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WANL-TME-2674
MARCH 1970

DATA ITEM S-54

**FUEL ELEMENT DESIGN
CONCEPTS (U)**

TRADE STUDY NO. 759

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DATA ITEM S-54

FUEL ELEMENT DESIGN CONCEPTS (U)

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TRADE STUDY NO. 759

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ACKNOWLEDGEMENT

WANL gratefully acknowledges the receipt from LASL of the composite and carbide Materials and Fabrication Data used in this Trade Study.

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ABSTRACT

(U) This report presents a Trade Study of design concepts for the Fuel Elements, a component of the NERVA Nuclear Subsystem, NSS. The purpose of this study is to evaluate and select the most logical and reasonable concept of this component for continued detail design and development consistent with the programmatic and technical requirements for the 10 hour, full flow cycle engine defined in the NERVA Program Requirement Document.

(U) The functional and technical requirements were identified by the Systems Engineering, December 1969 draft issue of component specification, EC-677566.

(U) The design concepts evaluated are:

- (U) Standard Geometry - Intermediate Expansion Graphite - Fuel Element
- (U) Standard Geometry - High Expansion Graphite - Fuel Element
- (U) Standard Geometry - Hybrid - Fuel Element
- (C-RD) 37 Hole - Hybrid - Fuel Element
- (U) Standard Geometry - Composite - Fuel Element
- (C-RD) Single Channel - Carbide - Fuel Element
- (U) Zigzag - Composite - Fuel Element
- (U) High Void - Composite - Long Core - Fuel Element
- (U) Standard Geometry - Graphite - Large Core - Fuel Element

(U) The evaluation and selection is based on the construction of a comparison matrix.

(U) Each concept is evaluated against the requirements and the tentative selection is based on an objective approach which identifies the concept having the maximum likelihood of meeting the functional, technical and programmatic requirements.

(U) The Standard Geometry, High Expansion Graphite Fuel Element and the Standard Geometry, Composite Fuel Element are the concepts selected and recommended for continued detail design and development. The major factors leading to this selection are described in the report. The R-1 concept will be recommended in a subsequent trade study prior to NSS PDR.

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INTRODUCTION

The fuel elements provide a source of nuclear heat and flow channels for the hydrogen propellant. In addition to these prime functional requirements, other requirements are imposed to obtain performance, design or manufacturing characteristics deemed necessary for the NERVA Nuclear Subsystem, NSS.

The requirements on this concept are presented in Section 1.0 and are based on draft issue of the component specification (EC-677566) and expanded to include SNPO-C technical directives and programmatic factors necessary to make a reasonable and logical design selection.

The design concepts are described in Section 2.0. This section includes a general description followed by detail descriptions relative to the requirements. The detail descriptions contain information summarized from the supporting data of the appendices.

Section 3.0 presents the comparison matrix which is constructed by abstracting in a few words the evaluation of each alternate relative to the requirements. These evaluations are utilized to select a design approach having the maximum likelihood of meeting all of the functional technical and programmatic requirements.

Section 4.0 presents a discussion of the selected design concept. The major factors influencing the evaluation are summarized in this section.

Unfueled support elements were not specifically traded in this Trade Study, because the considerations involved in the design of fueled elements include the effect of compatible unfueled support element.

Detailed information and analysis are contained in the appendices as listed below:

- A-1 Fuel Element and Core Size Parametric Study
- A-2 Layouts of Concepts Traded
- A-3 Nuclear Design Supporting Data

- A-4 Thermal Design Supporting Data
- A-5 Structural Design Supporting Data
- A-6 Reliability and Quality Assurance Supporting Data
- A-7 Materials Supporting Data
- A-8 Manufacturing Supporting Data
- A-9 Safety Supporting Data
- A-10 Radiation and Shielding Supporting Data

Discussions on the categorization of nuclear subsystem components in December, 1969, and subsequent SNPO direction by means of SNP-1, TD-11 and NP-1, TD-6 resulted in the agreement that this component was in category C - " Conceptual evaluation has or will indicate component design concepts are uncertain to such an extent that two or more concepts will probably have to be carried to PDR and possible into R-1 as well as considering alternates aimed at R-2." This report constitutes the necessary S-54 trade study work supporting the selection of the two concepts to be further developed for R-1 and subsequent reactors. The R-1 concept will be recommended in a subsequent trade study prior to NSS PDR.

