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REON

ROCKET ENGINE OPERATIONS - NUCLEAR

REPORT NO. RN-S-0028

TO

AEC-NASA SPACE NUCLEAR PROPULSION OFFICE

REON REACTOR TASK GROUP (U)
CY 1963 ACTIVITIES

NERVA PROGRAM
CONTRACT SNP-1
DECEMBER 1963

AEROJET-GENERAL CORPORATION

AZUSA, CALIFORNIA

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January 22, 1964

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AZUSA PLANT

Space Nuclear Propulsion Office
Cleveland Extension
National Aeronautics and Space Administration
21000 Brookpark Road
Cleveland 35, Ohio

Attention: L. C. Corrington

Subject: AGC Report No. RN-S-0028, REON Reactor Task
Group Contract Year 1963 Activities

Reference: (a) SNPO-C Ltr ESB:PER dtd 18 January 1963,
Distribution of NERVA Reports

Gentlemen:

Enclosure (1) is forwarded in partial fulfillment of Subtask 1.3, Contract
SNP-1.

This report presents a summary of Contract Year 1963 activities by the
REON Reactor Task Group including some preliminary design considerations
for NRX-B. Continuing effort for Contract Year 1964 related to NRX-B
has been concluded in accordance with SNPO direction effective 1 January
1964.

Distribution of this report has been made in accordance with Reference (a),
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Very truly yours,

AEROJET-GENERAL CORPORATION

G. L. Ryland

Manager of Contracts

Rocket Engine Operations - Nuclear

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REPORT NO. RN-S-0028
TO
AEC-NASA SPACE NUCLEAR PROPULSION OFFICE
REON REACTOR TASK GROUP (U)
CY 1963 ACTIVITIES

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ROCKET ENGINE OPERATIONS - NUCLEAR

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NERVA PROGRAM DECEMBER 1963 CONTRACT SNP-1

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ABSTRACT

The REON Reactor Task Group, initially organized in early 1962, has contributed several items for consideration in the design of reactors for NERVA. These items are summarized in this report with respect to the Kiwi B-2A, B-4A, B-4D, NRX-A and NRX-B reactors. Some preliminary design considerations for NRX-B are discussed and illustrated. Continuing effort of comparable magnitude for Contract Year 1964 is planned in support of LASL and WANL developments.

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Assistant Manager
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I. ORGANIZATION AND ACTIVITIES

The REON Reactor Task Group has been, at various times, composed of REON and other Aerojet-General Corporation personnel of different disciplines allowing reactor reviews to be conducted with due consideration to nucleonics, thermodynamics, mechanics, materials and system integration. In 1962, following the Kiwi B-1B power test, the committee was organized to review the B-1B design and determine means whereby the difficulties could be corrected or avoided. The results of this study were presented in Aerojet Report No. 2280, "Conceptual Design for Cooled Bottom Support Plate for NRX-B Reactor." The reactor projects Kiwi B-6 at IASL and NRX-B at WANL were initiated to further develop and refine the concept of a cooled metal bottom support plate design for a flight qualified reactor for NERVA.

In 1963, following the Kiwi B-4A power test, the Task Group reassembled to evaluate the Kiwi B-4A and also the NRX-A which was derived from the Kiwi B-4 design. A visit was made to the Livermore Radiation Laboratory to review their work on the development of a refractory dome support plate for the Pluto reactor and their proposed modification of the Kiwi B-1 using a graphite support plate. In addition, the Kiwi B-2, which had been redesigned to correct an unstable base support condition was subjected to a fairly intensive evaluation by the committee. Other reactors, the 1 March 1963 NRX-A redesign and the Kiwi B-4D, were also subjected to a design review and failure mode analysis.

In addition to the review of existing and developing reactor designs the further refinement of an NRX-B concept was considered and recommendations made to WANL to aid their effort on this project. Some details of the results of these 1963 activities of the REON Reactor Task Group are presented in the following sections.

II. KIWI B-2A EVALUATION

A. KIWI B-2A DESCRIPTION

The Kiwi B-2A is a graphite core, beryllium reflected reactor wherein the core is supported at the hot end of the subassembly and loaded in compression by spring members in a seal plate at the cold end. The core consists of structural graphite in the form of a disc at the hot end and modules from the seal plate to the disc. Fuel elements within the graphite structure have been designed to provide a power capability of 1000 Mw. Propellant (coolant) flow is directed to the fuel elements through seal members in the cold end-plate such that the core assembly is essentially surrounded by chamber pressure thus relieving the pressure-drop load on the disc. The modules, between the seal plate and the support disc, are pin-supported on each end. The disc is supported by a rocker assembly designed to distribute the loading evenly to each member. Radial loads on the support system are compensated by reflector entry coolant pressure such that radial compressive loading of the graphite disc minimizes the tensile loads within the disc.

Figures 1 and 2 illustrate the Kiwi B-2A core and core support features which were evaluated in the design review effort. The reflector, pressure vessel and nozzle which were common to other Kiwi-B reactor designs were not considered. Additional details on the Kiwi B-2A design are available in the LASL "Kiwi B-2A Design Review" package prepared for the June 4-7, 1963 design review meeting.

