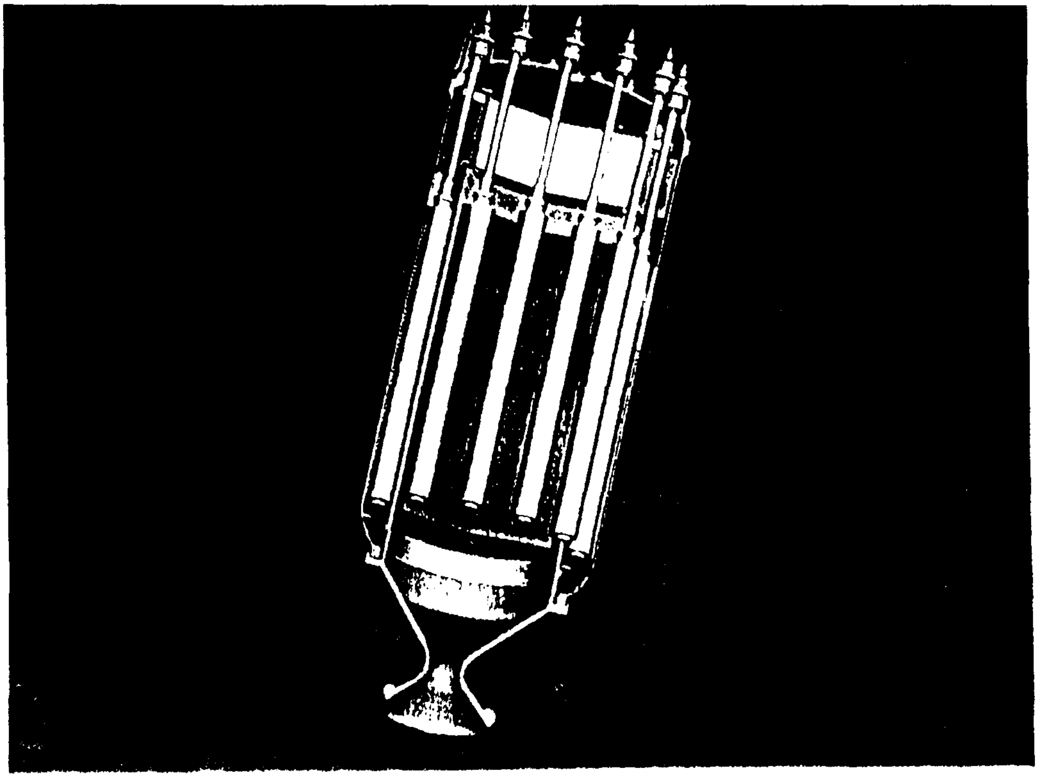
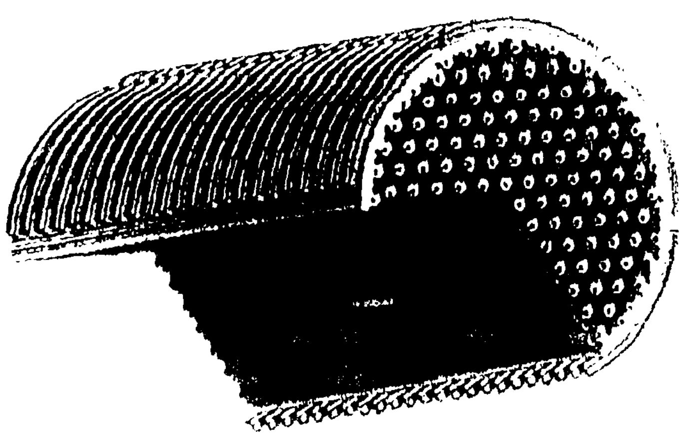


The 50K lb, engine reactor layout is based on the R-1 NERVA flight reactor design. The R-1 successfully completed an Air Force Preliminary Design Review before the termination of the NERVA project in 1972. The key dimensions of the reactor for the 50 k lbf engine are shown. These were established based on the required engine thrust (core size), and the neutronic requirements (reflector and shield).

C-2



Solid models of the major reactor components and assembly was generated using Pro Engineer. The viewgraphs show the assembly model of the reactor, where the major components, including the core, can be seen. Since the solid model is parametric in nature, it can be used for a series of reactor sizes of the same type. This makes trade studies easy to perform and weight estimates for these types of reactors can be established fast and accurately. The second major reason for utilizing the Pro-Engineer solid modelling approach is that it provides seamless interface to analytical tools. This integrated approach to design and modelling will be fully utilized in the development of the NERVA derivative reactor design.



NTR Nuclear Parameter Study

- Neutronics Model
- 25K lbf Engine Results
- 50K lbf Engine Results
- 75K lbf Engine Results
- Heterogeneity Evaluation

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Studies of the neutronics design of the NTR were based on three dimensional models derived from the NERVA design. The methodology selected for use in the parametric analyses was the MCNP Monte Carlo radiation transport method. Model parameters of the reactor system were derived from the R-Z model information of the NERVA R-1 reactor system. An automated model generation technique was used to define reactor system models for parametric analyses to size and predict performance characteristics for the various sizes of the NTR system. An R-Z annular ring model of the NTR core configuration was used in parametric analyses in a similar manner to the models in the NERVA database. Three dimensional model details were limited to the reflector control drums and used the geometric modelling capability of the MCNP method. The automated modelling technique and MCNP (Version 3B) were used to define the core and reflector sizes, fuel loading profiles, reactivity worths, and control drum worths and span for three NTR engine sizes; 25, 50, and 75 Klbf thrust levels. In addition, a limited study of the impact of heterogeneous versus homogeneous modelling of the prismatic fuel elements and tie-tubes within the NTR core was performed on a unit cell basis

Study Guidelines

- NERVA (Prismatic) Fuel Elements
  52" Long
  0.753" Hex
  19 Coolant Channels
  600 mg/cm^3 Maximum Fuel Loading

- ZrHx Moderated Tie-Tube
  SNRC (PeeWee) Maximum ZrHx
  2:1, 3:1, 6:1 Fuel Element to Tie-Tube Ratios

- Performance
  1.18 Mw/element
  2550K Chamber Temperature (Point Design)
  784 psia Chamber Pressure

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STUDY GUIDELINES/ASSUMPTIONS

- Reactor Sizes 25K, 50K and 75K lbf Thrust Engines
- Critical Drum Angle of 80°
- NERVA/R-1 Reactor Design Configuration
- R-Z Geometry with Explicit Control Drums
- Neutronics Calculations: MCNP-3B

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Technology Base

Design Features	Reactors						
	NERVA			PeeWee	25K	50K	75K
	Xe-Prime	A-6	R-1				
Tie-Tubes	No	No	Yes	Yes	Yes	Yes	Yes
Ratio, Fe/TT			6:1	3:1	3:1	6:1	6:1
ZrH Loading (Relative)			0.0	1.0	1.0	1.0	0.4
Power, Mwt					512	1024	1536
Core Diameter, In.			35.0		18.8	25.2	30.7
Power Density, MW/FE	0.67	0.67	0.75	1.18	1.18	1.18	1.18
Internal Shield	A1	A1	BATH	--	BATH	BATH	BATH
Fuel Type	Graphite	Graphite	Composite	Graphite	*	*	*

*Not Determined

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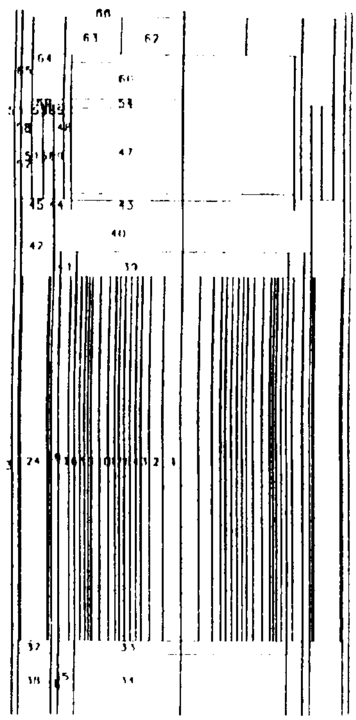
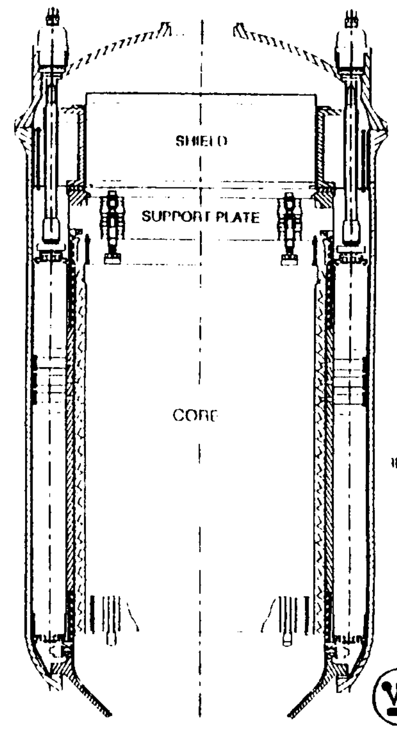
Neutronics Model





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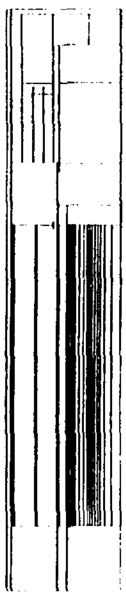
The neutronics model for the NTR system was derived from modelling information in the NERVA database and is shown as an elevation view to illustrate the modelling detail of MCNP models. The MCNP analyses used the ENDF/B V nuclear data library and were performed in the coupled neutron and photon solution mode to predict region power and required fuel loading to meet target objectives for key neutronics parameters. An actual NERVA system design configuration drawing is depicted to illustrate the modelling approach used.

MCNP Model for 25 Klbf Thrust Reactor



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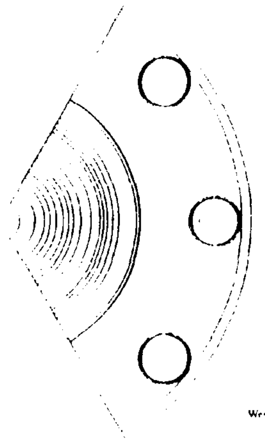
Key Parameters	25K
Effective Core Diameter (in.)	18.80
Fuel Support Ratio	3:1
Core Length (in.)	52.00
Reflector Thickness (in.)	8.00
No. of Drums	9
Peak Fuel Loading	600
ZrH Loading w.r.t. SNRE	1.0



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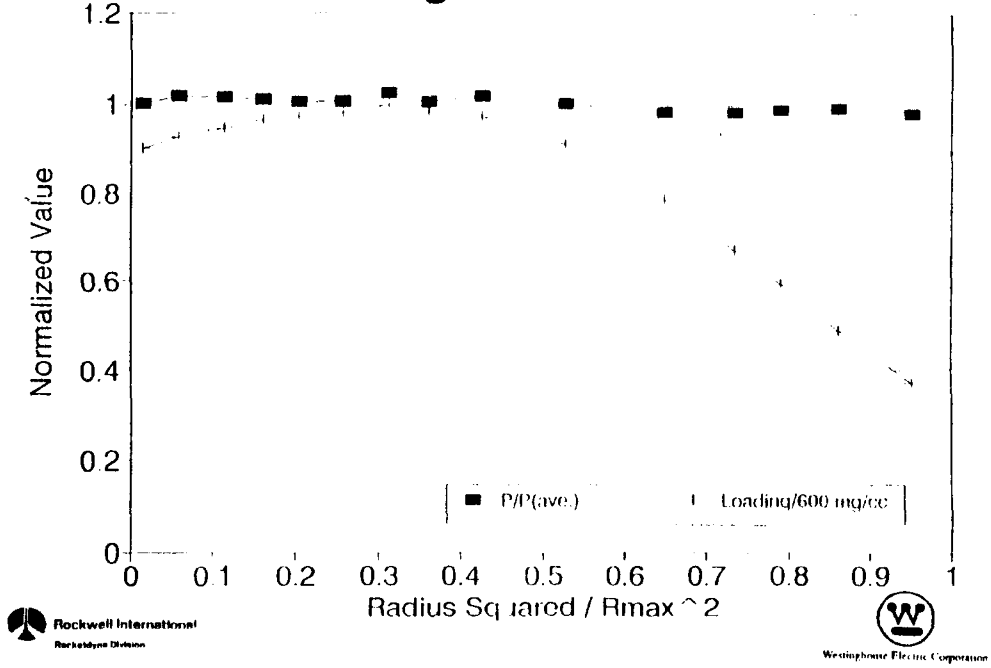
The MCNP model for the 25 Klbf NTR engine and the predicted key parameters are shown in the table on the right. The design bases selected for the small NTR engine size were derived from PEEWEE engine design information with a fuel-to-support tie tube ratio of 3:1, a 52 inch high active core, and 9 control drums of a fixed diameter located a the outer periphery of the Be reflector region. The peak fuel element uranium loading was limited to 600 milligrams/cm^3 and a maximum ZrH loading in the tie-tubes. An iterative process based on MCNP was used to size the reactor core and predict the fuel loading profile to meet the target objectives of an excess reactivity of 0.05 and a flat radial power distribution.

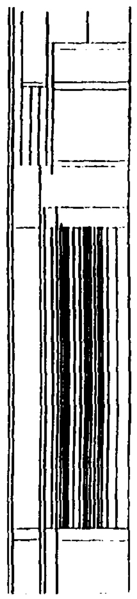
25 Klbf Thrust Engine
Core Fuel Loading & Power Distribution

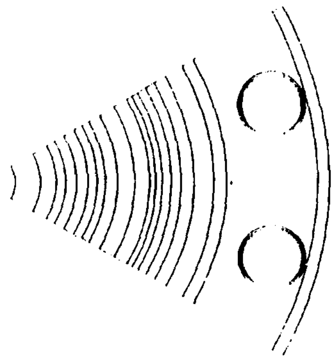


A normalized fuel loading profile predicted for the 25 Klbf NTR engine is shown as a function of normalized area parameter, R^2. The normalized radial power distribution as predicted in the final iteration of the analysis is shown to illustrate convergence to the target objective of a flat or uniform power profile. The MCNP tally method provides the cell or ring average value and more detailed tallying techniques would be required to predict the variation within each fuel annulus.

MCNP Model for 50 Llbf Thrust Reactor

Key Parameters	50K
Effective Core Diameter (in.)	25.18
Fuel:Support Ratio	6:1
Core Length (in.)	52.00
Reflector Thickness (in.)	5.10
No. of Drums	12
Peak Fuel Loading	600
ZrH Loading w.r.t. SNRE	1.0





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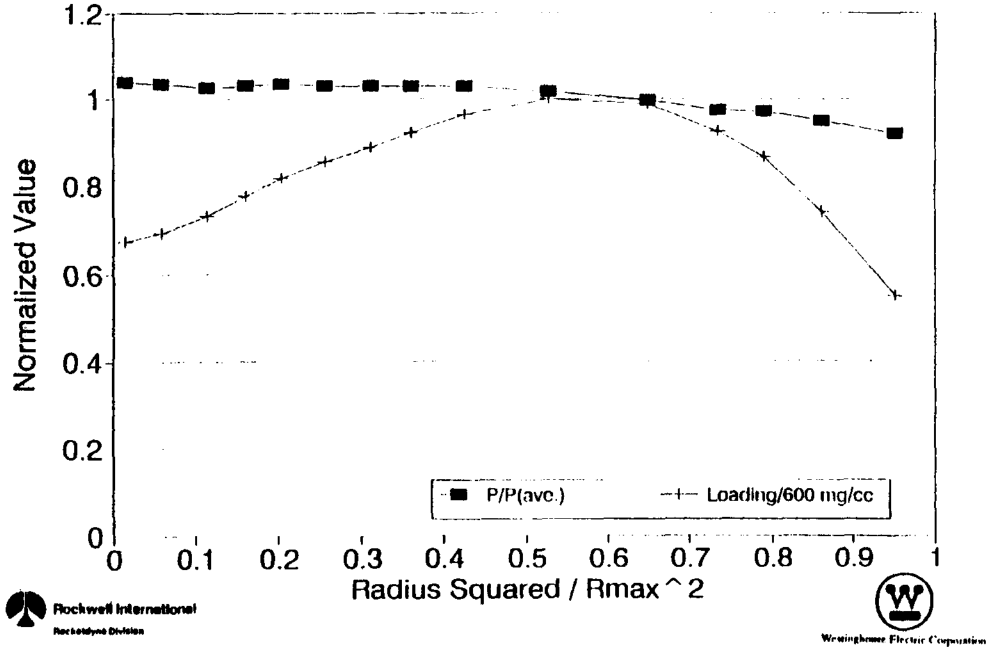
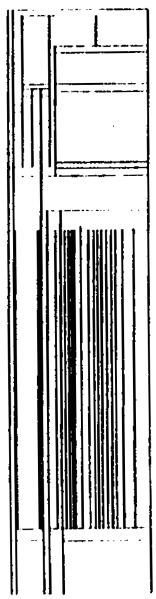
Predicted neutronics parameters for a 50 Klbf NTR engine are shown in the table. Key differences in the design bases selected for this size of engine were a fuel-to-support tie-tube ratio of 6:1 and reflector thickness and number of control drums. The effective core diameter required to meet target objectives is 25.18 inches.