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TABLE OF CONTENTS

<u>Section</u>	<u>Page</u>
INTRODUCTION	ii
1.0 IDENTIFICATION OF REQUIREMENTS FOR TRADE OFF	1-1
1.1 FUNCTION	1-1
1.2 SCOPE	1-1
1.3 FUNCTIONAL REQUIREMENTS	1-1
1.3.1 Nuclear Criticality	1-1
1.3.2 Flow and Thermal Characteristics	1-1
1.3.3 Reliability	1-2
1.4 TECHNICAL REQUIREMENTS	1-2
1.4.1 Bundling Load	1-2
1.4.2 Axial Load	1-2
1.4.3 Radial Expansion	1-2
1.4.4 Reactivity Shimming	1-2
1.4.5 Reactivity Loss	1-2
1.4.6 Corrosion	1-2
1.4.7 Safety	1-3
1.4.8 Interface Requirements	1-3
1.4.9 Weight	1-3
1.4.10 Nuclear Loading	1-3
1.4.11 Structural Criteria	1-4
1.4.12 Environmental Resistance	1-4

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<u>Section</u>	<u>Page</u>
(U) 1.5 NON-TECHNICAL REQUIREMENTS	1-4
(U) 1.5.1 Development and Qualification	1-4
(U) 1.5.2 Schedule	1-4
2.0 DESCRIPTION OF DESIGN APPROACHES (U)	2-1
(U) 2.1 STANDARD GEOMETRY - INTERMEDIATE EXPANSION GRAPHITE - FUEL ELEMENT	2-1
(U) 2.2 STANDARD GEOMETRY - HIGH EXPANSION GRAPHITE - FUEL ELEMENT	2-2
(U) 2.3 STANDARD GEOMETRY - HYBRID - FUEL ELEMENT	2-3
(C-RD) 2.4 37 HOLE - HYBRID - FUEL ELEMENT	2-4
(U) 2.5 STANDARD GEOMETRY - COMPOSITE - FUEL ELEMENT	2-5
(C-RD) 2.6 SINGLE CHANNEL - CARBIDE - FUEL ELEMENT	2-6
(U) 2.7 ZIGZAG - COMPOSITE - FUEL ELEMENT	2-6
(U) 2.8 HIGH VOID - COMPOSITE - LONG CORE - FUEL ELEMENT	2-7
(U) 2.9 STANDARD GEOMETRY - GRAPHITE - LARGE CORE - FUEL ELEMENT	2-8
3.0 COMPARISON MATRIX (U)	3-1
(U) TABLE 3-1 RELIABILITY CONSIDERATIONS	3-3
(U) TABLE 3-2 PERFORMANCE CONSIDERATIONS	3-4
(U) TABLE 3-3 WEIGHT CONSIDERATIONS	3-8