B. KIWI B-2A EVALUATION

The Kiwi B-2A Evaluation involved a check of all core assembly component drawings and a failure mode analysis, thermal analyses of potential operational conditions, and thermal and mechanical stress analyses. This evaluation study will be presented in Aerojet Report No. 2683, "Kiwi B-2A Reactor Design Review. The drawing review and failure mode analysis results are presented systematically for each assembly, subassembly and part of the Kiwi B-2A reactor. As an example the "Rocker," shown in Figure 1, is reproduced here.

<u>PART/ASSEMBLY NUMBER</u>	43Y65074-C227	Support Rocker
1. <u>Next Assembly</u>	43Y65074-E300	Core Reflector Cylinder Assembly

2. Mating Parts

43Y65074-C220	Rocker Socket No. 1
43Y65074-C221	Rocker Socket No. 2
43Y65074-C225	Rocker Socket No. 3
43Y65074-C229	Rocker Support Sub Assembly No. 2
43Y65074-C228	Rocker Support Sub Assembly No. 1

3. Material

National carbon - grade ADJ graphite

4. Function - Description

a. Non-Operational

Transmits core spring load from C220, C221 and C225 rocker sockets to C229 and C228 rocker support sub-assembly.

b. Transient

Pressure drop across core in addition to core spring load is transmitted from C220, C221 and C225 rocker sockets to C229 and C228 rocker support sub-assemblies.

c. Design Operation

Same as 2.

d. Shut-down and Cool-down

Same as 2.

5. Modes of Failure

a. Thermal Stress

Dependent on coolant flow.

b. Bearing Stress

$$S_b = \frac{824}{0.82} = 1,000 \text{ psi}$$

c. Recommended Test

See 43Y65074-C220 rocker socket.

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7. Conclusion

Dependent on test results.

Suggest design changes to obtain a uniform cooling of the rocker.

Thermal analyses of the Kiwi B-2 core support and insulation were performed utilizing two and three dimensions for various parameters involving the possible range of leakage flow and bearing loads associated with the rocker support system. Figure 3 identifies the model network used for these analyses. A sample of one of the thirteen rocker support thermal and flow conditions analyzed is shown in Figure 4.

A thermal stress analysis of the B-2A support disc was performed utilizing an independent thermal analysis of the disc axial temperature gradient. The thermal gradients were calculated to be more severe than given by IASL as illustrated in Figure 5. The detailed thermal stress analysis, though, indicated a less severe condition than had been calculated by IASL. The maximum radial tensile stress, including mechanical loading, was determined to be 550 psi compared with 1010 psi given by IASL. Comparative compressive stress maxima were 800 psi vs 70 psi given by IASL. The stress distribution is illustrated in Figure 6.

Results of the Kiwi B-2A Cold Flow Test during July 1963 were reviewed with particular attention given to the module fractures which occurred. This evaluation was presented in Aerojet Report No. 2719, "Evaluation of Kiwi B-2A Module Fractures During Cold Flow." It was tentatively concluded that air entrapment and later mixing with hydrogen resulted in small explosions which forced the flow sleeves into the module thus creating the longitudinal module fractures found on disassembly. Although it was not possible to draw firm conclusions from the study without model tests to check this failure mode theory, some general recommendations to avoid air entrapment in the reactor assembly were made which are important to any Kiwi or NERVA reactor design.

a. A thorough purge of air from the reactor is required. Each reactor assembly design should be reviewed to determine the probability of air

entrapment during purging. The purge procedure, reactor design, or both should be adjusted to insure no air entrapment.

b. A flow separator should be attached to the divergent section of the nozzle. Since the nozzle does not flow full during cold flow testing, air can condense on the nozzle surface and flow into the pressure chamber. A flow separator would catch this flow and separate it from the hydrogen. Insulation on the interior surface of the nozzle divergent section is also recommended to decrease the rate of liquid air formation.

c. A nozzle closure plate, which blows off during flow and then closes at shutdown should be investigated. This closure would permit evacuation of air within the nozzle and pressure vessel prior to purging. The technique of air evacuation might be applicable for reactor designs where inert gas purging is incapable of purging all plena and cavities.

C. KIWI B-2A CONCLUSIONS

It was generally concluded that the Kiwi B-2A design, with only minor modifications, would be suitable for NERVA engine development purposes. Although the growth capability with B-1 fuel elements is limited, the potential of being able to continue the NERVA engine development on an early schedule would be very good. The modifications desired for Kiwi B-2 hot power tests include such items as perforated rockers for distributed coolant flow, positive attachment of the reflector cylinder to the hoop and seal plate, improved fuel element retention, a positive purge flow for the core surrounding region, and some means of relieving potential shock loads upon the module pieces. A component development testing program would be required in conjunction with the hot power testing to insure qualification of the reactor for engine integration.

III. KIWI B-4A EVALUATION

A. KIWI B-4A DESCRIPTION

The Kiwi B-4A mechanical and materials description is well known to those for whom this document is written. For review and reference with respect to the following test, the essential features are identified here.