50 Klbf Thrust Engine
Core Fuel Loading & Power Distribution

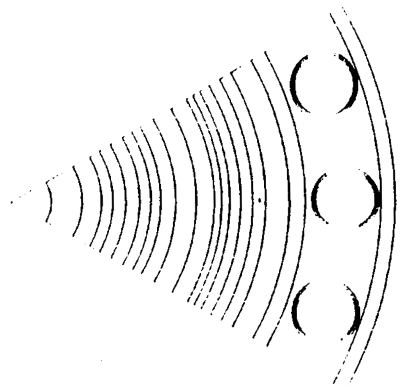


The normalized fuel loading profile predicted for the 50 Klbf NTR engine is shown as a function of normalized area parameter, R^2. As shown, the fuel loading profile differs from the 25 Klbf engine data due to the larger size and the change to a 6:1 fuel:support tie-tube ratio. The lower fuel loading required in the center of the core is related to the change in the moderation of the core and the increase in median fission energy and the effect of radial leakage.

MCNP Model for 75 Klbf Thrust Reactor



Key Parameters	75K
Effective Core Diameter (in.)	30.66
Fuel:Support Ratio	6:1
Core Length (in.)	52.00
Reflector Thickness (in.)	4.785
No. of Drums	18
Peak Fuel Loading	600
ZrH Loading w.r.t. SNRE	0.4



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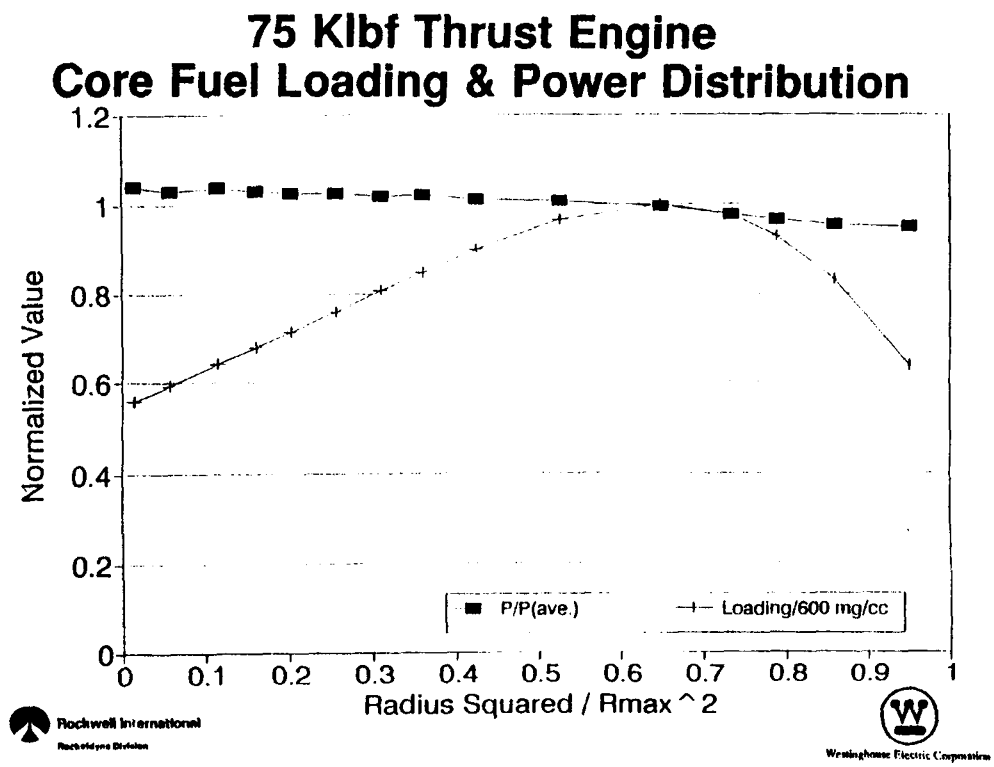
Predicted neutronics parameters for a 75 Klbf NTR engine are shown in the table on the right. The 75 Klbf engine size is similar to the NRX-A6 or R-1 size and the predicted parameters are comparable to the NERVA data. Key differences in the design bases selected for this size of engine were a decrease in the ZrH loading in the support tie-tubes of 0.4 with respect to the SNRE loading. The reflector thickness and number of control drums for the 75K engine are the R-1 dimensions. The effective core diameter required to meet target objectives is 30.66 inches which is similar to the NERVA design. .



The normalized fuel loading profile predicted for the 75 Klbf NTR engine is shown as a function of normalized area parameter, R^2. As shown, the fuel loading profile is similar to the 50K engine data and is comparable to NERVA loading profiles.

	Reactor Size, Klbf		
Parameter	25K	50K	75K
Thrust (lb)	25,000	50,000	75,000
Fuel: Support Ratio	3:1	6:1	6:1
Power (MW)	512	1024	1536
Flow (lb/sec)	28.2	56.3	84.5
Core Diameter (in)	18.8	25.2	30.7
ZrH Loading (Relative)	1.0	1.0	0.4
Reflector Thickness (in)	8.0	5.1	4.8
Pressure Vessel OD (in)	38.8	41.9	47.7
Reactor Mass w/o Shield (lb)	5180	6250	8040
Reactor Mass w/Shield (lb)	6590	8080	10480

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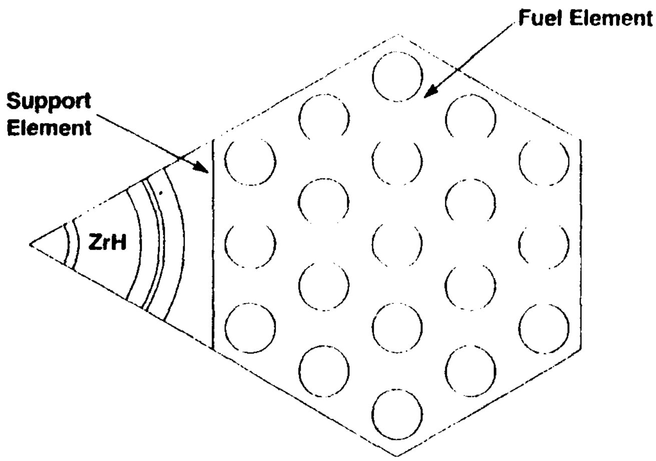
A summary of the results of the preliminary sizing of NTR engines in the 25Klbf-to-75Klbf size range is shown in the table. The design bases used in the parametric analyses are listed on the left. The prismatic fuel element length of 52 inches was adapted from NERVA and fuel performance limits defined based on the PEEWEE data. The predicted masses for the reactor system without and without shielding illustrate the effect of engine size on the engine performance characteristics and sizes. The use of ZrH in the 75K engine size differs from the NERVA design and the impact on a reduced reactor size and mass is shown. The shield masses included in the summary table are based on the same thickness of shield with the mass differences only showing the change in shield diameter.

Heterogeneity Analysis

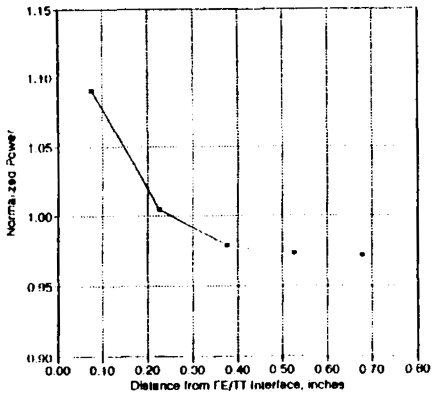


Unit Cell 6:1 Arrangement

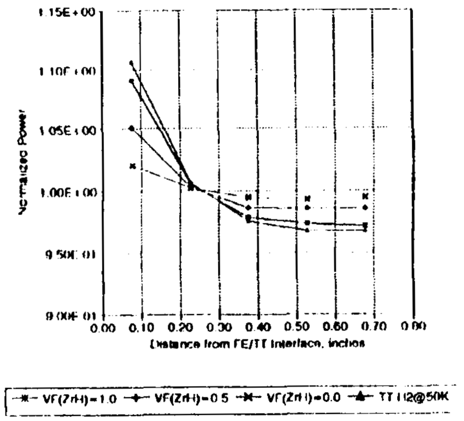
A limited study of the homogeneous region modelling technique for the prismatic fuel element core lattice with ZrH moderated support tie-tubes was carried out using the MCNP method. A unit cell model of a 6:1 fuel-to-support tie-tube configuration includes an annular model of the ZrH moderated tie-tube and the 19 coolant hole prismatic fuel element. A series of unit cell MCNP calculations were run to predict the effect of the ZrH tie-tube on local power distributions and to predict material or material interchange reactivity worths on a unit cell basis.

Effect of Modelling on Element Power Distribution

Effect of Modelling Approximations on Element Power Distribution



Effect of Material Changes on Element Power Distribution



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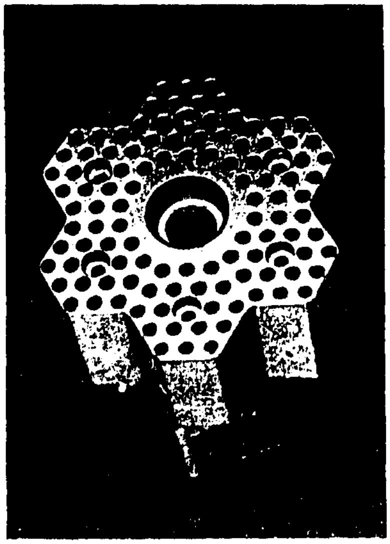
Comparisons of the effect of heterogeneous versus homogeneous modelling on the power distribution in the prismatic fuel assembly is shown in the left figure. The homogeneous model in a unit cell was derived by volume weighting of the prismatic fuel element, tie-tube materials, and hydrogen coolant of the tie-tube and fuel element. The comparison shows a peak to average local channel power of 9-10% for the explicit model of the unit cell. The smear modelling of each fuel element or tie-tube provides similar peak-to-average values. Shown in the right figure is the effect of a decrease in ZrH volume fraction or the introduction of cold (50K) hydrogen in the upward pass of the tie-tube. The maximum effect on local power occurs when the ZrH tie-tube is flooded with H_2 coolant at 50K.

ANALYSES OF 50K ENGINE DESIGN CONCEPT



- Reactivity Coefficients
- Component Nuclear Heating Rates
- Reactor Radiation Fields
  - Shielded (R-1)
  - Unshielded
  - Reduced Shield
- Material Temperature Limit Assessment
- Tie-Tube Thermal Analysis

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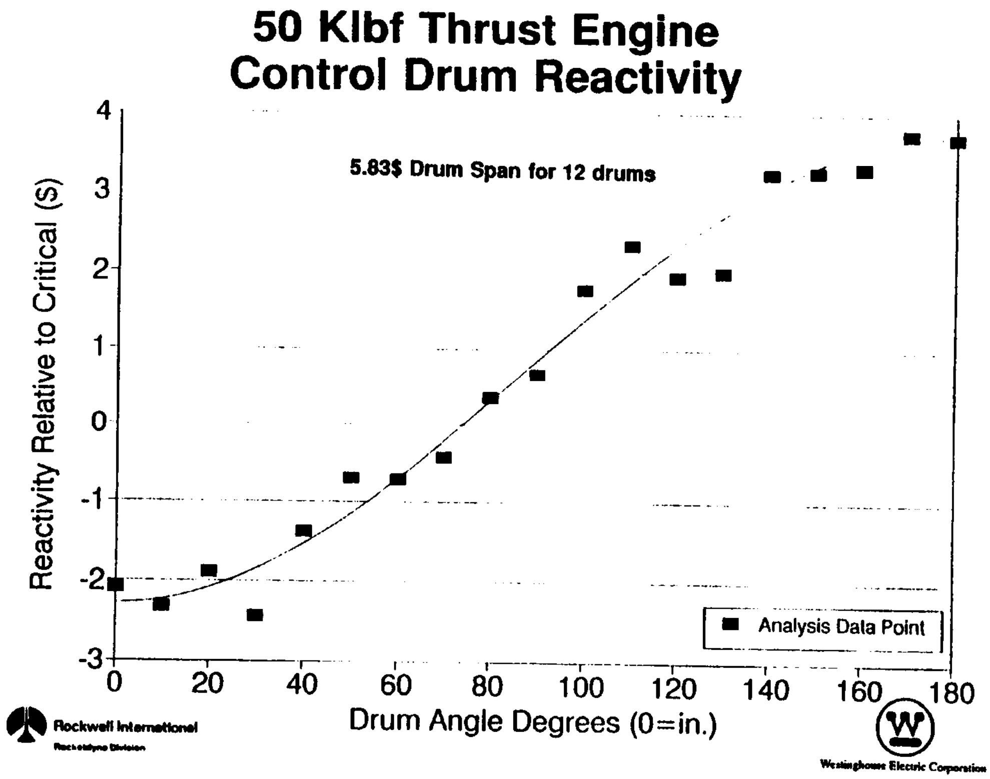
Neutronics analyses of the NTR 50K engine configuration defined earlier were expanded to provide more detailed core performance data. The limited analyses were performed with a more detailed MCNP model to predict the design data for key design parameters as listed on the facing page. Included in the more detailed analyses was; 1) the prediction of reflector control drum worths and span, and 2) reactivity change due to water immersion of the nuclear system. In addition, component nuclear heating rates and radiation fields external to the reactor system are predicted and shown in later pages.

In addition, evaluations of the component temperature limits needed for restartability studies and analyis of tie-tube thermal performance are shown in later pages.



The predicted reflector control drum reactivity relative to the critical condition is shown on the facing page. The results of the individual MCNP calculations with the explicit modeling of the control drums in MCNP method provide results in agreement with NERVA predictions and illustrate the drum span available for control and shutdown of the 50K engine.