THIS PAGE IS UNCLASSIFIED

<u>Section</u>		<u>Page</u>
TABLE 3-4	SAFETY CONSIDERATIONS	3-9
TABLE 3-5	DEVELOPMENT & QUALIFICATION CONSIDERATIONS	3-11
TABLE 3-6	SCHEDULE CONSIDERATIONS	3-13
TABLE 3-7	SUMMARY	3-14
4.0	SELECTION OF DESIGN	
A	APPENDIX	
A-1	FUEL ELEMENT AND CORE SIZE PARAMETRIC STUDY	A-1-1
A-2	LAYOUTS OF CONCEPTS TRADED	A-2-1
A-3	NUCLEAR SUPPORTING DATA	A-3-1
A-4	THERMAL DESIGN SUPPORTING DATA	A-4-1
A-5	STRUCTURAL DESIGN SUPPORTING DATA	A-5-1
A-6	RELIABILITY AND QUALITY ASSURANCE SUPPORTING DATA	A-6-1
A-7	MATERIALS SUPPORTING DATA	A-7-1
A-8	MANUFACTURING SUPPORTING DATA	A-8-1
A-9	SAFETY SUPPORTING DATA	A-9-1
A-10	RADIATION AND SHIELDING SUPPORTING DATA	A-10-1

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1.0 IDENTIFICATION OF REQUIREMENTS FOR TRADE OFF (U)

1.1 FUNCTION (U)

(U) The fuel elements provide a source of nuclear heat and flow channels for the hydrogen propellant. In addition to these prime functional requirements, other requirements are imposed to obtain performance, design and manufacturing characteristics deemed necessary for the NERVA Nuclear Subsystem.

1.2 SCOPE (U)

(U) This part of the trade study identifies the functional, technical and non-technical requirements the fuel must meet in order to be considered as an alternate for the R-1 reactor.

1.3 FUNCTIONAL REQUIREMENTS (U)

1.3.1 Nuclear Criticality (U)

(U) The fuel, as a part of the NSS, shall be capable of sustaining a controlled nuclear heat generation through 60 operating cycles during 10 hours operation with fuel loading limitations of: (a) 630 mg/cc for graphite fuel, (b) 700 mg/cc for composite fuel, and (c) about 2000 mg/cc for carbide fuel.

1.3.2 Flow and Thermal Characteristics (U)

(U) The fueled elements shall have axial flow passages which in conjunction with hardware specified in the Cluster Hardware Specification No. EC-677558 shall accommodate passage of the propellant through the reactor core, exhausting it at the nominal fuel exit gas conditions of 450 psia, 4284°R. Reference core pressure drop is 170 psia and a core inlet temperature 256°R.

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1.3.3 Reliability (U)

(U) The reliability design requirement for the fuel is a 0.998272 probability of success in meeting the primary performance characteristics of the NSS under the required life and environmental conditions.

1.4 TECHNICAL REQUIREMENTS (U)

1.4.1 Bundling Load (U)

(U) The fuel shall be capable of withstanding the maximum nominal mechanical and pneumatic bundling load as applied and distributed by the Core Periphery.

1.4.2 Axial Load (U)

(U) The fuel shall be capable of transmitting to the support hardware the axial load which acts upon it. The maximum load to be transmitted to the Cluster Support Hardware is 110 lbs. per fuel element.

1.4.3 Radial Expansion (U)

(C-RD) The radial expansion of the reactor core, during reactor operation, acting against the lateral support system shall not be larger than 1.665% of the as-assembled core dimensions.

1.4.4 Reactivity Shimming (U)

(U) The core reactivity change available through shimming shall be $\pm$ \$3.

1.4.5 Reactivity Loss (U)

(U) The total reactivity loss of the core, after the performance requirements have been met, shall not exceed \$3. (Design Goal)

1.4.6 Corrosion (U)

(U) Weight loss characteristics of the core, after meeting the performance requirements, shall be as follows:

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- a. The axial distribution of core-average element weight loss shall be such the axially integrated carbon reactivity loss shall not exceed \$3.
- b. The peak incremental weight loss in grams per inch of element shall not exceed the amount required to assure that the element, in conjunction with the hot end support components, remains sufficiently intact to support the imposed axial load and prevent ejection of parts.

1.4.7 Safety (U)

(U) The fuel shall meet the requirements specified in NSS Specification Number CP-677555. In addition, the fuel shall be designed to permit manufacture, transporting, storage, and handling of the individual elements in accordance with the requirements specified in Department of Transportation Regulations, 49CFR171 to 178, for transportation of explosives and other dangerous articles by land and water in rail freight service and by motor vehicle (highway) and water including specifications for shipping containers dated October 4, 1968.