1. Tie rod supported fuel clusters with hot end graphite support blocks and cold end cluster plates providing some fixity to seven hexagonal elements of each cluster.

2. Cold-end seal at the core periphery to minimize bypass flow outside of the fuel elements.

3. Light bundling of core assembly by means of spring loaded plungers working through a graphite barrel bearing against insulation slats on the full length core cylindrical surface.

B. KIWI B-4A EVALUATION

It was hypothesized, and later verified by the LASL "PIE" experiment, and the Kiwi B-4A Cold Flow experiment, that leakage flow between core elements forced the clusters apart. The internal pressure was greater than the peripheral pressure, because of the cold end seal, which allowed outward radial flow of the leakage coolant. With the irregular and variable radial flow path, this internal pressure forced the fuel clusters to vibrate and impact on each other sufficiently to break the graphite elements into a large number of pieces.

In the process of developing the hypothesis mentioned above, several aspects of the Kiwi B-4A design were investigated. Aerojet Report REON 746-10 "Dynamic Analysis of the Kiwi B-4A Reactor Core Structure" was prepared. The following abstract summarizes the contents of this report.

REON 746-10 ABSTRACT

A study is made concerning the dynamic behavior of the Kiwi B-4 reactor core elements. Free vibrations, and the effects of forcing functions arising from fluctuating pressure systems within the core complex, are investigated. The effects on dynamic response, as a result of mass flow

rates and gas velocity through the element, coupled with a variation in boundary restraints, are defined. The effects on the natural frequency imposed by a variation of axial load and boundary restraints, are determined singularly and in combination. Transient and steady-state environmental conditions are investigated, and a failure analysis is conducted on the fuel-element cluster assembly.

A statistical analysis of the fit-up of 2000 hexagonal elements in the reactor core assembly was made considering the dimensional and twist tolerances of the elements. These data are reported in Memoranda 736:0167 and 736:0184. Other minor aspects, with the viewpoint of design modifications for correcting the difficulties, were also investigated. However, the primary effort of WANL in redesigning the NRX-A was not duplicated to any significant extent.

In the statistical analysis of the core element fit up, it was determined that the probability of having gaps within the core greater than 0.008-in. was 4%. Translating this in terms of flow area between elements indicates an expected gap flow area of 0.15% of the fuel element flow area. Fluid flow analyses were considered for these flow gaps; however, the complexity and limited value of such work, considering the actual dynamics involved, precluded any effort along this line. Assuming adequate bundling forces, as in Kiwi B-4D to prevent cluster separation, or peripheral pressure matching as in the NRX-A redesign, the leakage flow is more readily estimated.

Leakage in a mechanically stable, hot reactor was concluded to be of concern with respect to corrosion of the fuel element exterior surfaces. An approximate calculation, from the statistical fit-up analysis indicates a leakage greater than 0.1 lb/sec which, for NERVA, would amount to 120 lbs or more of hydrogen. At 3 mol percent methane equilibrium, this would result in a loss of over 2 lb of carbon from the exposed fuel element surfaces. A 4% probability for gaps greater than 0.008-in. corresponds to 2000 in.² of exposed graphite. An average of 0.014-in. corrosion loss would therefore result. With the variability of corrosion and flow channels, it can certainly be expected that several fuel elements would have corroded into the outer channels.

It should be noted that the excellent fit up of fuel elements in the NRX-A1 assembly (Oak Ridge supplied elements) is promising for a reduction in this leakage effect. At worst approximately 1% of the fuel elements might suffer exterior corrosion damage in a 20 minute power run.

C. KIWI B-4A REDESIGN CONSIDERATIONS

Considerations for redesign of the Kiwi B-4A were directed towards means of avoiding damaging impact of fuel elements with each other and minimizing leakage flow between elements. The redesign features actually incorporated into the Kiwi B-4D and the NRX-A are not covered here.

1. Hot End Sealing

Sealing between clusters by modifying the support blocks to contain piston rings bearing against "triangular" cusp pieces was considered. This concept is illustrated in Figure 7, redrawn from REON memoranda 730:0268 and 730:0274. This would provide, in addition to the seal effect, an intercluster growth region such that the core would remain at a constant diameter allowing a more effective seal.

This concept is complex with respect to the number of pieces involved and the assembly. Also it leaves the fuel elements loose within the core assembly.

2. Fuel Element Cementing

The cementing of core clusters to form one, rather than seven, piece clusters was considered as a means of forming a module with sufficient rigidity to prevent damaging impact loads and allow greater axial loading without buckling. Also, by cementing elements to the unfueled center piece it should be possible to omit the base support blocks.

This concept still requires core bundling and in addition requires the development of graphite cementing processes. To avoid the bundling it would be necessary to make a much larger cluster and also to fix each end of the cluster between cold and hot end support plates. (Reference the NRX-B program for these aspects.) The NRX-A schedule precludes such a design modification being incorporated.

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3. Forced Bridging Paths

The variation in core element dimensions allows bridging within the core, thus leaving some elements loose while others carry a magnified bundling pressure. A suggestion for forced bridging paths by means of alternating oversize and regular size sections of each fuel element. Four inch long oversize sections might be created by depositing a .002 in. pyro coating on the elements with a 9 in. gap between oversize sections. By axial variation in the oversize station, the entire core assembly is insured of being bridged over at least four 4-inch sections along the length of the core.