Reactivity Coefficients

Case: 50 Klb_f Thrust Engine

PARAMETER CHANGED	REACTIVITY CHANGED
Drum Worth (@ 80°)	7.3¢/° Rotation
Core Volume	38.9¢/%
Fuel Loading	15.6¢/%
ZrH Loading	19.2¢/%
Reflector Thickness	18.7¢/%
18 Drums (7.34$ span vs. 5.83$ for 12 Drums)	-$2.1

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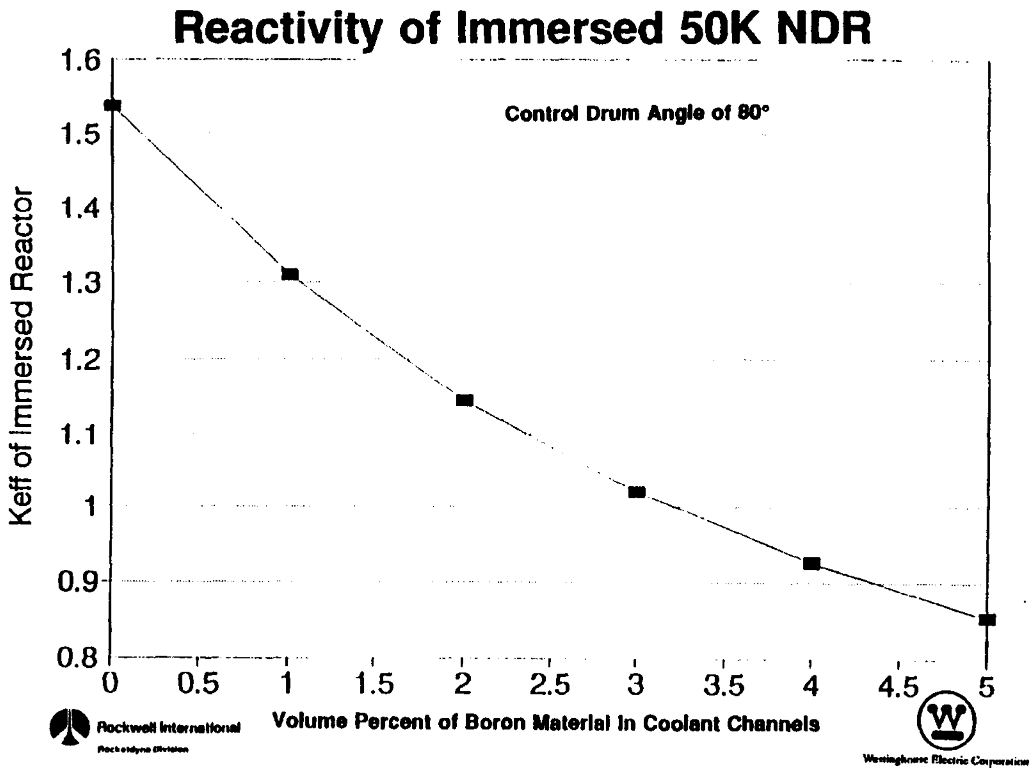
The predicted reactivity coefficients or worths for key design parameters are listed on the facing table. The predicted drum worth is based on the 80 degree postion. The value of 7.3 cents/degree is in close agreement with the NERVA predicted value. Reactivity coefficients for changes in the reactor configuration, fuel loading, ZrH loading in the tie-tubes, and reflector thickness provide data for evaluating design configuration changes. The largest value is the core volume coefficient which is attributed to the change in neutron leakage from the core. Shown also is the effect of changing the number of reflector control drums from 12 to 18 drums.



Predictions of the effect of water immersion of the entire reactor system was modelled in MCNP by replacing the H_2 coolant modelled in each region with water and surrounding the entire system with water. The reflector control drums were parked and a boron-containing material was substituted for a fraction of the fuel element coolant channel volume. The reactivity change from the base case is shown as a function of the volume percent of coolant channel displaced by the boron-containing material. A value of five (5) percent by volume of the coolant channel is a 62 mil boron wire in 7 out of 19 coolant channels in each prismatic fuel element of the core. The reactivity insertion provided by the 5% by volume of boron wires is approximately -74$ with the water immersion of the system resulting in a +50$ reactivity insertion.

COMPONENT HEATING IN 50K REACTOR

COMPONENT	Power, Mw	PERCENT OF TOTAL
Core Fuel and Supports	1000	97.66
Core Periphery (Filler & Seals)	6.6	0.65
Core Barrel Structure	2.4	0.24
Reflector & Control Drums	11.4	1.12
Core Support Plate & Hardware	1.4	0.13
BATH Shield	0.7	0.07
Balance of Reactor	1.4	0.13
Total	1024	

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A summary of the nuclear heating of the major components of the 50K engine is shown on the facing page. The MCNP cell tally method was used to predict the component heating rates.

REACTOR RADIATION FIELD TALLIES IN MCNP-3B CALCULATIONS

(Type and Units)

Radiation Field Type	Energy Bin	Units
Heating Rate In Hydrogen		W/kg
Heating Rate In Carbon		W/kg
Heating Rate In Stainless Steel		W/kg
Neutron Flux		n/cm²-sec
Neutron Flux, 1 MeV Equivalent in Si		n/cm²-sec
Neutron Fast Flux	> 1 MeV	n/cm²-sec
Neutron Intermediate Flux	0.1 MeV - 1.0 MeV	n/cm²-sec
Neutron Epithermal Flux	0.4eV - 0.1 MeV	n/cm²-sec
Neutron Thermal Flux	< 0.4eV	n/cm²-sec
Neutron Dose Rate In Hydrogen		Rad/hr
Neutron Dose Rate In Carbon		Rad/hr
Neutron Dose Rate In Stainless Steel		Rad/hr
Gamma Dose Rate In Hydrogen		Rad/hr
Gamma Dose Rate In Carbon		Rad/hr
Gamma Dose Rate In Stainless Steel		Rad/hr
Gamma Dose Rate In Silicon		Rad/hr

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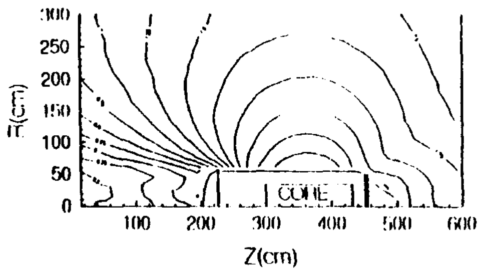
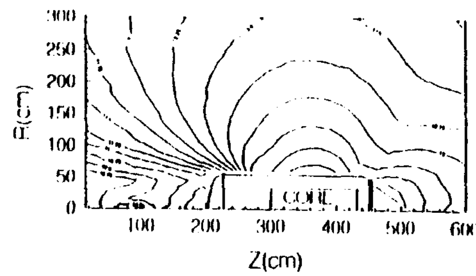
The prediction of the radiation environment external to the 50K engine were performed using the MCNP cell tally methods. Three engine models were analyzed; 1) the conceptual design sized using MCNP in the neutronics design tasks described earlier, 2) all internal shield materials removed, and 3) a modified design with a reduced mass of internal shielding. Each of these models only include the reactor system and the engine components external to the reactor vessel, e.g., tanks, piping, nozzle, are not included in the model. The engine components external to the reactor vessel can contribute to the environment within the internal shield shadow cone and should be included in future studies. The MCNP modelling used a series of annular ring cells imposed external to the MCNP R-Z model of the NTR reactor system for purposes of tallying the desired radiation environments. The facing page summarizes the type of radiation field tallies used in MCNP and either the neutron energy range of the neutron flux tally or the units of heating or neutron or gamma dose rates.

RADIATION FIELDS FOR A 50K SHIELDED REACTOR

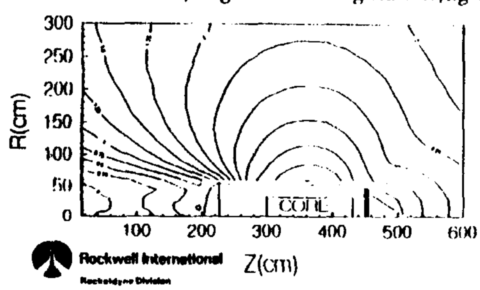
50K Shielded, Log Gamma Rad(C)/hr



50K Shielded, Log Fast Neutron Flux n/cm2-sec



50K Shielded, Log Total Heating Rate W/kg C



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- **Standard R-1 Shield**
- **12.3" Thick Bath Shield**
- **1.3" Thick Lead Shield**

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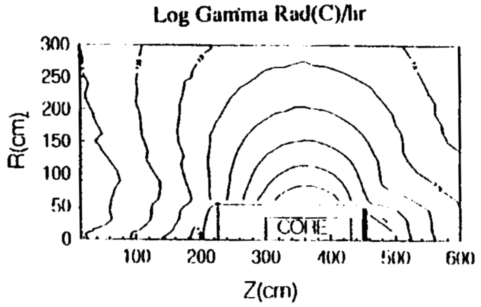
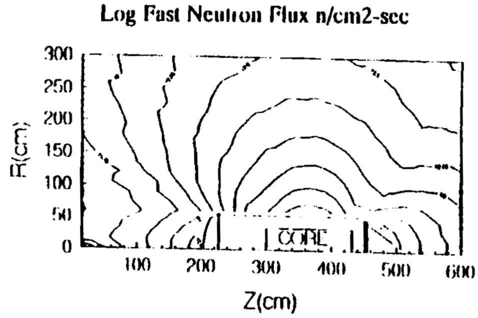
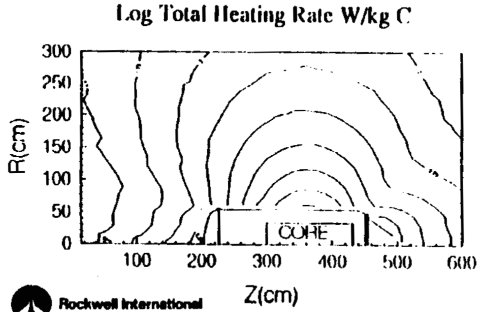
The radiation environment of the original 50K engine design is shown on the next two facing pages for three key tallies. The first 50K engine design used for this analysis included the standard NERVA R-1 internal shield configuration of 12.3 inches (31.25 cm) of BATH shield material and 1.3 inches (3.3 cm) of lead (Pb) shielding. The second page is for an engine design with the internal shields removed. The predicted radiation environments for the shielded case are lower than the design requirements.

RADIATION FIELDS FOR A 50K UNSHIELDED REACTOR







- **Zero Added Shielding**
- **No Bath or Lead**
- **No Shield Support Plates**

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RADIATION FIELDS FOR A 50K REACTOR WITH A REDUCED INTERNAL SHIELD

- **Radiation Field Criteria* (In Shield Shadow Cone)**
  - **Gamma Dose** $< 1.8 \times 10^{7}$ **Rad(C)/hr**
  - **Fast Neutron Flux** $< 2.0 \times 10^{12}$ **n/cm²-sec**
  - **Intermediate Neutron Flux** $< 3.0 \times 10^{12}$ **n/cm²-sec**
  - **Thermal Neutron Flux** $< 6.0 \times 10^{11}$ **n/cm²-sec**

- **Reduced Shield Concept:**
  - **Eliminates Lead Gamma Shield**
  - **Reduces BATH Thickness from 12.3" to 9"**

- **Reduced Shield Peformance ~ 900 lb. Reactor Weight over Standard Shield**
  - **900 lb Mass Savings versus R-1 Type**
  - **Meets Above Criteria (Design Margin > 2.0)**

* **Per NASA Directive**

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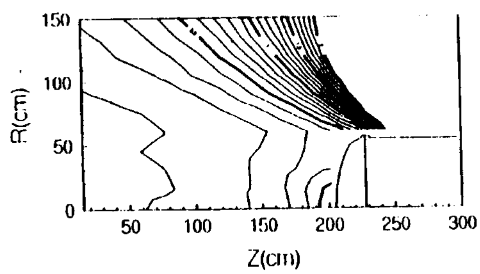
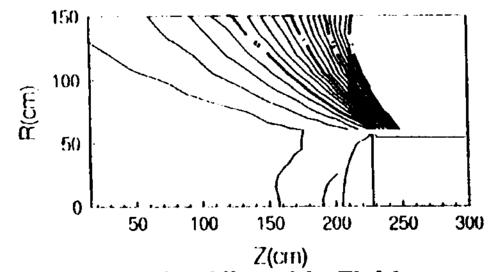
Based on the design requirements imposed on the internal shield design of the NTR engine, a reduced internal shield with nine (9) inches of BATH shield material and no lead (Pb) shielding was modelled and the resulting radiation environments compared to the standard design described earlier. The facing page lists the radiation field design requirements specified for the NTR engine. The reduced shielding configuraton meets design requirements with a design margin in the shadow cone of the internal shield of a factor of 2. The design change results in a reduction in shield mass of approximately 900 pounds..

RADIATION FIELD COMPARISON TO CRITERIA FOR 50K WITH REDUCED INTERNAL SHIELD

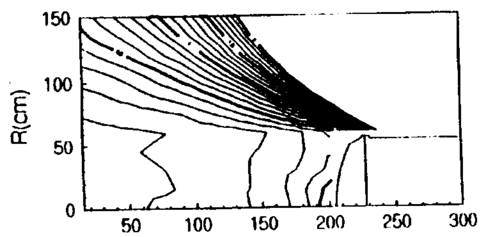
Ratio Gamma Rad/Allowable



Ratio Fast n Flux/Allowable



MAX(Each Ratio Flux/Allowable)



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- **Shield Size by Allowable Field**
- **9.0" Thick Bath, No Lead**
- **Meets Criteria by a Factor of 2**
- **Saves 900 lb. in Engine Weight**

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The contour plots on the facing page provide data on the performance of the modified shield configuration for the 50K engine relative to the design requirements. The contour data is the ratio of the predicted radiation environment level to the design requirement discussed before. As shown by the data, the reduced shield configuration meets the design requirements within the shadow cone of the internal shield. The design margin in the shadow cone is a factor of 2 or greater in the shadow cone. As discussed before the mass savings of the reduced internal shield design is 900 pounds.

MATERIAL TEMPERATURE LIMITS

REGION	MATERIAL	REUSE TEMP (K)	ALTERNATE MATERIAL**	TEMP (K)
Fuel Element	Graphite	2500		
Other Core Materials	ZrH_x	1000*	No	
	I-718	900	HD-Moly	2000
	A-286	900	Superalloys	~1400
	SS-304	750	Superalloys	~1400
Reflected Materials	Cu-B	1200		
	Be	1400	No	
Vessel Materials	Al-6061	400	Ni, Fe Alloys	
	Ti	800	Ni, Fe Alloys	
Shield Materials	BATH	550		
	Lead	~550	Tungsten	

*Must be pressurized with hydrogen (> 10 TORR)
**No materials identified which provide a capability without significant mass, performance or design penalty

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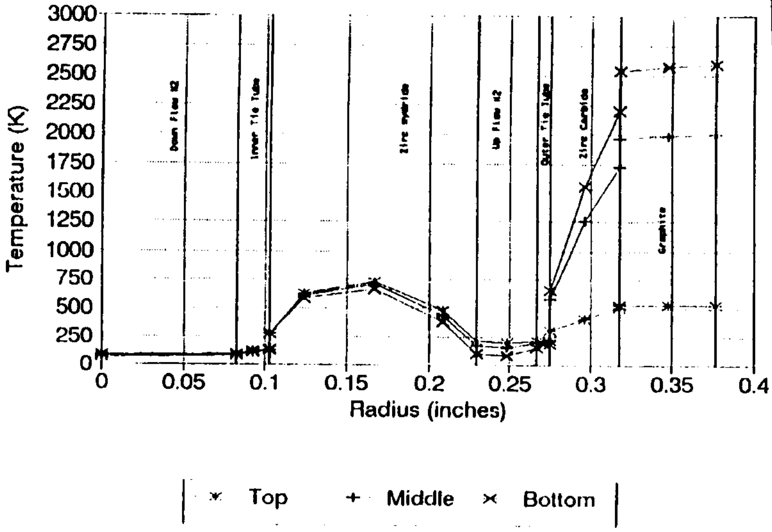
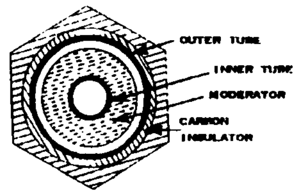
Temperature Profiles @ Elevations
in Tie Tube for Full Power Conditions





The tie tube assembly serves two purposes: provides the lateral support for the fuel elements, and heats hydrogen propellant used to drive the turbopump.

The thermal analysis of the tie tube assembly was performed to establish the adequacy of the design in terms of component temperature and to determine the energy transferred to the hydrogen. The thermal model used for the analysis employed axially dependent heat generation and boundary temperature conditions, temperature and flow dependent hydrogen heat transfer coefficient, and temperature dependent material properties. The thermal model will be used to perform parametric steady-state analysis, as well as transient analysis of throttling conditions.

The radial temperature distribution at three locations (top, middle, and bottom) of the tube assembly is shown on the facing chart for full power conditions.

The temperatures of the ZrH are critical since it has the lowest temperature capability of the materials used in the tie tube assembly. As shown, the maximum calculated temperature for the conditions used exceed 1000 K by a small amount at an internal node in the ZrH cylinder. The calculated heat transferred to the tie tube is 0.18 MW.

The thermal model has been verified against the small engine in the Nuclear Engine Definition Study. The analysis demonstrates that the thermal conductivity of the ZrC insulation is the largest factor in achieving the goal of 0.31 MW per tie tube.