1.4.8 Interface Requirements (U)

(C-RD) The fuel elements shall not restrict or prevent interfacing components from performing their function. The across flats hexagonal dimension of the reference fuel (939J685, Rev. E) is .753 inch and the length is 52.0 inches.

1.4.9 Weight (U)

(U) The weight of the fuel shall be minimized within the limits imposed by the functional requirements. The weight of the Reference Design (939J685, Rev. E with High Expansion Graphite Fuel) fuel envelope is 3645 pounds.

1.4.10 Nuclear Loading (U)

(U) The fueled elements shall have varied fuel loadings as required to limit the peak-to-average reactor radial power factor from 0.6 to 1.1. The loading range for the Reference Design (939J685, Rev. E High Expansion Graphite Fuel) is from 268 to 630 mg/cc.

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1.4.11 Structural Criteria (U)

(U) The structural criteria for the fuel shall be in accordance with SNPO-C-1, "NERVA Program Structural Design Requirements".

1.4.12 Environmental Resistance (U)

(U) The fuel shall be compatible with or protected from degradation, such as hydrolysis, by the non-operating environment.

1.5 NON-TECHNICAL REQUIREMENTS (U)

1.5.1 Development and Qualification (U)

(U) The state of the available technology shall be such that the fuel will be producible, can be evaluated, and is suitable for the reactor. The required development effort must be compatible with CY-70 to CY-72 budgetary plans.

1.5.2 Schedule (U)

(U) The time required for development and production of fuel shall be considered and used as a consideration in the selection.

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2.0 DESCRIPTION OF DESIGN APPROACHES (U)

(C-RD) This section describes fuel element designs evaluated in this trade study. The nine designs were chosen by a parametric study which considered the effects of design variables on performance. The design variables considered were: core length and diameter; fuel element flow channel size, pitch and number; across flats size of fuel element; fuel material, and coating. The effect of variables on the performance was evaluated on the basis of thermal stresses, core pressure drop fuel corrosion, weight and nuclear loading.

(C-RD) The fuel element designs evaluated in this trade study include NRX type fuels, i.e., fuels of standard (.753 across flats) hexagonal lateral geometry containing 19 flow channels and made of graphite. Other fuels considered are made of composite or solid solution materials, have larger across flats dimensions, are longer and have non-hexagonal peripheries.

2.1 ALTERNATE 1 - STANDARD GEOMETRY, INTERMEDIATE EXPANSION GRAPHITE FUEL ELEMENT (U)

(U) The design of the fuel element made of intermediate expansion graphite and having standard geometry is shown on WANL Drawing, Number 711J596. The salient features of this design are defined in WANL Drawing, Number 947C812, which is shown in Appendix A-2 as Figure A-2.1.1-1.

(C-RD) The geometry of this fuel element design is basically similar to the geometry of fuel elements used in NRX reactors. The fuel element of this design has 19 flow channels located within a .753 hex lateral geometry and extending throughout the 52-inch element length. The material of the fuel element consists of pyrocarbon coated UC_2 fuel beads dispersed through a graphite matrix of intermediate CTE and a NbC graphite composite tip brazed to the element aft end. The fuel element is coated with an NbC-ZrC coating in the flow channels and at the aft end.

(U) A significant property difference between fuel elements of this design and the fuel elements used in NRX reactors is a decrease of CTE difference between the graphite matrix and the coating; a property which lowers the stresses due to matrix-coating interaction.

The diminished stresses cause fewer cracks in the coating and, therefore, afford better corrosion protection of the graphite matrix by the hydrogen propellant. The improvements in coating which have come about since the NRX reactors further enhance the graphite matrix corrosion resistance. It shall be noted that the difference between coating and matrix CTE's for this design alternate, though lower than in the NRX fuel, is greater than for any other alternate considered in this study.