Fabrication and assembly of core elements would obviously be complicated by this design concept. This fact plus the better fuel tolerances attainable and the improved bundling system design eliminated the concept from detailed investigation.

4. Core Peripheral Insulation

Pyrolytic foil built up into full, core-length, pieces matching the irregular core periphery are felt to be a very promising substitute for the small pyrolytic tiles now planned for both Kiwi B-4D and NRX-A. These blocks, made up of cemented pyrolytic foil, would have about 2 to 3 times the conductivity of regular pyro graphite. However, they would also be made 2 to 3 times as thick, forming a stable single piece body.

This insulation material requires development and testing for the suggested application. Nothing has been done with it on the NERVA program yet, but the promising simplification of certain Kiwi and NRX configurations remains.

IV. KIWI B-4D REVIEW GROUP

A. REVIEW GROUP ORGANIZATION

The Kiwi B-4D Review Group was organized in August 1963, at the request of LASL, to review the Kiwi B-4D design.

The first meeting between the Kiwi B-4D Review Group and LASL personnel took place on September 4-6, 1963. The purpose of this meeting was to:

1. Gain familiarity with the B-4D design.
2. Identify potential problem areas for REON review and evaluation.

B. TENTATIVE REVIEW PROGRAM

The tentative review program adapted by REON, as a result of this meeting, is presented in Memorandum 771:RS:057. In summary, the program consisted of the following elements:

1. Drawing Review
 - a. Cold Flow Temperature Conditions
 - b. Hot Flow Temperature Conditions
 - c. Basic Parts List
2. Analysis Checks
 - a. Thermal Analysis
 - (1) Lateral Support System
 - (2) Transient Analysis
 - b. Stress Analysis
 - (1) Support rods - notch sensitivity
 - (2) Core frictional lockup
 - (3) Peripheral support rods - unequal loading
 - c. Leakage Flows
3. Material Properties Review
4. Comparison of Kiwi B-4D and NRX-A

C. PROGRAM RESULTS

A tentative report outline, and responsibility assignment for various report sections, were prepared and presented in Memorandum 771:RS:059.

As the review progressed, changes were made in the original program. The leaf spring was stress analyzed, omega seal development was reviewed, and the planned cold flow instrumentation was reviewed. Thermal transient analysis and the B-4D-NRX-A comparison were postponed, because of time and shifting manpower requirements.

A preliminary evaluation of the B-4D design was presented to IASL on October 3, 1963. Memorandum 771:RS:065 summarizes this presentation. The major points made in this presentation are given below.

1. Support Rod and Leaf Spring Materials

Avoid the use of 17-7 PH stainless steel for support rods and lateral support springs in the cold flow test. This material loses ductility at cryogenic temperatures and suffers brittle impact fracture. The use of Inconel 718 is recommended for these components.

2. Purging

A thorough nitrogen purge, preferably preceded by air evacuation prior to cold flow testing, is recommended. These steps are required to minimize the possibility of a reaction between hydrogen and liquid air.

3. Liquid Hydrogen in Core

Terminate the cold flow run early enough to avoid the flow of liquid hydrogen into the core. Changing material characteristics and differential thermal expansion can cause problems at liquid hydrogen temperatures which would not be experienced in a hot firing.

4. Thermal Analysis

Thermal analysis of the core periphery system confirms IASL analysis that this area is adequately cooled during a hot firing.

5. Support Rod Design

A rolled upset thread on the support rod to improve strength and notch sensitivity characteristics is recommended.

6. Drawing Check

The drawing check at ambient temperature conditions does not reveal any serious problems. Numerous changes on tolerance specifications are suggested to reduce stringent machining and quality control requirements.

7. Leaf Spring Design

Relocate the leaf spring guide holes to the extreme sides of the spring to eliminate local yielding during a hot firing.

8. Pressure Bundling Force

REON concurs with IASL that little core damage can be expected from differential fuel element thermal expansion coupled with a high bundling force.

9. Perimeter Fillers

It is recommended that relief be given to axial motion of the perimeter filler lugs. These lugs are presently locked in place between the reflector cylinder and the transition ring. They can be expected to move axially during cold flow and hot firing tests.

10. Omega Seals

The omega seals should be tested for seal surfaces and environmental conditions approaching those of cold flow and hot firing testing.

11. Instrumentation

An acoustic, differential-pressure transducer in the plenum surrounding the core wrapper during the cold flow test should be included. This transducer, coupled with similar transducers planned for the entrance plenum to the support plate and for the pressure chamber, should cover areas most likely to incur a hydrogen-liquid air reaction.

Subsequently to this presentation, the drawing check was completed for temperature conditions at a uniform 50°R and at operating conditions. The most significant finding was the possibility that the stainless steel tube surrounding the support rod would thermally expand toward the support cone, thus blocking the cone flow penetrations and cutting off coolant flow. REON recommended either shortening the tube 0.25 inches, or reversing its orientation, with the flare seat placed between the support cone and the pyrographite washer. This recommendation was transmitted to LASL by telephone on November 5, 1963.