ASSESSMENT OF FUEL TECHNOLOGY

- Review of ROVER/NERVA Test Experience
- Evaluate the Corrosion Mechanisms Affecting Fuel Performance
- Define Problem Areas Needing Near-Term Solution
- Establish Near-Term Fuel Performance Limits
- Compare Near-Term Performance to Fast Track Needs

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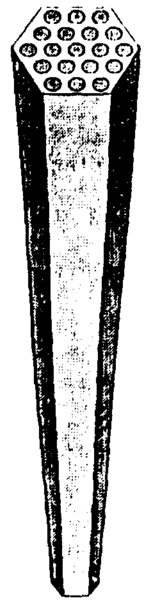
An assessment of the NERVA/Rover fuel technology of 1972 is needed to establish expected performance parameters for the Fast Track engine. The fuel life for the Nerva graphite type prismatic fuel element is determined by the amount of graphite weight loss which can be tolerated before the neutronic margin has been lost. The weight loss from the fuel element is due to the corrosive effect of hydrogen on the graphite, which is categorized as either "mid-band corrosion," basically results in a chemical reaction of hydrogen and carbon in intimate contact, or "hot end corrosion," carbon diffusion through a protective coating on the graphite surface.

Great strides were made near the end of the NERVA/Rover program in understanding and eliminating the mid-band corrosion, and it is a basic premise that this corrosion mechanism be suppressed in order to support the needs of the Fast Track program.

Based on the reactor/engine testing program, and the non nuclear corrosion testing of fuel elements using the improved GEM coatings, the performance limits of "near term" fuel elements were established. The expected fuel element performance was then compared against the needs of the Fast Track program

Fuel Elements

- **Sustain controlled nuclear heat generation**
  - **Pyrocarbon coated UC_2 fuel beads dispersed through AXM graphite matrix (630 mg/cc maximum fuel loading**
  - **UC_2-ZrC in composite with graphite (700 mg/cc maximum fuel loading)**
- **Limit total reactivity loss to $1.00 at end of life**
  - **Carbide coating of flow channels**
- **Promote heat transfer from fuel element to H_2 propellant**
  - **19 flow channels in each 3/4 in. HEX 52 in. long fuel element**



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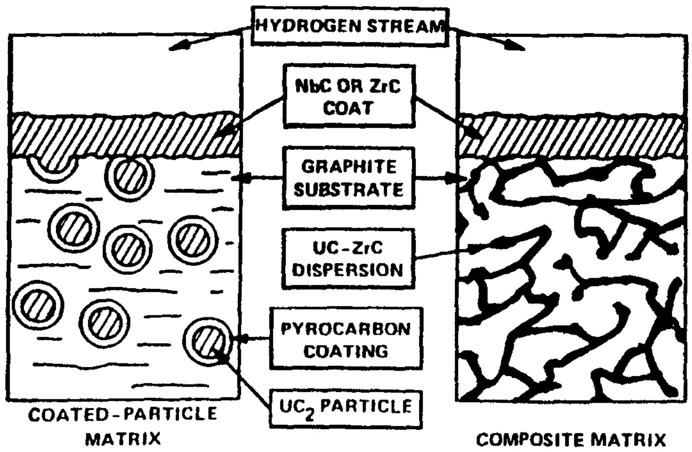
The NERVA/Rover prismatic graphite fuel element is 0.75 inch across the flats, and 52 inches long. It contains 19 flow holes (approximately 0.1 inch in diameter). All graphite surfaces have a protective ZrC or NbC layer to protect it from the hydrogen.

UC_2 fuel beads coated with pyrocarbon are dispersed through the matrix at a maximum fuel loading of 630 mg/cc. For the more recent composite type fuel element a maximum fuel loading of 700 mg/cc is achievable.

Nuclear design of the NERVA reactor limits the reactivity loss to approximately 1$ at the end of fuel life. Since the reactivity loss is mostly a result of loss of carbon due to the hydrogen corrosion, protective coatings are used to reduce the rate of carbon loss.

Fuel Element Comparison



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For all the NRX reactor and engine tests, the graphite-type fuel was used. However, toward the end of the NERVA/Rover program composite fuel emerged as the most promising candidate in reducing the hydrogen corrosion and in increasing the temperature capability of the prismatic fuel element.

The composite fuel element consisted for a dispersion of UC-ZrC web in the graphite substrate. Since this web is continuous, and essentially unaffected by hydrogen, it acts as a barrier and limits the carbon loss from the fuel.

Major Milestones in Fuel Development

- Graphite Fuel Element/HED NbC Coating (NRX-A2/A5)
- Graphite Fuel Element/HED NbC + Molybdenum Coatings (NRX-A6/XE')
- Graphite Fuel Element/GEM NbC/ZrC Coatings (PEWEE)
- High CTE Graphite Composite Fuel Element/GEM ZrC Coating (Nuclear Furnace -1)
- Carbide Fuel Element (Nuclear Furnace -1)

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The standard graphite fuel element with a HED NbC coating was used on NRX-2A/5A reactor series. The HED coating process resulted in a coating with a significant number of cracks, which seemed to have an adverse effect on the mid-band corrosion protection. In order to improve the mid-band corrosion performance of these elements, a molybdenum overcoat was applied to the fuel for NRX-A6/XE prime reactors.

The next improvement in the coating technology came with the lower temperature coating process, GEM, whereby ZrC or NbC coating could be applied without cracks in the coating. Fuel elements with this coating process were run in Pewee, but resulted in significant mid-band corrosion.

The fuel elements for the Nuclear Furnace-1 (NF-1) were of the high CTE graphite composite type with GEM ZrC coating, which were predicted to have eliminated the mid-band corrosion based on non-nuclear corrosion testing. Pure (U,Zr)C fuel elements were also tested in the nuclear furnace. These were manufactured as small hexagonal rods with a single cooling channel in the center. The carbide fuel elements were projected to have very low corrosion rates and very much higher temperature capability than both the graphite and the composite fuel elements.

NERVA/Rover Fuel Performance

REACTOR	FUEL EXIT TEMP. (K)	TIME AT MAX TEMP.(min)	AVG.WEIGHT LOSS PER ELEMENT (g)	TOTAL REACTIVITY LOSS (C)
NRX-A2	>2200	3.4	0.7	12
NRX-A3	>2400	16.5	16.5	58
NRX-EST	>2400	28.6	31.5	320
NRX-A5	>2400	30.1	27.1	223
NRX-A6	>2556	62.7	13.2	70
NRX-XE*	>2400	10.3	7.3	-
PEWEE-1	2750	43	20	
NF-1	2450	109	13.7	
PHOEBUS 1B	2445	30	13.7	

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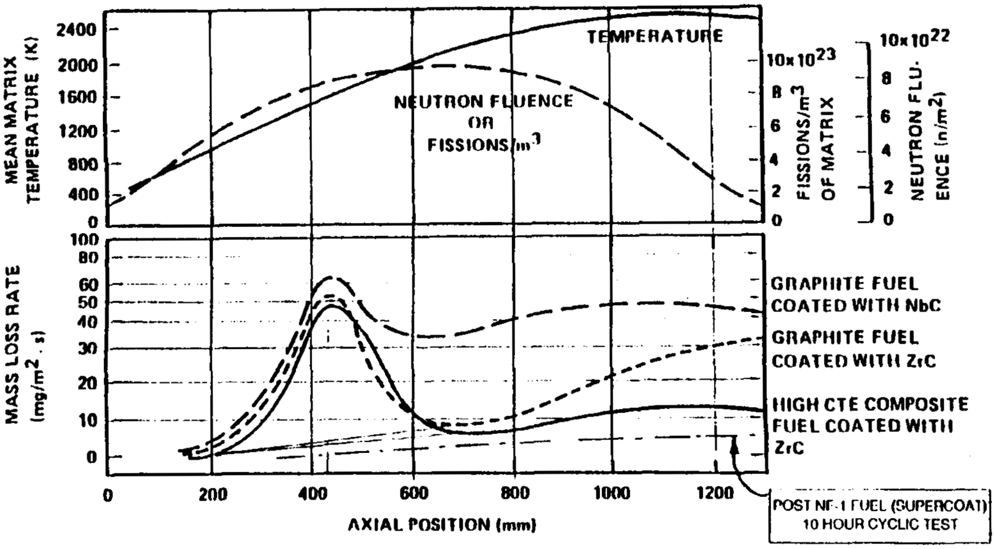
As a result of the improvements in the corrosion resistance of the fuel elements, the NERVA/Rover reactor tests showed a gradual increase in temperature capability and time at maximum temperature. The fuel life is dependent on the weight loss for the elements, and the resulting reactivity loss. Based on a reactivity margin of 1$ for corrosion from the fuel, the NERVA/Rover fuel life corresponds to a 15 to 20 g fuel element weight loss.

Corrosion of Rover/NERVA Fuel Elements



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The corrosion behavior along the length of the fuel element showed two different characteristics. From an axial position of 200mm to approximately 650mm from the cold end, an enhanced corrosion (called the mid-band corrosion) dominated. The temperature regime for this mechanism is 1000 to 2000 K, significantly below the maximum fuel temperature. In the progression of coating and fuel element improvements, there seemed to be negligible improvement in mid-band corrosion except for the demonstrated benefit of the molybdenum overcoat. From approximately 650mm to the hot end of the fuel element (called the hot end corrosion), the corrosion rate seemed to temperature related, and a significant decrease in the corrosion rate was observed as the coatings were improved. Electrically heated fuel element corrosion tests performed after the NF-1 testing demonstrated further improvements in the hot end corrosion rate, including a 10-hour life of a fuel element demonstrated by Westinghouse.

Key Reference Points For Fuel Experience

REACTOR TEST	TEMP FUEL EXIT (K)	TIME (MIN)	CYCLES	TOTAL LOSS (G)	MIDBAND (G)	HOT END (G)
NF-1 *	2444	108.8	4	13.7	8.6	5.1
NRX-A6 **	2556	62.7	1	12.8	2.3	10.5
NRX-XE' **	~2450	10.3	28	7.3	0.6	6.7

* Replacement composite fuel elements with crack free ZrC coating (GEM)
** Graphite fuel elements with NbC coating and molybdenum overcoat

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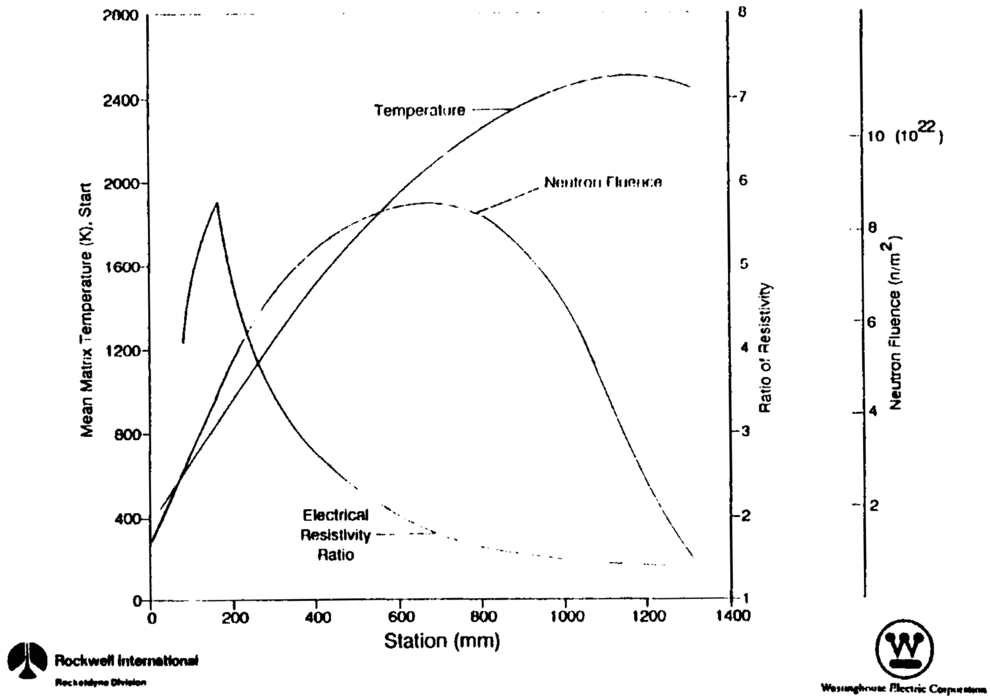
The most successful graphite fuel elements were those tested in NRX-A6, which were also used in NRX-XE prime engine configuration. These fuel elements utilized the HED NbC coating with molybdenum overcoat, and demonstrated a significant reduction in the mid-band corrosion compared to earlier NRX series tests.

The alternative fuel element technology is the composite, which was tested in NF-1. These elements, which were called the "replacement elements," were high CTE graphite coated with a superior ZrC coating (free of initial cracks) applied by GEM process.

The weight loss results for the A-6 and the XE prime fuels indicate that the A-6 vintage fuel has a significant sensitivity to thermal cycling. The NF-1 composite fuel elements demonstrated better hot end corrosion than the A-6 graphite fuel; however, a surprising degree of mid-band corrosion was still present.

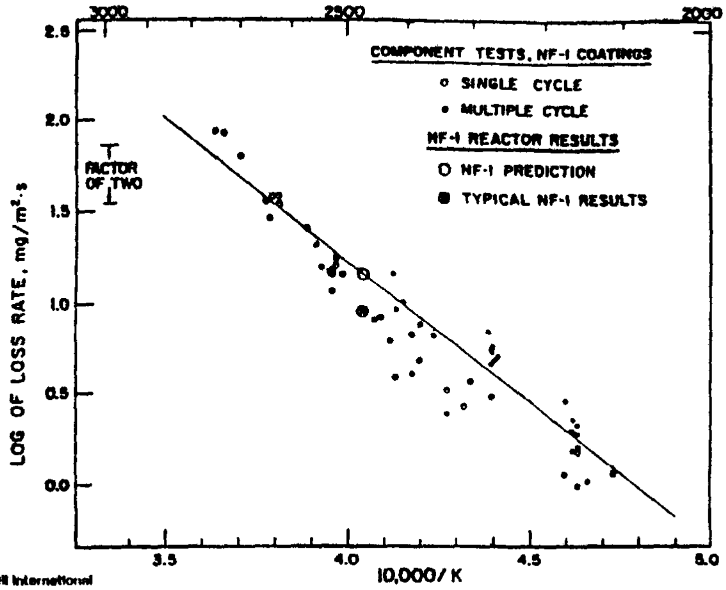
NF-1 Fuel Damage Explanation



Mid-band corrosion did not occur in the electrical testing of the composite fuel elements for NF-1, but caused the most significant weight loss during the reactor testing. Mid-band corrosion is believed to be a result of decreased thermal conductivity, possibly caused by fission fragment damage to the graphite matrix. The reduced thermal conductivity results in higher thermal gradients and increased thermal stresses, which causes cracking of the protective coatings and allows hydrogen to react with the graphite substrate. Mid-band corrosion must be fully understood and suppressed to meet performance requirements of the Fast Track program. Use of a molybdenum overcoat on composite fuel elements, or improved fuel particle coating in the graphite fuel to trap the fission fragments, are potential design solutions to mid-band corrosion.

Composite Fuel Element

Hot End Corrosion Rate
Bore Surface Temperature, K



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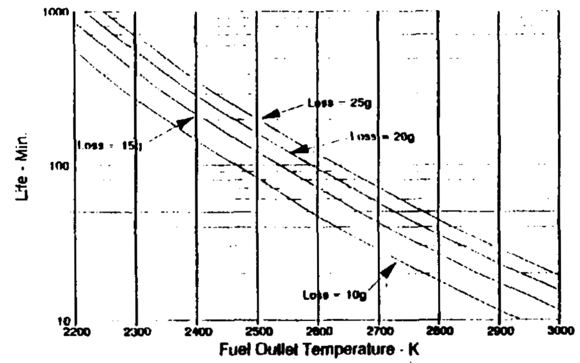
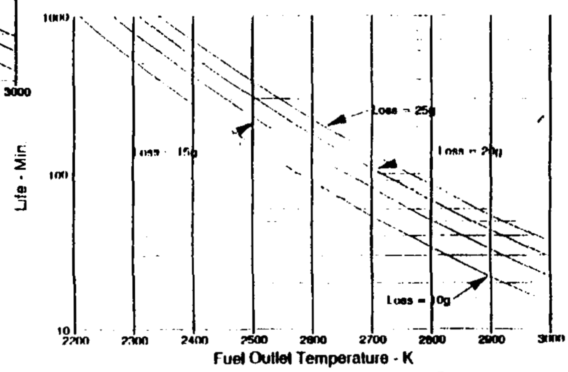
Composite fuel element testing provided a good correlation between the hot end corrosion measured in electrical testing and that observed in the NF-1. Hot end corrosion is caused by carbon diffusion through a protective coating and, therefore, is sensitive to the integrity of the coating, the coating thickness, and the temperature of the coating and fuel substrate.