(C-RD) The intermediate graphite fuel element is relatively insensitive to stresses caused by transverse temperature gradients across the element and has the lowest core radial expansion (1.146% nominally) of any alternate.

(U) The weight of the core made of the Alternate 1 fuel and the weight of required shielding are 7510 lbs., which is the lowest of all the alternate designs considered.

(U) Significant advantages of the Alternate 1 fuel design are the low weight, the availability of fabrication processes and facilities, and the high probability of meeting reasonable production and test schedules.

(U) The disadvantage of the intermediate expansion graphite, Alternate 1 fuel design, is the observed tendency for the aft end corrosion rates to increase with cyclic operation. This causes loss of reactivity and low margin of safety. The ability of the brazed joint to withstand the cyclic operation at the 30° temperature level may be a problem.

2.2 ALTERNATE 2 - STANDARD GEOMETRY, HIGH EXPANSION GRAPHITE FUEL ELEMENT (U)

(U) The salient design features of high expansion fuel element are defined on WANL Drawing, Number 947C813, which is shown in Appendix A-2 as Figure A-2.1.1-2.

(C-RD) The geometry of this fuel element design is basically similar to geometry of fuel elements used in NRX reactors. The fuel element of this design has 19 flow channels located within a .753 hex lateral geometry and extending throughout the 52-inch element length. The material of the fuel element consists of pyrocarbon coated UC₂ fuel beads dispersed through a graphite matrix of high CTE and a NbC graphite composite tip brazed to the element aft end. The fuel element is coated with an NbC-ZrC coating in flow

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channels and at the aft end. Fuel elements of Alternate 2 design are planned to be incorporated in Pewee-2 test. The increase in CTE of the graphite element matrix of this element versus the element design of Alternate 1 is due to a further attempt to diminish the matrix-coating interaction and the resulting element corrosion.

(C-RD) The high expansion graphite fuel elements are relatively insensitive to stresses caused by transverse temperature gradient across the element and cause the core to expand during operation 1.48% nominally or well within the imposed limits.

(U) The weight of the core made of the Alternate 2 fuel and the weight of required shielding are 7542 lbs., which is second lowest of all the alternate designs considered.

(U) Significant advantages of the Alternate 2 fuel design are availability of fabrication processes and facilities, the high probability of meeting reasonable production and test schedules and the low weight.

(U) The disadvantage of the Alternate 2 design is the observed tendency for the aft end corrosion rates to increase with cyclic operation. This causes loss of reactivity and low margin of safety. The ability of the brazed joint to withstand the cyclic operation at the 3000 temperature level may be a problem.

2.3 ALTERNATE 3 - STANDARD GEOMETRY - HYBRID FUEL ELEMENT (U)

(U) The salient features of the hybrid fuel element are defined on WANL Drawing, Number 947C810 which is shown in Appendix A-2 as Figure A-2.1.1-3.

(C-RD) The geometry of this element design is basically similar to the geometry of fuel elements used in NRX reactors. The element contains 19 flow channels within the .753 hex lateral geometry and 52-inch length. The fuel element of Alternate 3 design consists of a 35-inch long forward section (composed of pyrocarbon coated UC_2 fuel beads dispersed through high CTE graphite matrix as defined in Section 2.2) brazed to a 17-inch long aft section composed of 30 v/o of UC-ZrC and graphite composite. The flow channels and the aft surfaces of the fuel element of Alternate 3 design are coated with ZrC. The fuel elements of Alternate 3 design were planned for inclusion in Pewee-2 test.

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(U) The hybrid construction of Alternate 3 element design has an improved aft end corrosion resistance over Alternates 1 and 2 (the all graphite fuel elements). The length of the aft section of the element (composite material) is due to a compromise between the desirable high corrosion resistance and the undesirable increase in weight coupled with the placement of the braze joint in the zone of lower temperature. This provides more temperature margin for the braze and lower thermal stress due to CTE mismatch of the graphite and composite sections.