Presently in progress is an analysis of leakage flows between adjacent perimeter fillers at operating temperature conditions. Also being completed is an analysis of the heat conducted axially by the pyrographite insulating tile, and subsequent dumping of this heat into the hot end of the perimeter filler in the vicinity of the brazed coolant tube orifice.

The Basic Parts List has been completed and is current as of September 27, 1963.

The checked and marked drawings are being assembled for transmittal to LASL. Upon completion of the thermal analysis in progress, a summary report of the B-4D review will be prepared.

V. NRX-B DESIGN STUDIES

A. NRX-B DESIGN BASIS

The outstanding consideration of the REON Reactor Task Group in its review efforts has been to arrive at a new reactor design configuration which would not only resolve those problems encountered by Kiwi B-1 and B-4 but also those anticipated in meeting the NERVA performance requirements. Of particular concern has been the ease of fabrication and assembly of components. Very fine tolerance specifications may well resolve the problems of mechanical and thermal operation but they can also eliminate many potential suppliers (perhaps all) from providing the required hardware. Even though the tolerance limits are met in fabrication, the careful handling required during assembly can be a further problem. Finally, the lifetime requirements of the assembly must be considered. The question must be asked whether fine tolerances can be maintained during operation involving mechanical loading, vibration, shock, thermal cycling, hydrogen environment, extreme radiation and vacuum.

The following ground rules were established to reduce the scope of reactor design configurations to something compatible with the NERVA program.

1. Retain Kiwi B reactor dimensions
2. Retain Kiwi B control drum system
3. Retain the hexagonal, 19-hole, extruded fuel element as the basic building block.
4. All major loads introduced into the graphite components of the core assembly shall be compression rather than tension.
5. Minimize variation of the nozzle entry gas temperature from the core.
6. The design and materials will be compatible with the environmental conditions that will exist for the core.

The first two ground rules imply that the nucleonics and controls will be essentially the same, although it is reasonable and desirable to increase the beryllium region at the expense of the graphite inner reflector, thus gaining on control worth.

The third ground rule is necessary since the fabrication techniques and safety requirements have been established for the hexagonal, 19-hole, extruded fuel element. Should another design be considered for a fuel element, it would involve additional time and cost. The 19-hole fuel element may be used to make a larger element, as by bonding.

The next ground rule is due to the difference in tensile and compressive strengths of graphite. As a comparison, the compressive strength is approximately five times the tensile strength. Also, the deviation in the tensile strength may be significantly broad so as to interject some concern in the reliability aspect of the core assembly.

The fifth ground rule implies that leakage flow is zero, or at least minimal, and that the flow meters or orifices are sized properly. By reducing the leakage flow, the corrosion and thermal stresses are minimized and performance is increased.

The last ground rule is essentially a reiteration of the basic engineering considerations of any high performance, reliable, system.

B. POSSIBLE DESIGN VARIATIONS

The primary investigations have been centered on the design of the core with respect to the fuel element, flow control, and mechanical support for the fuel and core assemblies. A summary of possible fundamental design variations is given in Table I.

TABLE I				
Axial Support	Lateral Support	Core Subassembly	Flow Control	
			Primary	Leakage
1. Cooled Metal Bottom-Support (Kiwi B-6)	1. Metal Girdle	1. Monolith	1. Orificing	1. Seals
2. Refractory Uncooled Bottom Dome (Pluto)	2. Pressure Pistons	2. Modules	2. Channel Sizing	2. Controlled Leaks
3. Tie Rod-Bottom Supported (Kiwi B-4)	3. Slats	3. Individual Elements		
4. Tie Rod-Top Supported (Kiwi B-1 and Kiwi B-3)	4. Springs			
5. Side Shear Supported (Kiwi B-2)	5. Graphite felt Packing			
	6. Cantilever			
	7. Pinned at Both Ends			

The various concepts as described above may be combined to give many designs. The object is to determine the best combination which produces an optimum design and then apply these results to the NRX-B program.

1. Consider the most straight forward core design, one which consists of a minimum number of parts or components, and a simple support system. The design consists of a monolith core, hot bottom support plate, top seal, and top load springs. The core is supported axially by the top load springs and the bottom support plate. This load puts the core in compression and the pressure vessel or an inner containment barrel in tension. Since the longitudinal stress on the vessel for internal pressure is one-half that of hoop stress, it is apparent that a large compression load may be applied to the core to hold it in place without exceeding the pressure-vessel design capability. The lateral loads are taken out at each end of the core since the monolith has a high rigidity. This design gives a minimum of possible leakage flow paths.