Near Term Fuel Element



Graphite
(A6 Technology)

Composite
(NF1 Technology)



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Based on the assumption that the mid-band corrosion will be suppressed in near-term fuel elements, and the hot end corrosion rates measured in electrical testing and NF-1 testing, performance limits for near-term composite fuel can be calculated. Similar performance data can also be generated for the NRX-A6 type graphite fuel.

Comparing the projected near-term graphite fuel performance NRX-6A type with the composite fuel (NF-1 type) shows a 100-120 K temperature advantage for the composite fuel.

The improved performance of composite fuel is attributed to either the projected improvements in corrosion due to the composite fuel form or improved coatings used for NF-1 fuel elements. The improved coatings of NF-1 fuel elements are considered the most likely contributor to improved fuel performance.

Summary and Conclusions

- 1972 Fuel Technology was making progress toward meeting life/performance specifications consistent with current Fast Track requirements
  - understanding of midband corrosion was being developed
  - excellent hot end corrosion protection (ZrC on high CTE graphite) was demonstrated
- Corrosion limit for fuel elements was established based on 1$ reactivity loss
  - for NERVA type reactors, this translates into 15 to 20 grams corrosion loss per element
- Near term fuel development must resolve midband corrosion problem
  - fission fragment damage to graphite may be reduced by beaded fuel in graphite and composite matrix
  - molybdenum overcoat may suppress midband corrosion
  - improved graphite matrix may reduce or eliminate problem
- Near term composite fuel will have 4.5 hours life at 2470K to 2520K fuel outlet temperature
  - near term graphite fuel based on GEM ZrC/high CTE graphite is expected to perform similarly to near term composite fuel
- Near term fuel elements are expected to provide ISP ~850 seconds

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Assessment of Nuclear Safety Issues

- **Nuclear Safety Policy Working Group (NSPWG) Recommendations**
- **Accidental Criticality Sources for NERVA Derivative Reactor Design**
- **NERVA Safety Approach**
- **SP-100 Safety Approach**
- **NERVA Derivative Safety Approach**

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An assessment of the nuclear safety issues for a nuclear thermal propulsion system must be made based on the current regulatory guidelines, and the recommendation from the Nuclear Safety Policy Group (NSPWG). Starting with the accidental criticality sources for the NERVA derivative reactor design, the safety approached developed for the NERVA flight engine and the current SP-100 reactor safety approach, and the planned NERVA derivative safety approach will be discussed.

Assessment of Nuclear Safety Issues
From NSPWG & NP002 Safety Recommendations

- **No inadvertent reactor startup**
  - **Zero power testing on ground**
  - **Startup after achieving planned orbit**

- **No inadvertent criticality**
  - **Subcritical under all credible accident conditions**
  - **Highly reliable control system**

- **No significant radiological release or exposure**
  - **Only zero power testing prior to achieving planned orbit**
  - **29CFR1910.96 dose limits to flight crew**
  - **Insignificant impact to population of Earth**
  - **Insignificant impact on Earth and space environment**
  - **Spacecraft not rendered unusable when crew survives accident**
  - **Radiological release not impair use of spacecraft**

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Assessment of Nuclear Safety Issues
NSPWG Safety Recommendations (Cont'd)

- **No planned reentry**
  - **Minimize probability of inadvertent reentry**
  - **Minimize consequences of inadvertent reentry (high alt. disposal or intact reentry)**
  - **Subcritical at all times**
  - **Minimize impact dispersion**
- **Minimize hazardous materials release**
- **Ensure safe disposal**
  - **Part of mission planning**
  - **Adequate and reliable cooling, control and protection**
  - **Ensure non-premature final shutdown**
- **Safeguard nuclear material**
  - **Positive measures to prevent theft, diversion, loss or sabotage**
  - **Features to enhance safeguards and permit proven methods to be employed**
  - **Positive measures or features for recovery including jettison and tracking**

The NSPWG recommendations for safety requirements and guidelines addresses the protection of the public, the crew, the environment (both Earth and space environment), and includes recommendations for the safe disposal of the spent reactor system.

Assessment of Nuclear Safety Issues

Accidental Criticality Sources and Potential Countermeasures

Accidental Criticality Sources:

Source	Maximum Reactivity Insertion
Core Compaction	~ $6 (80% Theoretical Density)
External Neutron Reflection	≈ $3
Control Drum Roll-Out	$3 ($4.50 Drum Span)
Hydrogen Insertion	≈ $83*
Water Immersion	≈ $76

Potential Countermeasures:

C/M	Negative Reactivity Worth
Central + Peripheral Poison Wires	≈ $90
Central Poison Wires Only	≈ $10
Control Drums "Locked" Full-In	$1.50 at Ambient Temperature
Safety rods	≈ $90

*Limiting reactivity addition if the core could be completely flooded with high density LH_2.

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The sources for accidental criticality of a NERVA derivative reactor are core compaction, external neutron reflection (from water or soil), control drum rollout, hydrogen flooding, and water immersion. The countermeasures for reactivity events must assure a subcritical condition with a negative reactivity margin of 1$.

For the NERVA reactors the criticality margin was assured using poison wires (7 for each fuel element). Other reactivity control means for NERVA-type reactors are the control drums or the possible introduction of safety rods within the core.

Assessment of Nuclear Safety Issues

NERVA Safety Approach

- **Poison wires in core after assembly for shipping**
  - **~7 Boron/aluminum wires/elements**
  - **Wires would be removed before launch**
- **Redundant safety features to preclude drum roll out**
  - **Permanent magnet stepping motor used in control drum drive actuator**
  - **Drum rollout requires erroneous command signal and closing electrical power circuit**
- **Anticriticality Destruct System (ACDS)**
  - **To fracture reactor by use of explosives**
  - **No more component greater than 3 fuel element**
- **Prevention of hydrogen insertion**
  - **Closing PFS valves when flooding is detected within 300 seconds of full leakage**

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For the NERVA reactors the poison wires were primarily used to maintain the fully assembled and fueled reactor in a safe condition during transportation from the assembly area in Large, Pennsylvania, to NRTS. For a flight reactor the poison wires were to be removed prior to launch. Redundant safety features were used to preclude drum roll-out prior to the planned reactor startup in a safe orbit. To preclude criticality events for a launch accident or an inadvertent reentry event, an Anticriticality Destruct System (ACDS) would be used to break up the core.

Hydrogen flooding of the core was precluded using redundant valves and hydrogen sensors.

Assessment of Nuclear Safety Issues
SP-100 Safety Approach

- **Two redundant shutdown systems**
  - **Moveable reflector segments**
  - **Multiple safety rods**
    - **Only planned use for ultimate shutdown**
- **Rhenium liner at core periphery to absorb thermal neutrons**
  - **Water immersion**
- **Inadvertent reentry and Earth impact**
  - **Reactor remains intact and subcritical**

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The SP-100 space power reactor system (SPRS) has been subjected to more extensive safety evaluations based on current guidelines. The decisions made and planning for the SP-100 SPRS will most probably apply to the NTR.

The SP-100 safety approach employs two redundant systems, moveable reflectors, and safety rods. The safety rods are designed to provide for permanent shutdown of the SP-100 reactor system after the completed mission However, the safety rod design allows for the retraction of the rods from an unplanned insertion.

In addition to the moveable reflectors and safety rods, the SP-100 reactor includes a rhenium liner internal to the reactor vessel to capture neutrons thermalized external to the vessel and precludes back reflection from a water or earth immersion event. The SP-100 safety approach includes reactor system design features to assure an intact inadvertent reentry and earth impact event.

Assessment of Nuclear Safety Issues
Safety Approach for NERVA Derivative Reactor

- **Preliminary safety evaluations have been initiated**
- **Current safety guidelines appears to require dual shutdown systems**
    - **Control drums for normal operation**
    - **Safety rods for ultimate shutdown**
- **As part of the safety study, the design team is evaluating**
    - **Retractable safety rods**
    - **Neutron absorption at core periphery for Earth and water immersion**
    - **Impact of intact reentry**

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There has been no in-depth safety evaluation of the NERVA derivative reactor system completed to date. However, it is expected that the results of such an evaluation will be similar to SP-100 safety approach adapted to the reactor design. Based on the current safety guidelines, incorporation of dual or redundant safety shutdown systems will be needed to meet today's requirements.

As part of an ongoing safety evaluation for the NERVA derivative system Westinghouse will evaluate the use of retractable safety rods in the core and neutron absorbing liners at the core periphery to achieve the current safety guidelines. The design impact of an intact reentry will be evaluated for the reactor design.

Reactor Development
Summary and Conclusions

- **Engineering and analysis of NERVA derivative reactors were successfully benchmarked against the NERVA/Rover test reactors for:**
  - **Reactor size and neutronics performance**
  - **Design characteristics such as fuel loading, ZrH moderator requirements, and control drum span**
  - **Internal shielding performance**
  - **Thermal performance of tie tubes**
- **A reactor conceptual design for the NASA 50K fast track engine was validated neutronically and thermally by analysis.**
- **The 1972 NERVA/Rover fuels technology must be recaptured and demonstrated.**
- **Near term fuel technology must resolve the mid-band corrosion problem.**
- **Near term fuel technology will meet fast track requirements.**
- **NERVA/Rover safety shutdown systems appears inadequate for today's requirements.**
  - **A secondary shutdown system will be developed for the NERVA derivative reactor designs.**

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In this project we performed trade-off studies, developed a single point conceptual reactor design and validated this design thermally and neutronically.

The engineering and analysis supporting the trade-off studies and the point design were successfully benchmarked against the NERVA/Rover reactor designs. The reactor size and neutronic performance was established for a range of reactor sizes for engines producing 25 K to 75 K lbs thrust. The design characteristics such as fuel loading requirements and radial loading profile, ZrH moderator requirements, control drum worths and control span, were established.

A reactor concept for the 50 K lbf engine was developed and validated neutronically and thermally by analysis.

Internal shielding performance was established for the standard R-1 shield configuration, an unshielded, and for a reduced shield reactor.

The tie tube thermal performance was modelled, and evaluated for steady state conditions. Trade studies will establish the range of ZrC insulation properties and thermal transient performance.

The fuel technology of 1972 (the end of the NERVA program) was evaluated. This technology must be recovered and demonstrated as a baseline. Further, this technology must be advanced by eliminating or suppressing the midband corrosion problem to meet the fuel life requirements for the proposed missions. This "near term" fuel technology will meet the needs of the fast track program.

Reviewing the current requirements and recommendations for nuclear safety for the NTR, and the approach taken by other space power reactor systems, leads to the conclusion that the NERVA/Rover safety approach must be upgraded. Current plans are to evaluate a secondary shutdown system for the NERVA derivative reactor design.

50K PEWEE-DERIVED DUAL TURBOPUMP ENGINE

T_c 2550 K

SPLIT-FLOW EXPANDER CYCLE

ϵ 200:1

% L 110%

I_s 870 SEC

T/We 5.3 (W/SHIELD)

L_e 7.66 M

D_e 2.44 M

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50K PEWEE-DERIVED DUAL TURBOPUMP ENGINE

The 50Klb.-thrust engine is based on Rover/NERVA reactor core technology. Average fuel element exit gas temperature and core power density milestones were established by the Pewee reactor tested at NTS to a power level of 500 Mw in December 1968. The average fuel element exit gas temperature was approximately 2550K (4600R) with an average 1.18 Mw per fuel element. The Pewee reactor was also the first to incorporate zirconium hydride moderator in the tie-rod core support elements. Regeneratively-cooled tie-tubes rather than dump-cooled tie-rods were used in the Phoebus-2 reactor tested at NTS to a power level of 4,000 Mw in July 1968. Westinghouse used these important characteristics and results in establishing core preliminary design for the 50K reactor and the companion 25 and 75K reactors.

Dual turbopumps are used to provide an element of redundancy, as were used in the NERVA R-1 engine design. Valves provide isolation for a shutdown turbopump. The turbines are powered through a split-flow expander cycle where energy is derived from cooling the tie-tubes and nozzle in parallel.

Based on envelope considerations, the nozzle expansion ratio was set at 200:1 and the bell contour length at 110% of that for a 30° conical nozzle. For the 200:1 expansion ratio, this length provides the optimum nozzle thrust coefficient for hydrogen, resulting in a specific impulse of 870 seconds.

Hydrogen is bled from the turbine exhaust and regulated to provide engine pneumatic power and stage tank pressurization. Hydrogen is stored in two tanks to provide pneumatic power for engine restart. Pressurization of the stage Liquid Hydrogen tank is not required for starting due to 2-phase pumping capability of the pumps. If LH_2 tank pressure is below approximately 35 psia, then boost pumps are probably required.

The engine thrust-to-weight ratio is 5.3, including shielding which provides approximately half the maximum radiation field requirements of NPO. Without shielding, the T/We is 5.8.

Provisions for thrust vector control have been made by incorporation of a gimbal bearing at the top of the pressure vessel dome, and by two orthoganal outriggers for accepting gimbal actuators. The outriggers are mounted to the upper portion of the pressure vessel. Motion of the pump inlet during gimballing is accommodated by the scissor bellows, similar to those used on J-2 engines.

The engine length from the top of the gimbal bearing to the exit plane of the nozzle is 7.66M. The exit diameter of the nozzle is 2.44M.

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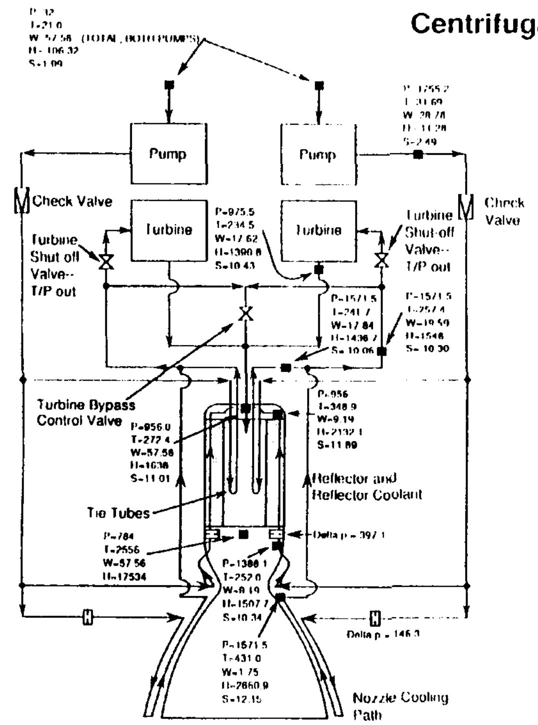
50K NTR, Expander Cycle, Dual T/P*
Centrifugal Pump



DESIGN VALUES	
PUMP FLOWRATE (TOTAL)	57.56 LB/SEC
PUMP DISCHARGE PRESSURE	1,755 PSIA
NUMBER OF PUMP STAGES	2
PUMP EFFICIENCY	72.58 %
TURBOPUMP RPM	47,500 RPM
TURBOPUMP POWER (EACH)	3,870 HP
TURBINE INLET TEMP	257.4 K
NUMBER OF TURBINE STAGES	1
TURBINE EFFICIENCY	72.56 %
TURBINE PRESSURE RATIO	1.611
TURBINE FLOW RATE (EACH)	17.62 LB/SEC
REACTOR/ENGINE THERMAL POWER	1,031.7 MW
FUEL ELEMENT TRANSFERRED POWER	965.1 MW
CORE THERMAL POWER (FUEL ELEMENT+TIE TUBE)	1,019.6 MW
ENGINE THRUST	50,000 LBF
NOZZLE CHAMBER TEMPERATURE	2,556 K
CHAMBER PRESSURE (NOZZLE STAGNATION)	784 PSIA
NOZZLE EXPANSION AREA RATIO	200:1
NOZZLE PERCENT LENGTH	110%
VACUUM SPECIFIC IMPULSE (DELIVERED)	868.72 SEC

Heat loads are as follows:		
	Nozzle-con (total):	29.44 MW
	Nozzle-div (total):	9.86 MW
	Reflector (total):	12.10 MW
	Tie-Tubes (total):	54.50 MW

P = PSIA
T = DEG K
W = LB/S
H = BTU/LB
S = BTU/LB-R

***Note: Flows indicated are for one-half of system.**

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50K NTR, EXPANDER CYCLE, DUAL T/P CENTRIFUGAL PUMP

The 50K full-thrust system balance propellant conditions are shown at key points on the schematic. Significant design values and component heat loads are presented in the tables.