(U) The weight of the core made of hybrid fuel and the required shielding is 8022 lbs., or 512 lbs. more weight than Alternate 1.

(U) The significant advantages of Alternate 3 fuel design concept are potentially improved aft end corrosion resistance, a higher temperature margin than the all graphite concepts considered and a high margin from the maximum allowable fuel loading.

(U) The disadvantages of this fuel concept are the relatively high fission product release and increased weight over the graphite elements of Alternates 1 and 2. Development of surveillance techniques, joint integrity, and manufacturing processes impose a developmental and schedular penalty.

2.4 ALTERNATE 4 - 37 HOLE - HYBRID FUEL ELEMENT (C-RD)

(U) The salient design features of the hybrid fuel element are defined on WANL Drawing, Number 947C816, which is shown in Appendix A-2 as Figure A-2.1.1-4.

(U) The Alternate 4 fuel element design was devised in order to reduce thermal stresses. The reduction is achieved through a decrease in flow channel pitch and diameter, thus creating a better heat transfer condition. The extent of improvement makes this alternate second choice in reliability considerations.

(C-RD) The longitudinal geometry and materials construction is the same as Alternate 3, however, the transverse geometry of the fuel element is different in that it contains 36 flow channels within a 0.799 hex envelope. The flow channels and the aft surface of the fuel element are coated with ZrC.

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- (U) This element has a larger bore surface area than any other alternate, which results in higher corrosion and reactivity loss rates than the other composite designs. In addition, the thin webs make the element more susceptible to corrosion damage.
- (U) The weight of the core made of Alternate 4 fuel and the required shielding is 7957 lbs., or 447 lbs. more weight than Alternate 1.
- (C-RD) Because the 37 hole element geometry is unlike previously manufactured fuel elements extensive development effort will be required.
- (U) The significant advantages of Alternate 4 fuel are lowered combined stresses, high reliability and high temperature margin.
- (U) The disadvantages of this alternate are high fission product release, increased corrosion rate sensitivity, extended development requirement and low fuel loading margin.

2.5 ALTERNATE 5 - STANDARD GEOMETRY - COMPOSITE FUEL ELEMENT (U)

- (U) The salient features of composite fuel element are defined on WANL Drawing Number 947C809 which is shown in Appendix A-2 as Figure A-2.1.1-5.
- (C-RD) This design has the same geometry as fuel elements of Alternate 1 design; it differs in that it consists entirely of 30 v/o UC-ZrC and graphite composite. All flow channels and aft surfaces of this design are carbide coated with ZrC. The philosophy of this design is based on the need for increased corrosion resistance necessary for long life reactor. This design of fuel elements is planned to be incorporated in a portion of the Pewee 2 test reactor.
- (U) The weight of the core made of Alternate 5 fuel and the required shielding is 9153 lbs., or 1643 lbs. more than Alternate 1.
- (U) The significant advantages of Alternate 5 fuel design concept are: high corrosion resistance potential because of the presence of carbide in the matrix, high fuel loading margin, lowest sensitivity to stresses contributed by element-to-element temperature differences, and the absence of a brazed joint.

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- (U) The disadvantages of Alternate 5 fuel are: high fission product release and increased core weight over the graphite elements.