Outstanding characteristics of the monolith core design are:

- a. No cold gas leakage flow directly onto hot components.
- b. Rigid core
- c. No low temperature material in core which requires cooling
- d. Simple one piece support plate
- e. Simple lateral support (no lateral load required through insulation)

The obvious disadvantages of this design relate to fabrication difficulty, safety, thermal stresses, etc. Thermal stresses were considered as the most immediate problem. Using the average axial material temperature distribution as shown in Figure 8 and assuming that the radial distribution is uniform at any station, it was found that high thermal stresses could occur in the region of maximum thermal gradient differential. The highest stress of 5000 psi was at the hot end. By modification of the axial power distribution it would be possible to obtain a linear axial temperature gradient and thus zero thermal stresses throughout the monolith core. This, though, would require a varied axial fuel loading. Also transient thermal conditions, which have not been analyzed, could be troublesome. Further consideration of the monolith core was therefore eliminated.

The monolith thermal stress is due to the bending of peripheral elements caused by the second derivative of temperature. If the fuel body diameter were reduced to less than nine inches, the thermal stress caused by the axial temperature would be less than 500 psi. This leads the design toward a modular core concept wherein the modules are as large as practical for simplifying the supporting structure. The supporting structure might still be any of the concepts indicated in Table I, but in consideration of simplicity, pin-ended modules supported between two plates appear most attractive.

2. The other support concepts, while they might be made to work, have several disadvantages. These are briefly summarized in the following:

a. Tie Rod-Bottom Support

Tie rods extending through the core assembly require insulation and separate cooling for the metal support. This is a complexity and a performance loss factor either by mixing cold propellant with the hot chamber gases or an increased pumping pressure requirement if the rods are to be cooled in series flow ahead of the fuel. Also the thermal problems of cooldown operation are amplified.

b. Tie Rod-Top Support

This involves tensile support of the graphite.

c. Side Shear Support

The Kiwi B-2 is interesting in that central tensile loads are reduced by the lateral compression. However, the structures are complex.

d. Bundled Core Lateral Support

In general the lateral support concepts are complex with respect to the number of parts, and they involve coolant flow requirements which influence the leakage into the core. The bundled core concepts are under development and therefore consideration of such by the REON Reactor Task Group for the new reactor design would be redundant.

If a bundled core design were to be combined with a cooled metal bottom support plate, additional complexities are introduced because of

differential expansion of the core and plate. With respect to a graphite support plate, which would expand with the core, a new problem of differential expansion between the plate and metal support core is introduced.

e. Graphite Felt Packing

The bundled core arguments generally apply to metal girdles, pressure pistons, slats and spring systems. The graphite felt packing, though, does offer the possibility of avoiding the leakage flow problem. A combination of functions, including insulation and resilient structure, is available using graphite felt. Wool and cloth might also be adaptable. With development, the bundled core concept, using a graphite resilient material for lateral support and a graphite plate for base support, could be attractive. These development problems were felt to involve more time than should be devoted to a new reactor design. Further effort on this concept may be applied if there is a continuing interest.

f. Cantilever Support

Cantilever support of fuel elements, such that lateral support can be omitted, requires a sufficiently small length-to-diameter ratio as to preclude vibration amplitudes which would impact adjacent fuel elements. Also the one supported end must be fixed such that rotation, which also allows impact, can be avoided. The mechanical features required for this concept appear excessively difficult except possibly in combination with some lateral support.

C. REON NRX-B BASE-SUPPORTED MODULE DESIGN EFFORT

1. Cemented Modules

Cemented modules consisting of a large number of extruded B-4 fuel elements have some problems associated with their production and use. The production problems appear to be solvable as indicated by preliminary work done at LASL, but the practicality for use in nuclear rocket engines has been questioned. The variability of temperature within the core caused by fuel distribution, flux softening near the reflector, orificing tolerance control and transient operation have raised the fear of excessive thermal stress. Also, the variability of the coefficient of thermal expansion between different pieces of graphite could impose even more severe thermal stresses on the bond between elements.

In evaluation of the influence of the variable power density and heat removal tolerance on fuel modules, a 61 element hexagonal cluster was modeled for analysis. This analysis is presented in Memorandum 734:MO358. Each element was given a heat generation - coolant flow parameter selected on a statistical basis. Computer cards for each element were literally dealt in a random distribution so as to get various temperature distributions. Six different cases were run on a two-dimensional nodal network computer model as shown in Table 2. These cases are shown with the results in terms of maximum and minimum temperatures. Figure 9 is an isotherm map of case No. 4, a $\pm 5\%$ distribution, and Figure 10 is for case No. 6, the stacked deck distribution of $\pm 5\%$ cards.

A thermal stress analysis of case No. 4, done by considering the axial distribution shown in Figure 8, to be applicable to each gas exit temperature, indicated maxima of +874 and -830 psi. Thermal stress analyses were done for case No. 6 also assuming freedom of the module to bow. The results were +936 and -1390 psi. Deflections of this case were 0.025 in. across the flats and 0.015 in. across the corners of the hexagonal assembly.

It is apparent that the control of temperatures by means of accurate fuel loading specifications, measurement of power density in critical experiments and final shimming by means of orificing is necessary. The $\pm 5\%$ random distribution limit appears to be attainable and is not excessive with respect to thermal stresses. Combined with other stress loads, though, it would be desirable to reduce the temperature differentials. A goal of $\pm 100^\circ\text{R}$ (as contrasted to $\pm 200^\circ\text{R}$ for the $\pm 5\%$ case) might be attained through actual reactor power tests and improved measurement techniques.