The pump inlet pressure was set at 32 psia, allowing a tank pressure of approximately 35 psia. The engine flowrate is 57.56 lb/sec (28.78 lb/sec for each pump). The pump pressure is 1,755 psia, resulting in a turbopump power of 3,870 Hp (each). Approximately 60% of pump flow goes to the tie-tubes and approximately 5% goes to the nozzle jacket, providing cooling and energy to power the turbine. The balance of the pump flow cools the chamber jacket and the reflector before joining the flow exhausted from the turbines.

The nozzle jacket (nozzle-div) and tie-tubes provide total power of approximately 64 Mw to heat the turbine drive loop to 257K with a turbine inlet pressure of 1571 psia. The single stage turbine has a flowrate of 17.6 lb/sec and a pressure ratio of 1.61 to drive the pump at 3,870 Hp and 47,500 rpm. Approximately 10% of turbine flow is bypassed around the turbine through a control valve to provide overall engine control in conjunction with the reactor control drum actuators.

The turbine exhausts and turbine bypass flows are combined and discharged into the pressure vessel dome where they join with the flow which cooled the chamber jacket and reflector. The total engine flow of 57.56 lb/sec then cools the fuel elements of the reactor with a power of 965 Mw. Hydrogen exits the reactor at 2556K (4600R) and 784 psia. This expands through the 200:1 expansion ratio, 110% length bell nozzle, providing a specific impulse of approximately 869 sec and thrust of 50,000 lb.

A check valve function is provided at each pump discharge and a shut-off valve function at turbine inlet so that a malfunctioning turbopump can be isolated while maintaining engine operation. These functions may be satisfied by a series/parallel arrangement of valves as was done with the NERVA R-1 engine. Likewise, the schematic indicates only a single turbine bypass control valve. The arrangement of valves to provide redundancy and meeting "Nuclear Thermal Rocket Engine Requirements," NASA N.P. #002, has not yet been addressed.

SPECIFIC IMPULSE ADVANCEMENT

• XE-PRIME	710 SEC
Tc 2270K BLEED CYCLE (TURBINE 10% FLOW) NOZZLE ε OF 10	
- REGEN COOL FUEL ELEMENT SUPPORTS AND CORE PERIPHERY	+35
- INCORPORATE EXPANDER CYCLE	+35
- INCREASE ε TO 200, 110%L	+50
- INCREASE Tc TO 2550K (PEWEE)	+40
• PEWEE-BASED, GRAPHITE ELEMENT ENGINE I_s	870 SEC
- INCREASE Tc TO 2700K	+30
• COMPOSITE ELEMENT, ENGINE I_s	900 SEC

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SPECIFIC IMPULSE ADVANCEMENT

The XE-Prime is the baseline for nuclear thermal rocket engines, since it is the only engine configuration ever tested. This experimental engine was tested during much of the year in 1969, but full power and performance were most notably achieved in June when a chamber temperature of 2270K (4090R) was achieved. Due to the facts that a low expansion ratio (10:1) ground test nozzle was used, and that a bleed cycle was used to power the turbine which exhausted 10% of the engine flow at low specific impulse; an engine specific impulse of only 710 seconds was realized. Specific impulse is increased by 35 seconds by using regeneratively cooled (tie-tube) fuel element supports in place of dump-cooled, tie-rod fuel element supports, and by using regenerative cooling instead of dump-cooling in the core periphery where the transition is made from the irregular boundary of the hexagonal fuel elements to the circular boundary of the seal segments for sealing and bundling the core.

Specific impulse is increased by 35 seconds by using the expander cycle where the turbine exhaust is combined with the balance of the engine flow and the total flow is exhausted at the high reactor exit temperature rather than using the bleed cycle where the turbine flow (10% of the engine flow) is exhausted at low temperature and degrades engine specific impulse.

Specific impulse is increased by 50 seconds by increasing the nozzle expansion ratio from the experimental ground test engine value of 10:1 to 200:1 expansion ratio for a flight engine and using a 110% bell contour which provides the optimum thrust coefficient for hydrogen at this expansion ratio.

Specific impulse is increased by 40 seconds by increasing reactor exit gas temperature from the 2270K (4090R) of XE-Prime to the 2550K (4600R) of the Pewee Reactor test.

Cumming the above advancement results in the Pewee-Based, Graphite Element, Engine Specific Impulse of 870 seconds, since the Pewee Reactor used graphite fuel elements.

Specific impulse is increased by an additional 30 seconds if composite fuel elements where a reactor exit gas temperature of 2700K (4860R) can be achieved based on data from Nuclear Furnace, are used rather than the graphite fuel element with a reactor exit gas temperature of 2550K (4600R) based on data from Pewee Reactor testing. This results in the Composite Element (Nuclear-Furnace-Based) Engine Specific Impulse of 900 seconds.

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ENGINE LENGTH AND NOZZLE SIZING

25K ENGINE

STAGE REQUIREMENTS

ENGINE LENGTH - 6.0M
ENGINE I_s - 870 SEC

CHAMBER PRESSURE INCREASED

FROM 621 PSIA (PEWEE)
TO 784 PSIA
MEETING REQUIREMENTS AND RESULTING IN:

ε 200:1
110% LENGTH

50 AND 75K ENGINES

USED 25K NOZZLE PARAMETERS

ε 200, 110%L

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ENGINE LENGTH AND NOZZLE SIZING

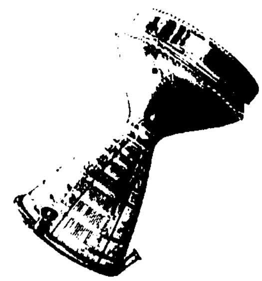
Initial effort in the program covering 25, 50 and 75K thrust engines was directed on the 25K engine, since stage requirements were provided for this engine. Engine length was limited to 6.0 meters with a specific impulse of 870 seconds.

To meet the stage requirements, the chamber pressure of 621 psia from the Pewee test condition had to be increased to 784 psia. The higher pressure is beneficial to the reactor core with regard to heat transfer and pressure drop. The resulting nozzle has an expansion area ratio of 200:1 and a bell contour length of 110% of that for a 30° conical nozzle. With hydrogen, the 110% length provides maximum nozzle thrust coefficient for an area ratio of 200:1.

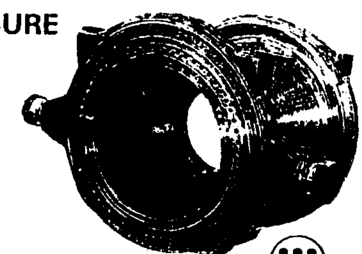
For the 50 and 75K engines, the same nozzle parameters of 200:1 expansion ratio and 110% length were used to result in a consistent family of engines from the standpoints of envelope, performance and weight.

NON-NUCLEAR COMPONENT TECHNOLOGY

- CHAMBER TECHNOLOGY
  - ROVER-KIWI, PHOEBUS
    - INCONEL-X TUBES
    - INCO 718 SHELL
    - FURNACE BRAZED ASSEMBLY
    - LIGHTWEIGHT
    - HIGH TEMP CAPABILITY



  - SSME
    - SLOTTED FORGED NARLOY
    - ELECTRODEPOSITED CU/NI CLOSURE
    - INCO 718 SHELL
    - HIGH HEAT FLUX CAPABILITY
    - LOW WALL TEMPERATURE
    - SUPERIOR LIFE CAPABILITY



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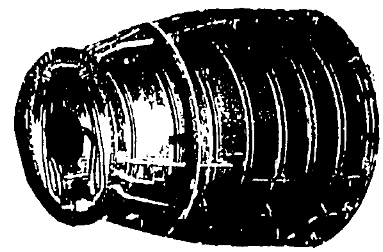
NON-NUCLEAR COMPONENT TECHNOLOGY CHAMBER TECHNOLOGY

Rocketdyne has two technologies applicable to the NTR Chamber, the convergent and low area ratio divergent component which attaches to the bottom of the pressure vessel and ducts the reactor exit gas through the sonic region, delivering it to the high expansion ratio nozzle. One technology comes from earlier rocket engine programs, including the Rover program where tubular-wall chambers were employed, and still are today for engines such as Atlas and Delta. The other technology is the slotted, one-piece, copper-wall chamber used for the SSME.

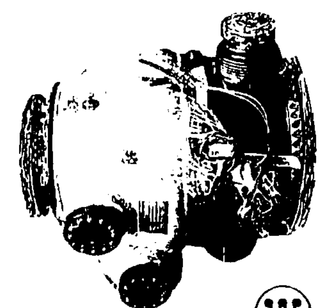
The Rover tubular-wall chambers (as shown in the photo) were used for 7 of the 19 reactors tested in the Rover/NERVA program, including Phoebus 1B at conditions approaching 1500 Mw, 750 psi chamber pressure, and throat heat flux of 30 BTU/in^2 sec. These chambers have a contraction ratio of approximately 20 to interface with the reactor at an inlet diameter of approximately 35 inches, and an expansion ratio of 12 to exhaust into the atmosphere at NTS conditions. Inconel-X tubing was used with an Inconel 718 one-piece forged Shell/Flange. The chamber was a furnace-brazed assembly. This technology provides a lightweight chamber with approximately 1000K (1760R) wall temperature capability.

The SSME slotted, one-piece, copper-wall chamber (as shown in the photo) was developed for and used on all Space Shuttle Main Engines. Three SSME's on each Space Shuttle flight have now powered over 50 missions, and flight configuration engine testing exceeds 120 hours. The SSME chamber operates at a chamber pressure of approximately 3000 psi with a wall temperature at the throat of approximately 800K (1460R) and heat flux of approximately 100 BTU/in^2 sec. The chamber has a contraction ratio of approximately 3 and an expansion ratio of 5 with a throat diameter of approximately 10 inches. The slotted, forged NARloy (Rocketdyne copper alloy) chamber liner is electrodeposited on the outer envelope with a thin copper and then heavier nickel closure of the coolant slots. A welded Inconel 718 shell, manifold and flange assembly complete the chamber. This technology provides high heat flux (100 BTU/in^2 sec) capability with low (1000F) wall temperature. Although the weight is somewhat higher for an NTR chamber than with the tubular-wall Rover chamber technology, the SSME chamber technology is favored due to superior Life-Cycle capability and general robustness.

NON-NUCLEAR COMPONENT TECHNOLOGY (CONT'D)



- NOZZLE TECHNOLOGY
  - SSME
    - A-286 TUBES
    - FURNACE BRAZED ASSEMBLY
- TURBOPUMP TECHNOLOGY
  - INDUCER
    - Mk 15F, Mk 25
      - 2-PHASE PUMPING CAPABILITY
      - TITANIUM
  - IMPELLERS
    - SSME HPFTP
    - TITANIUM
  - BEARINGS - HYDROSTATIC
    - Mk25, Mk29FD
  - TURBINE
    - TITANIUM, A-286, OR 718



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NON-NUCLEAR COMPONENT TECHNOLOGY (CONT'D.) NOZZLE AND TURBOPUMP TECHNOLOGY

Rocketdyne high-expansion-ratio, regeneratively-cooled, nozzle technology is exemplified by the SSME nozzle shown in the photo. The construction is tubular-wall, using A-286 tubes, furnace-brazed assembly, in order to reduce the weight of this large nozzle. The nozzle inlet is an area ratio of 5 with an exit area ratio of 77.5. The nozzle length is approximately 10 ft. with an exit diameter of approximately 7-1/2 ft. The nozzle employs approximately 1,000 thin wall, A-286 tubes. As with the chamber, this SSME technology provides capability beyond the requirements of the NTR, resulting in a robust design.

Rocketdyne technology applicable to the NTR turbopump draws on elements from several programs; however, is best exemplified by the Mark 29F (shown in the photo) which was developed as the liquid hydrogen turbopump for the J-2S engine. Rocketdyne initiated design and development of large, liquid-hydrogen turbopumps in 1958 under the Rover program for application to nuclear rockets. Successful testing of the first large liquid-hydrogen (Mark 9) pump in 1960 allowed commitment to the J-2 engine which used the Mark 15F (derived from Mark 9) axial, liquid-hydrogen pump. The Mark 9 and evolutionary Mark 25 turbopumps were used for 11 of the 19 reactors tested in the Rover/NERVA program and in the PlumBrook B1 NTR cold-flow engine simulation teststand.

Inducer technology for liquid-hydrogen pumps is exemplified by 2-phase testing of the Mark 15F and Mark 25 at inlet vapor volume fractions of up to 30%. Low flow-coefficient, larger diameter inducers were then fabricated for these pumps and tested to even higher vapor volume fractions. This capability provides for pumping of liquid hydrogen from a saturated tank without the need for pressurization to provide net positive suction head at the pump inlet. This provides weight savings to the stage in tankweight, pressurant and storage tankweights, and vented propellant weight.

The liquid hydrogen centrifugal pump technology of the Mark 29F was advanced with the SSME High Pressure Fuel Turbopump. Significant improvement in efficiency was achieved.
Hydrostatic bearings were demonstrated in the Mark 25 pump in testing in 1972 at NTS. These bearings used interior rolling element bearings which provided the rotation a lower speeds during the slow start-up and very slow shut-down associated with NTR's. This arrangement considerably reduces the DN requirement and life requirement for the rolling element bearing and allows use of radiation-resistant cage materials. Pure hydrostatic bearings in liquid hydrogen is an ongoing development with the Mark 29FD.

Due to the low inlet temperature (approximately 300K) and single stage of the expander cycle turbine, the turbine technology for the NTR is simplified compared to the high temperature, multi-state turbines developed for most rocket engines. Areas of concern are hydrogen embrittlement and hydriding in which Rocketdyne has much of the world's applicable experience

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50K PEWEE-DERIVED DUAL-TURBOPUMP NTR ENGINE CHANGING REACTOR SUPPORT RATIO SIGNIFICANTLY IMPROVES T/W_e

	3:1 FUEL ELEMENT/ SUPPORT RATIO	Δ WEIGHT 3 TO 6:1 RATIO	6:1 FUEL ELEMENT/ SUPPORT RATIO
REACTOR	8,200	-1,950	6,250
NOZZLE	1,200	+20	1,220
TURBOPUMP	270	+10	280
LINES AND CONTROLS	860	+60	920
SHIELD	1,100	-250	850
	11,630 LB	-2,110 LB	9,520 LB
ENGINE T/W	4.3		5.3

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50K PEWEE-DERIVED DUAL-TURBOPUMP NTR ENGINE CHANGING REACTOR SUPPORT RATIO SIGNIFICANTLY IMPROVES T/We

Between the conceptual sizing of the 50K reactor and preliminary sizing, Westinghouse determined through nuclear analysis that a 6:1 Fuel Element to Support Element Ratio could be used rather than a 3:1 ratio, resulting in higher power density and less weight for the 50K reactor. Thus, the 50K core is more similar to the 75K core, which uses a 6:1 support ratio (as used in KIWI-B4, NRX and Phoebus reactors), rather than the 25K core, which uses a 3:1 support ratio (as used in Pewee).