The problem of the variability of graphite coefficient of thermal expansion appears to be more serious than that of power-temperature control. Variations of $\pm 20\%$ (3 σ) have been measured. Even with perfectly uniform temperatures among elements of a cemented fuel module, the thermal stresses could be excessive by a factor of two over the minimum shear strength of graphite bonding materials.

There are some extenuating factors associated with the thermal expansion coefficient variability. It has been shown to exist within a single fuel element and, in unfueled graphite it has been shown to be variable from one

TABLE 2

61 ELEMENT MODULE THERMAL ANALYSES

<u>Problem Set No.'s</u>	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	<u>6</u>
Power Distribution						
<u>+10%</u>		1	1			
<u>+ 9%</u>						
<u>+ 8%</u>		1	1			
<u>+ 7%</u>						
<u>+ 6%</u>		2	2			
<u>+ 5%</u>	30,31	2	2	1	1	1
<u>+ 4%</u>		3	3	2	2	2
<u>+ 3%</u>		5	5	5	5	5
<u>+ 2%</u>		6	6	7	7	7
<u>+ 1%</u>		7	7	10	10	10
0		7	7	11	11	11
T _{max} gas	4330	4512	4503	4268	4298	4324
T _{min} gas	3900	3722	3803	3909	3938	3899
ΔT_{max} gas between adjacent nodes	367	554	399	263	298	58

[REDACTED]

measurement to another in the same sample. The question can then be asked as to why a single fuel element does not break up when heated to reactor operating temperatures. Also, thermal stress calculations for differential temperatures indicate breakage should take place whereas experimental evidence does not confirm analyses. There is at least a factor of two discrepancy between analysis and experiment.

It appears that graphite has some unusual characteristics, possibly associated with, but more complex, than plastic relief of stress loads. The variability of thermal expansion coefficient, a possible variability in the modulus of elasticity and a seeming adjustment to excessive stresses imposed in a particular manner should receive considerable basic research attention. For the objectives of the NERVA program, though, empirical methods of establishing graphite material capabilities and limitations should be used. The development testing of cemented fuel modules must include environmental (thermal) simulation both including and exceeding the conditions expected in service. The excess is necessary to establish statistical data suitable for evaluating, by nondestructive means, the acceptance of modules for reactor service.

2. Module Transition Members

The support of fuel modules between two support plates requires some means of collecting and directing the flow of gas through the support members and also taking up the stressing loads imposed on the module. An early thought along this line was to form a graphite dome structure to be supported by a cooled metal support plate. The initial design of such a transition member, illustrated in Figure 11, was fabricated in lucite for experimental use in comparing stress analysis theory with measured data. The result of this experimental work is summarized below from "An Analytical Procedure for the Structural Analysis of the Reactor Core Base Adapter" a REON Technical Staff Department 746 Engineering Memorandum dated July 1963.

A procedure was established for the structural analysis of perforated circular plates which are utilized in supporting the reactor core structure. The method consists of replacing the perforated plate by an imaginary homogeneous plate which has modified material mechanical properties, E^* (modulus of elasticity)

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and u^* (Poisson's ratio). The stresses developed in the ligaments between the perforations in the actual plate, were then evaluated readily in terms of the stresses calculated for the homogeneous plate. In the case of the core base-adapter, this procedure was further modified to account for curvature effects.

In order to verify the validity of the structural analysis procedure, the core base-adapter was subjected to a structural load test which simulated actual environmental conditions. The results compared very well with the analytical data, thus making the procedure available for calculating stress and deflections in similar designs.

After the above adapter piece effort was initiated, other simpler designs were suggested. These involved single element transition pieces on each module peripheral fuel element such that a complete module would appear as illustrated in Figure 12. This concept of transition pieces on module peripheral elements alone is suitable for modules up to 61 elements. With 91 elements, the bearing surface on the support plate is quite small compared with the pressure-drop load. Also the side loads imposed by shock and vibration become marginal with the larger module. For the 61 element module, though, the analysis indicated maximum stresses of 1600 psi compression.

3. Cooled Metal Support Plate

The Kiwi B-6 and the NRX-B programs resulted in two base-plate designs during the latter part of 1962. The B-6 base plate was to be a fabricated assembly of A286 members brazed together to form the structure illustrated in Figure 13a. The WANL experience with this design, with respect to fabrication, involved many difficulties with respect to brazing and other tooling required. An alternate design, by WANL, involved machining of the primary structure from a single forging. The forged plate design is illustrated in Figure 13b.

In review of the forged plate design, prior to approving the completion of the plate for developmental testing, several questions were answered of the REON Materials and Manufacturing Engineering personnel in Memorandum 746:B.10:422. These involved procurement of acceptable forgings, machining operations, machining error repairs and machining cost. The consensus was that the forged base plate would be a difficult but practical design although not

available from any typical machining shop. REON's manufacturing engineering estimate of 50 to 60 thousand dollars compares with the low bid of \$65,000 for this work.

Further consideration of both the fabricated and forged plate designs will be given after experience in both manufacture and testing is completed in Contract Year 1964 on the NRX-B program. In the meanwhile REON considers a modified forged plate concept to be more promising. The most significant modification involves utilizing larger fuel clusters such that the base plate grid is more coarse, thus reducing the chance for machining error. Also it would be possible to achieve a greater probability of acceptance of forgings since small imperfections may be located such that they are removed in the coarse machining requirement.

4. NRX-B Thermal and Nuclear Design

A preliminary NERVA Design Criteria for NRX-B, NDC-64, was prepared in March 1963 for review and guidance prior to initiating effort on the NRX-B redirection. This document was withheld from distribution when the NRX-B redirection allowed for a much broader range of reactor conceptual design than was being considered by REON.

The NRX-B design criteria included most of the ground rule and philosophy factors indicated in the text above plus the specific nuclear requirement of increased control worth over that of the Kiwi B and NRX-A reactors. This requirement was specified in terms of a 60-sec time limit to shut down the reactor from design point power to 2% fission power (in comparison to fission product decay power which would be about 3% in the same time interval). The object of this specification form, rather than as a shutdown reactivity, is to identify the objective rather than any particular parameter.

Thermal design considerations for NRX-B were suggested apart from the primary heat exchanger for the main propellant flow. The NERVA engine as a hot bleed cycle requires a bleed port in the nozzle and a means of mixing cold hydrogen to be compatible with the turbine temperature limitations. The mechanical and fluid flow complexity of the nozzle bleed port design could be eliminated by

[REDACTED]

providing a separate heating source of 20 Mw or less within the reactor assembly . A center island heater, occupying the space of one 6l element module could be included in the NRX-B design development with the additional advantage of providing a reinforcement of the base plate structure.

5. NRX-B Reactor Integration

The NRX-B, as a modular, cooled metal support plate reactor, with considerably more control worth than is available with the NRX-A configuration, will require a significantly modified pressure vessel-nozzle system. It is expected that the graphite inner reflector of NRX-A will be eliminated thus requiring some other means of forming the pressure wall between the first and second fluid flow pass. The core modules might all be regular hexagonal assemblies, thus giving a fluted periphery for the core reflector interface. If the reflector is no thicker than necessary for the control and reactivity requirements, the core will be larger than that of NRX-A thus requiring a larger diameter bolt circle for the nozzle flange.

While these features need not be incorporated in a backup NRX-B design, they are desirable for an NRX-B which can be developed on a more leisurely schedule. Figure 14 illustrates the significant NRX-B design features anticipated. These features are briefly identified and described below.

a. Core Modules

Thirty-seven Required ; 6l element modules with end transition pieces.

b. Reflector Assembly

Eighteen control drums with each drum region enclosed in a pressure shell.

c. Base Support Plate

The base support plate extending to the full diameter of the pressure vessel with a toroidal surface closure from the cylindrical pressure vessel to the conical nozzle interfaces. This provides considerably more rigidity to the base plate structure.

[REDACTED]

d. Pressure Vessel

The pressure shells between the core and reflector would extend to the pressure vessel where they would be brazed in place. This portion of the pressure vessel would be just the length of the seal plate from the upper surface of the base plate to the lower surface, such that the mating surfaces could be accurately machined. Bolt flanges would be formed at each end with coolant provided to the bolts and connected, between flanges, by vessel exterior lines taking the bolt coolant flow in parallel with the reflector. The pressure vessel head would continue past the elliptical equator sufficient to enclose the required shielding of the reactor assembly.

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Figure 1

Kiwi B-2A Hot End Detail

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Figure 2
Kiwi B-2A Cold End Detail

Figure 3
Key to Nodal Network

CONDITION 1

$h_1 = 4 \times 10^{-4} \text{ Btu/sec in.}^2 \text{ } ^\circ\text{R}$
 $h_2 = 1 \times 10^{-3} \text{ Btu/sec in.}^2 \text{ } ^\circ\text{R}$
 $h_3 = 1 \times 10^{-3} \text{ Btu/sec in.}^2 \text{ } ^\circ\text{R}$
 $h_4 = 1.5 \times 10^{-3} \text{ Btu/sec in.}^2 \text{ } ^\circ\text{R}$
 $q''' = .413 \text{ Btu/sec in.}^3$
 $T_{\text{gas}} = 500^\circ\text{R}$

Figure 4
Thermal Analysis - Rocker Support

Figure 5
B-2 Graphite Disc Axial Temperature Distribution
 $T_c, 4461^\circ\text{R}$; $P_c, 550 \text{ psia}$

Figure 6
Stress Distributions
Thermal, Mechanical, and Net Radial
B-2A Support Disc

Figure 7
NRX-A Hot End Sealing Concept

Figure 8
Axial Temperature Distribution

Figure 9
Isotherm Map $\pm 5\%$ Random Q

Figure 10
Isotherm Map $\pm 5\%$ Stacked Q

Figure 11
61 Element Cluster Dome Adapter

Figure 12

61 Element Cluster Cemented Module and Transition Structure

Figure 13A
Brazed Support Plate

Figure 13B
Forged Support Plate

Figure 14
NRX-B Reon Concept

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