Elimination of virtually half the supports (with their Zirconium Hydride moderator, tie-tubes and graphite parts), together with the core and reflector diameter reduction effects, results in a reactor weight reduction of approximately 2,000 lb. or approximately 25%. The shield likewise decreases approximately 25% due to the reduction in diameter.

The non-nuclear components increase slightly (approximately 5%) in weight due to increase in pump discharge pressure to provide higher pressure ratio to drive the turbine as a result of lower turbine inlet temperature because of reducing the number, and therefore, total power of the tie-tubes (one contained in each support element) by 50%.

Due to the reduction in engine weight as the result of basically cutting the number of support elements in half (going from 3 to 6:1 Fuel Element to Support Element ratio), the engine weight is reduced by approximately 20%, and therefore, the engine thrust-to-weight ratio improves by 20%.

LIFE IMPACT ON CHAMBER TEMPERATURE AND SPECIFIC IMPULSE ADVANCEMENT

• XE-PRIME		710 SEC	
Tc 2270K BLEED CYCLE (TURBINE 10% FLOW) NOZZLE ϵ OF 10			
- REGEN COOL FUEL ELEMENT SUPPORTS AND CORE PERIPHERY		+ 35 SEC	
- INCORPORATE EXPANDER CYCLE		+ 35 SEC	
- INCREASE ϵ TO 200, 110% L		+ 50 SEC	
- INCREASE Tc WITH GRAPHITE FUEL ELEMENT TO:		2550 K	2450 K
	Life	1.5 HR	4.5 HR
	ΔI_s	+ 40 SEC	+ 20 SEC
• GRAPHITE ELEMENT ENGINE I_s		870 SEC	850 SEC
- INCREASE Tc WITH COMPOSITE ELEMENT TO:		2700 K	2550 K
	Life	1.5 HR	4.5 HR
	ΔI_s	+ 30 SEC	+ 20 SEC
• COMPOSITE ELEMENT, ENGINE I_s		900 SEC	870 SEC

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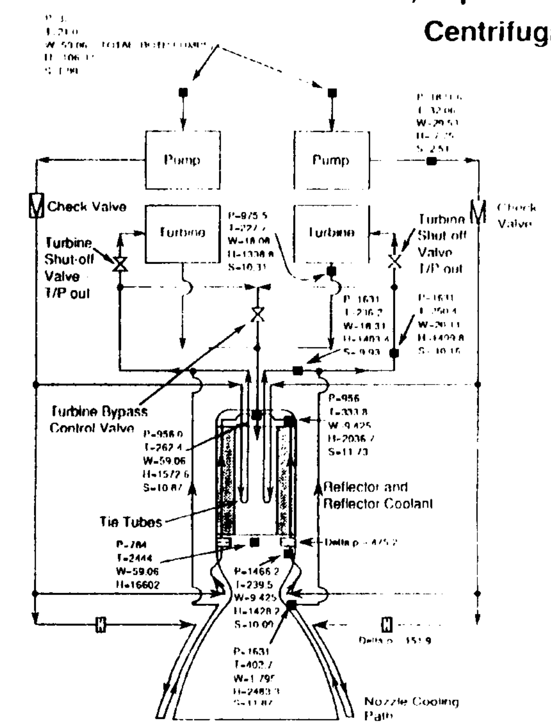
LIFE IMPACT ON CHAMBER TEMPERATURE AND SPECIFIC IMPULSE ADVANCEMENT

As in the prior chart, "Specific Impulse Advancement," the XE-Prime is the baseline for specific impulse at 710 sec. Also, as in the prior chart, advancements by 1) regenerative cooling of core structure (+35 sec), 2) using the expander cycle (+35 sec), and 3) using the 200:1 expansion ratio nozzle (+50 sec), increase specific impulse by 120 seconds.

However, Westinghouse evaluation of Rover/NERVA Fuel Element Mass Loss resulted in life capability of 1.5 hours for the prior chart's Graphite Fuel Element at Pewee Average Exit Gas Temperature of 2550K and resulting specific impulse of 870 sec, and Composite Fuel Element Gas Temperature of 2700K with specific impulse of 900 sec. This 1.5 hour data is presented in the left hand column. "Near-Term" engine life requirements are for a Life Capability of 4.5 hours. To meet the 4.5 hour Life with the allocated reactivity loss of 1$, translates into 18 to 20 grams mass loss per element and the resulting temperatures and specific impulses as shown in the right hand column. For the 4.5 hour life requirement, the resulting Graphite Element Engine Specific Impulse is 850 sec, and the Composite Element Engine Specific Impulse is 870 seconds.

50K NTR, Expander Cycle, Dual T/P*
Centrifugal Pump



DESIGN VALUES

PUMP FLOWRATE (TOTAL)	59.06 LB/SEC
PUMP DISCHARGE PRESSURE	1,022 PSIA
NUMBER OF PUMP STAGES	[illegible]
PUMP EFFICIENCY	72.58 %
TURBOPUMP RPM	47,500 RPM
TURBOPUMP POWER (EACH)	4,118 HP
TURBINE INLET TEMP	250.4 K
NUMBER OF TURBINE STAGES	1
TURBINE EFFICIENCY	71.71 %
TURBINE PRESSURE RATIO	1.67[illegible]
TURBINE FLOW RATE (EACH)	18.08 LB/SEC
REACTOR/ENGINE THERMAL POWER	1,002.8 MW
FUEL ELEMENT TRANSFERRED POWER	936.2 MW
CORE THERMAL POWER (FUEL ELEMENT+TIE TUBE)	990.7 MW
ENGINE THRUST	50,000 LBF
NOZZLE CHAMBER TEMPERATURE	2,444 K
CHAMBER PRESSURE (NOZZLE STAGNATION)	784 PSIA
NOZZLE EXPANSION AREA RATIO	200:1
NOZZLE PERCENT LENGTH	110 %
VACUUM SPECIFIC IMPULSE (DELIVERED)	846.64 SEC

Heat loads are as follows:	**Nozzle-con (total):**	**28.55 MW**
	Nozzle-div (total):	**9.44 MW**
	Reflector (total):	**12.10 MW**
	Tie-Tubes (total):	**54.50 MW**

P = PSIA
T = DEG K
W = LB/S
H = BTU/LB
S = BTU/LB-R

***Note: Flows indicated are for one-half of system.**

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50K NTR, EXPANDER CYCLE, DUAL T/P CENTRIFUGAL PUMP

In conjunction with the reduction in graphite fuel element average exit gas temperature from a nominal 2550K to 2450K to meet the 4.5 hour Life requirement, a revised system balance was performed

Compared to the balance shown on the prior chart, the average reactor exit gas temperature (nozzle chamber temperature) is reduced by approximately 4% to 2,444K. This results in an approximate 2% reduction in specific impulse to 846.64 sec. To maintain engine thrust at 50,000 lbf requires increasing flowrate by approximately 2% to 59.06 lb/sec. The increase in flowrate and reduction in temperature result in an approximate 3% reduction in Reactor/Engine Thermal Power to 1,002.8 Megawatts.

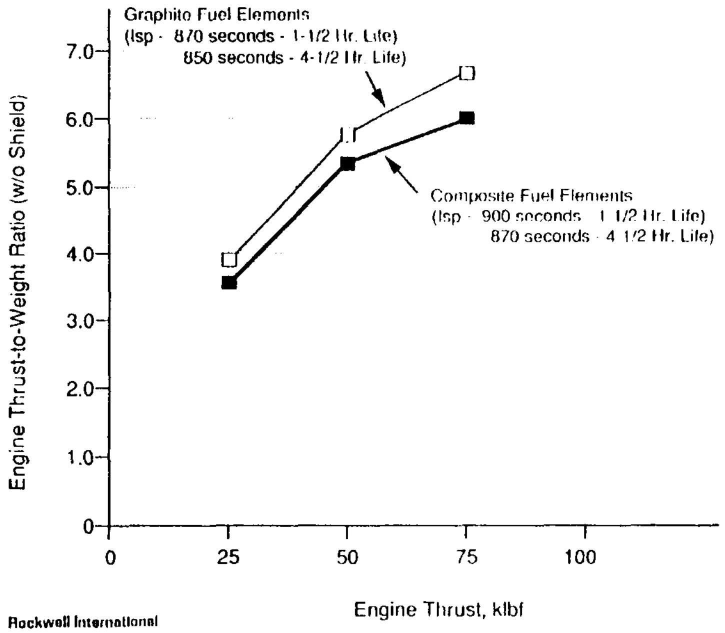
The reactor configuration for the 1.5 hr and 4.5 hr life would be the same. Fuel element thermal conditions and stresses actually reduce due to the 4% reduction in temperature and 3% reduction in power. Fuel element mechanical stresses stay the same since the reactor exit pressure is fixed (784 psia) and the core pressure drop is essentially the same due to the 2% reduction in velocity (2% increase in flowrate and 4% increase in density due to lower temperature) and 4% increase in density. So the reactor weight remains essentially the same between the 1.5 hr and 4.5 hr life cases.

The chamber and nozzle sizes remain the same, due to the 2% increase in flowrate and 4% reduction in temperature resulting in the same throat area.

The pump flowrate increases by 2% and the discharge pressure increases by 4% due to the 6% increase in turbine pressure ratio required to provide the 6% higher turbopump power. This results in a 4% increase in turbopump weight which is equivalent to approximately 0.1% in engine weight. Due to the 2% increase in flowrate and the 4% increase in pump discharge pressure, the turbopump line weight increases by 6% which is equivalent to approximately 0.4% in engine weight.

So the engine weight effect in going from the 1.5 to the 4.5 hour life is an approximate 0.5% increase in weight due to the 2% increase in flow and 4% increase in pump discharge pressure with the majority of the effect being due to the pump discharge and turbine lines.

Effect of Thrust and Fuel Element on Engine Thrust-to-Weight Ratio



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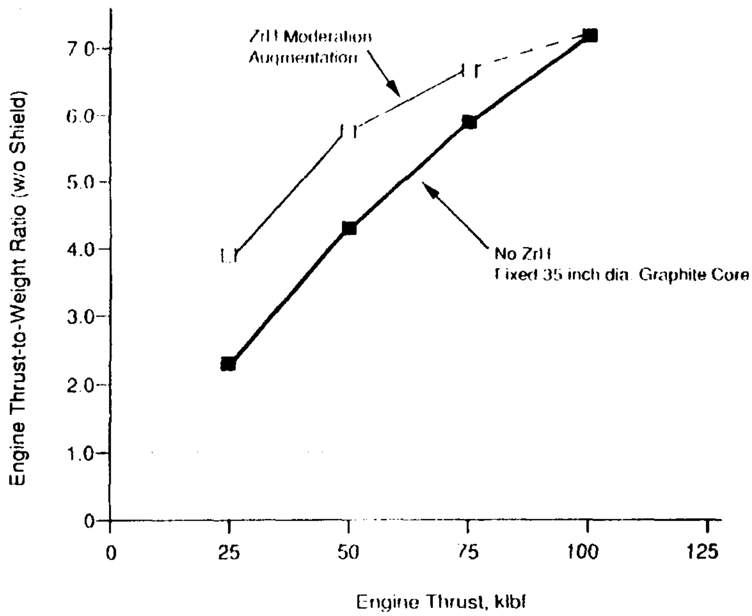
EFFECT OF THRUST AND FUEL ELEMENT ON ENGINE THRUST-TO-WEIGHT RATIO

The effect of thrust for the 25, 50 and 75K lb engines on engine thrust-to-weight ratio (without including radiation shielding) is shown for both Graphite and Composite Fuel elements.

As a result of discussion related to the previous chart regarding engine weight changes in going from 1.5 to 4.5 hr Life, the engine weight increases by approximately 0.5% primarily in line weight due to the 2% increase in flowrate and the 4% increase in pump discharge pressure. This is a negligible effect to these thrust-to-weight ratio plots. Therefore, the plot for each fuel element applies for the range of Life and Specific Impulse shown.

Effect of Thrust and Use of ZrH Moderator on Engine Thrust-to-Weight Ratio



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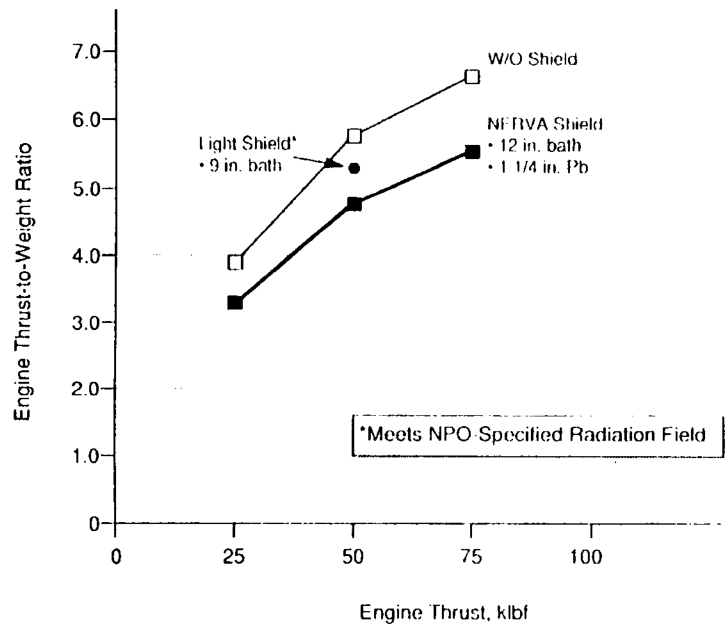
EFFECT OF THRUST AND USE OF ZrH MODERATOR ON ENGINE THRUST-TO-WEIGHT RATIO

The effect of Zirconium Hydride moderation on engine thrust-to-weight ratio (without including radiation shielding) is shown over the thrust range of 25 to 100Klb. The lower curve represents engines using a fixed 35-inch diameter, 52-inch long core containing no ZrH. The upper curve represents engines using reactors containing ZrH as necessary to minimize size and weight for the 25, 50 and 75Klb thrust reactors as analyzed by Westinghouse. Other Westinghouse preliminary analysis indicates that ZrH does not reduce the weight of a reactor with a 35-inch diameter core for a 100Klb thrust engine. On this basis, the dashed line was constructed between the 75Klb ZrH moderated point and the 100Klb point without ZrH.

So ZrH moderation provides no advantage at 100Klb thrust, approximately 10% weight advantage at 75K, approximately 35% weight advantage at 50K, and approximately 75% weight advantage at 25K thrust.

Effect of Shield on Engine Thrust-to-Weight Ratio



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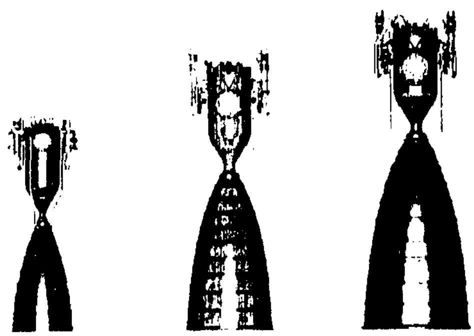
EFFECT OF SHIELD ON ENGINE THRUST-TO-WEIGHT RATIO

The effect of shield weight on engine thrust-to-weight ratio is shown. The upper data represents the engine thrust-to-weight ratio without including radiation shielding. The lower data represents engine thrust-to-weight ratio using the internal shield used for the NERVA R-1 engines; namely, 12 inches of BATH (Boron, Aluminum, and Titanium Hydride) and 1-1/4 inches of lead.

During the program, NPO specified neutron flux levels and gamma dose level to be met by the shielding for the "Near-Term" reactor. Due to concern about lead melting during decay heat removal, the lead was removed. The NPO-specified radiation field also allowed reduction in the BATH thickness from 12 inches down to 9 inches. The Westinghouse analysis of the resulting radiation field for the 50K engine results in neutron fluxes and gamma dose approximately half that specified by NPO, indicating that a small further reduction in BATH thickness may be made.

At the 50K thrust level, the Light Shield provides approximately 10% improvement in engine thrust-to-weight ratio over the NERVA shield. The light shield represents an approximate 9% reduction from the thrust-to-weight ratio of 5.8 for the unshielded 50K engine.

PEWEE - DERIVED NTR's



THRUST	25K	50K	75K
T/We (W/SHIELD)	3.6	5.3	6.0
Le	6.00M	7.66M	8.74M
De	1.73M	2.44M	2.99M

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PEWEE-DERIVED NTR'S

Conceptual designs were performed for 25, 50 and 75K thrust engines based on Rover/NERVA reactor technology. Fuel element power was based on Pewee test results of approximately *1.2 Megawatts per fuel element average.* The engine thrust-to-weight ratio requirement of >4 with shielding was not met by the 25K engine which has a value of approximately 3.6 with a shield meeting the NPO radiation field requirements. Unshielded, the 25K engine has a thrust-to-weight ratio of approximately 3.9, so the requirement of 4 is not met even without the radiation shield.

Certainly, based on engine thrust-to-weight ratio, the preferred engine thrust would be 100K or more based on the elimination of the weight penalty associated with the use of Zirconium Hydride moderator and achieving a shielded engine thrust-to-weight ratio of approximately 6.5, or approximately 7.1 without radiation shielding.

NPO selected the 50K engine for more detailed analysis and specified radiation field values to determine the shield. The radiation field requirements allowed lightening the shield by approximately 900 lb resulting in an improvement in shielded engine thrust-to-weight ratio from 4.8 to 5.3, with an unshielded thrust-to-weight ratio of 5.8.

The 50K engine has an overall length of 7.66 meters and a nozzle exit diameter of 2.44 meters.

These parameters all apply regardless of whether 1) the engine life is 1.5 hours operating at a chamber temperature of 2550K with 870 sec specific impulse, or 2) the engine life is 4.5 hours operating at a chamber temperature of 2450K with 850 sec specific impulse.

Key Technology/Streamline Development Assessment

- Approach
  - Identify critical design areas and technology issues
  - Assess actions and program impacts
  - Determine critical path
- Areas addressed
  - Safety, hydrogen pumping, nozzle, valves, instrumentation and controls, reactor, engine system and test facility

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Key Technology/Streamline Development Assessment

In performing this assessment, design areas of the engine system were reviewed and critical technology issues were identified, together with actions required to address these issued and their impact on the program. A critical path was inferred from this analysis. The design areas addressed were safety, hydrogen pumping, the nozzle, valves, instrumentation and controls, the reactor assembly, and the engine system and test facility. In most instances it was found that recovering or referencing existing technology provides the design basis. However, several system design issues exist where new design solutions and test verifications would be required, and effort to resolve these items should be emphasized early in the program.

KEY TECHNOLOGIES ASSESSMENT

AREA OF DESIGN	TECHNOLOGY ISSUE	ACTION REQUIRED	PROGRAM IMPACT
Safety	Water immersion criticality	Analysis of several design options for reactivity control in thermal range; recover NERVA plans	Engine test facility to test as-designed engine, no additional cost or schedule
Safety	Intact reentry or total dispersal	Design for reentry heating; recover reentry data; consider engine/vehicle interactions	Verification testing

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Key Technologies Assessment

Ready technology from many sources forms the foundation for the Rocketdyne-Westinghouse NERVA-derived engine concept. In reviewing key technology areas the goal was to assess both the needed actions to resolve the particular issues and the programmatic impact. In many cases technology recovery was the principal action, and there was no programmatic impact. In a few areas, issues not anchored in ready technology were identifed, and their resolution should be addressed early in the program. We expect no intractable problems.

Safety issues in all phases of the program have to be adequately identified, and procedures and design solutions have to be qualified. Four safety issues are noted here: (1) water immersion criticality, (2) intact reentry or total dispersal, (3) the concern over flammability and dispersion of nuclear materials in a launch explosion and fire, and (4) the impact of the continued nuclear power generation of a shutdown engine in a cluster. The latter affects engine-cluster specific performance, but is also a safety issue because the overheating potentially can damage the stage placing the crew at risk, and ejection of the engine with its potential for generating debris may pose a threat to the stage or to future missions.

Restartability requires adequate systems for decay heat removal that do not consume excessive quantities of hydrogen. A flight-qualified decay heat removal system was never demonstrated in the Rover/NERVA program.

KEY TECHNOLOGIES ASSESSMENT

AREA OF DESIGN	TECHNOLOGY ISSUE	ACTION REQUIRED	PROGRAM IMPACT
Safety	Launch fire resistance of nuclear and hazardous materials	Analysis and design of fire retarding or resisting features such as plug in throat	Mockup test in simulated launch explosion/fire
Safety	Engine-out continued power generation; dead weight	Analysis of alternatives: auxiliary cooling, shielding, ejection, etc.	No additional impact
Restartability	Decay heat removal	Analysis and design of optimum method for conserving propellant	Test decay heat removal system during engine tests

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KEY TECHNOLOGIES ASSESSMENT

AREA OF DESIGN	TECHNOLOGY ISSUE	ACTION REQUIRED	PROGRAM IMPACT
Hydrogen pumping	Two-phase pumping	Design pumping system compatible with tank pressurization limits; recover Rover test data, Mark 25 and Mark 15 data	Test candidate configuration in pump test facility
Hydrogen pumping	Bearings	Design for 10 restarts, and slow start and shutdown transients; recover Mark 25 data with hybrid hydrostatic bearings, SSME experience, Mark 29FD with hydrostatic bearings	Demonstrate during pump qualification test
Hydrogen pumping	Seals	Select radiation-hard seal materials	Part of turbopump design and test

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Key Technologies Assessment

The hydrogen turbopumps, while based on mostly proven hardware, must be qualified for the radiation environment, 10 restarts in space, slow startup and shutdown transients, and 4.5 hours of accumulated full-power operation. The solutions to these issues are anchored in existing technology, but a rigorous test program will be required to demonstrate, adeqately, the design integrity.

Chamber and nozzle experience with the SSME satisfies most design requirements, except those dealing with the radiation environment, such as joint seal design. Testing of seals in a radiation environment would be required.

Radiation resistance of valves--bodies, stem, guides, actuators, seals, seats--must be incorporated in the design and verified by test, and turbine bypass valve functional performance must be assured by test.

KEY TECHNOLOGIES ASSESSMENT

AREA OF DESIGN	TECHNOLOGY ISSUE	ACTION REQUIRED	PROGRAM IMPACT
Nozzle	Radiation resistant joint seal	Test seal configuration in radiation environment	Need to identify test facility
Nozzle	High heat flux	Use SSME NARloy-Z slotted channel approach	No additional impact
Valves	Radiation resistance	Select radiation resistant materials; Recover data from Rover/NERVA, SP-100, LMFBR	Life-cycle test valves separately, and evaluate after engine test

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KEY TECHNOLOGIES ASSESSMENT

AREA OF DESIGN	TECHNOLOGY ISSUE	ACTION REQUIRED	PROGRAM IMPACT
Valves	Turbine bypass control	Modify SSME and J-2 valves and perform life-cycle test	Test in hydrogen flow facility
Reactor assembly	Fuel element midband corrosion	Develop and test coating materials and processes	Identify reactor test facility; test and evaluate in engine testing
Reactor assembly	Vessel design for intact reentry or dispersal, decay heat removal	Select compatible materials and configuration	Safety requirements drive the design

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Key Technologies Assessment

In the reactor assembly, midband corrosion found in the Rover/NERVA fuel elements was being addressed when the program was canceled. Resolution of this issue will be of prime importance at the outset, with in-pile testing of fuel elements and clusters necessary to validate the solution. The reactor vessel must be designed to meet the safety requirement of intact reentry or total dispersal in the event of an inadvertant reentry from space. The control drum actuators may feature electrical or pneumatic drives, or both, based on Rover/NERVA or current design technology. The support plate must contain the tie-tube inlet and outlet flow passages, operate with minimal thermal distortion, and be structurally robust. Data from the Phoebus 2A reactor and from the NERVA design would be the bases for the new design. To achieve higher operating temperatures and performance development of composite fuel would be continued from the Rover program baseline. The instrumentation and control design area would initially address key sensors and the engine health monitoring system. Current technology would serve these areas.

Finally, the ground testing of the complete engine system is the key step in qualification for piloted operation. An operational facility will be needed with adequate engine-exhaust scrubbing to meet environmental and safety concerns, and with well-designed altitude simulation diffusers and ejectors. Because design, the environmental approval process, construction, and acceptance testing will require about 6 years to complete, embarking on this effort almost immediately is essential to meeting the desired 10-year development goal. We believe that this facility is the critical path.

KEY TECHNOLOGIES ASSESSMENT

AREA OF DESIGN	TECHNOLOGY ISSUE	ACTION REQUIRED	PROGRAM IMPACT
Reactor assembly	Drum actuators	Incorporate Rover/NERVA or SP-100 designs; evaluate fluidic stepping motors	Testing required
Reactor assembly	Support plate with tie tubes	Evaluate Phoebus-2A and NERVA-R1 designs, fabricate and test unit	Test in hydrogen flow facility--parallel with pump testing
Reactor assembly	Higher temperature fuel	Develop composite fuel	Test reactor or nuclear furnace required
Instrumentation and controls	Hydrogen flow measurement	Evaluate candidate flow meters, including fluidics; procure candidates and test	Test in hydrogen flow facility

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KEY TECHNOLOGIES ASSESSMENT

AREA OF DESIGN	TECHNOLOGY ISSUE	ACTION REQUIRED	PROGRAM IMPACT
Instrumentation and controls	Reactor temperature, pressure	Incorporate Rover/NERVA and advanced reactor sensors	Life test in a reactor in hydrogen; evaluate in engine test
Instrumentation and controls	Health monitoring system	Incorporate Rocketdyne state-of-the-art diagnostics	Evaluate in engine system test
Engine system test	Scrubbing engine exhaust, diffuser and ejector technology; environmental concerns	Proceed with site selection and facility design and construction; recover NF-1 scrubber data, state of the art Rocketdyne diffuser/ejector technology	Critical path to engine qualification

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Technology Assessment Results

- Technology available for most issues

  Rover/NERVA, SSME, Rocketdyne state of the art, SP-100, terrestrial advanced reactors, state-of-the-art electronics and computers

- Unresolved system design issues

  Loss of turbopumps, lifetime, intact reentry--water subcriticality (or total dispersal), decay heat removal, engine-out cooling during operations, fuel midband corrosion

- Critical path is engine test facility

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Technology Assessment Results

The assessment of key technologies led to the conclusions that (1) existing technology in reactors and engine systems is applicable to most design areas, (2) there are issues requiring attention early in the program to assure satisfactory resolution, and (3) the assured early availability of an engine/reactor test facility is critical to meet, successfully meet the 10-year engine qualification goal.

Streamline Development Requires Early Agreement on Requirements

NASA prepares:	Rocketdyne/Westinghouse:
• Mission and performance requirements	• Provide comments on draft requirements and specifications
• Safety requirements	• Perform QFD analysis
• Interface control structure	• Prepare design criteria and test plans
• Engine specification	• Specify test facility needs

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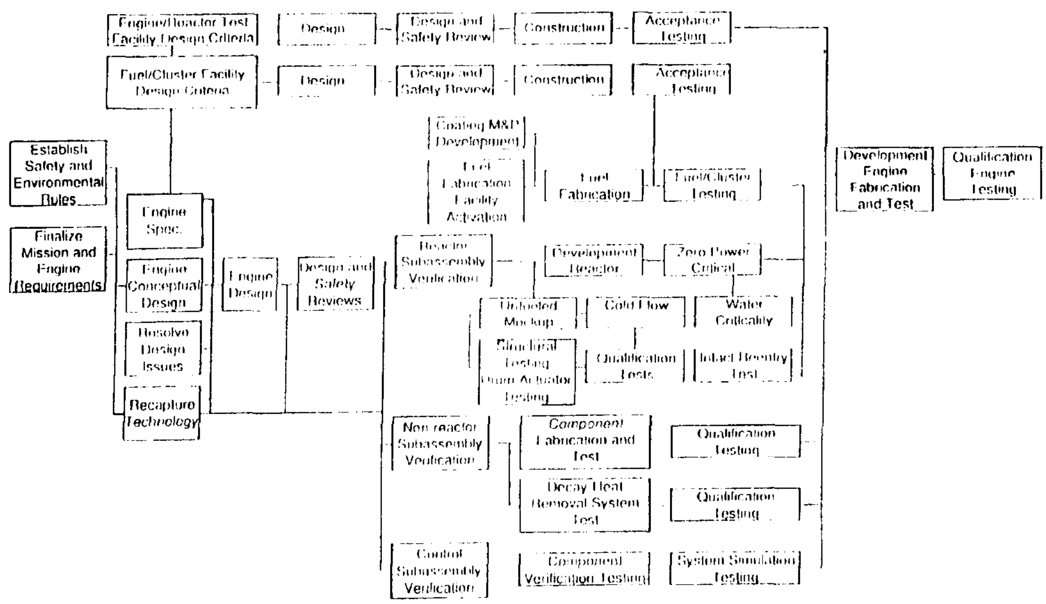
Streamline Development Requires Early Agreement on Requirements

To establish a good foundation for a successful development program both NASA and the Rocketdyne/Westinghluse team must understand and accept the design requirements, program and technical interface requirements, and design criteria and testing needs. Poorly understood or shifting requirements can lead to delays and cost escalation. We believe that a QFD analysis of the program will lead to well-understood requirements and optimum design and hardware results.

NTR Streamline Development Logic



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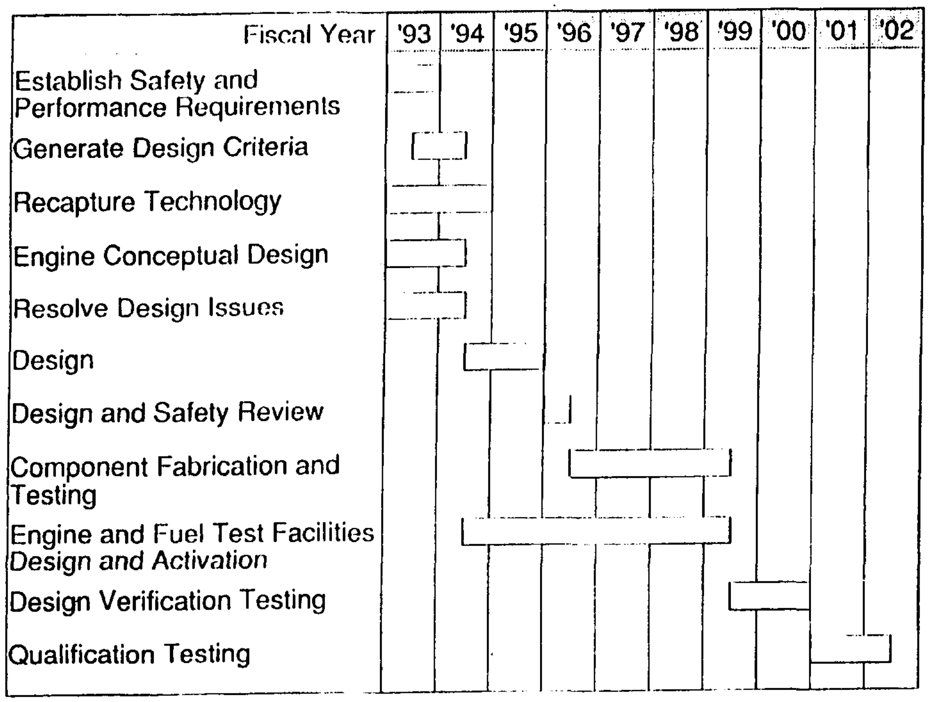
NTR Streamline Development Logic

A development logic diagram can include many layers of detail and be organized in many different ways. This high-level diagram shows many necessary tasks in setting requirements, recapturing technology, resolution of key design issues, facility design, construction, and activation, and testing of components and systems. The most important message is that the program must start with well-defined requirements and design criteria, and that the availability of key test facilities will drive the rate of achievement of the 10-year goals. Near-term activities of conceptual design, technology recovery, and resolution of design issues will provide a sound basis for proceeding quickly as substantial funding becomes available.

NTR Streamline Development Plan Summary



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NTR Streamline Development Plan Summary

The time-phasing of key groups of activities from the development logic diagram shows that several tasks should be emphasized at the start: setting requirements, technology recapture, and establishing design criteria. Test facility design, construction and activation must also begin promptly to assure that the 10 year schedule can be met.