2.6 ALTERNATE 6 - SINGLE CHANNEL - CARBIDE FUEL ELEMENT (C-RD)

- (U) The salient design features of the carbide fuel element are defined on WANL Drawing, Number 947C817, which is shown in Appendix A-2 as Figure A-2.1.1-6.
- (C-RD) This fuel element is an array of single flow channel short length sections (5.2 inches) stacked end to end. The element is made entirely of UC-ZrC. The geometry of this fuel design alternate is a .170 hex of 5.2" length with a .120" diameter flow channel.
- (U) The weight of the core made of Alternate 6 fuel and the required shielding is 12,075 lbs., or 4,565 lbs. more than Alternate 1.
- (U) The fuel of Alternate 6 design has the highest temperature margin and corrosion resistant potential among all alternates considered because of high temperature of matrix phase change and absence of free carbon and coating.
- (U) The reliability of the all carbide fuel is relatively high and the margin of fuel loading is the highest of all concepts considered.
- (U) The disadvantages of this alternate fuel design are: small stress margin of safety, high stress due to bundling, and high radial expansion.
- (U) The core made of Alternate 6 fuel is the heaviest of all considered. Because of many small pieces of fuel and the probability of tolerance stack-up, the poisoning of fuel for personnel safety is very difficult. Except for fission product release, which is about average of all considered concepts, all the safety emphasis factors of Alternate 6 fuel are the lowest.
- (U) Alternate 6 fuel is disadvantageous from the development and schedular point of consideration.

2.7 ALTERNATE 7 - ZIG-ZAG, COMPOSITE FUEL ELEMENT (U)

- (U) The salient design features of the zig-zag fuel element are defined on WANL Drawing Number 947C807, which is shown in Appendix A-2 as Figure A-2.1.1-7.

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(C-RD) The geometry of this fuel element design with the exception of the element periphery is basically similar to the geometry of fuel elements used in NRX reactors. The fuel of this design has 19 flow channels within a quasi-hexagonal lateral geometry of a mean hex dimension of 0.753 and 52-inch length. The fuel element consists entirely of 30 v/o UC-ZrC and graphite composite and has all the flow channels and aft surfaces carbide coated with ZrC.

(U) The weight of the core made of Alternate 7 fuel and the required shielding is 9153 lbs. or 1643 lbs. more than Alternate 1.

(U) The advantages are the same as those given for the standard geometry composite element (Alternate 5). The disadvantages are also the same with an additional development effort required because of the irregular outer surface.

2.8 ALTERNATE 8 - HIGH VOID, COMPOSITE, LONG CORE FUEL ELEMENT (U)

(U) The salient design features of this fuel element are defined on WANL Drawing, Number 947C814, which is shown in Appendix A-2 as Figure A-2.1.1-8.

(C-RD) The lateral geometry of this fuel element is basically similar to the geometry of fuel elements used in NRX reactors. The flow channel pitch is the same as in Alternate 1; however, the flow channel diameter is larger. The length of the fuel element is increased from 52.00 to 62.00 inches. These changes were made in order to reduce thermal stresses and result in this alternate being the first choice for reliability considerations. The material of this fuel element design is a 30 v/o UC-ZrC and graphite composite having all the flow channels and the aft surfaces coated with ZrC coating.

(U) The weight of the core made of fuel of Alternate 8 design and the required shielding are 9844 lbs., or 3064 lbs. more than Alternate 1. However, because of greater fuel length, the reflector the nuclear subsystem will require is also longer and heavier by 730 lbs. as compared with the reflector of Alternate 1.

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(U) The main advantage of Alternate 8 fuel design is the highest combined stress margin of safety of all the alternates considered.

(U) The disadvantages of the high void, composite fuel design concept are high fission product release and the weight penalty.

2.9 ALTERNATE 9 - STANDARD GEOMETRY, GRAPHITE, LARGE CORE FUEL ELEMENT (U)

(U) The salient design features of this fuel element are defined on WANL Drawing Number 947C814, which is shown in Appendix A-2 as Figure A-2.1.1-9.

(C-RD) With the exception of length, 76.00 inches, the fuel element design is the same as Alternate 2.

(U) The weight of the core made of fuel in Alternate 9 design and the required shielding are 9904 lbs., or 4664 lbs. more than Alternate 1. However, because of larger core diameter and length, the reflector will be larger and 2270 lbs. heavier than the reflector for Alternate 1.

(U) The main advantages of Alternate 9 fuel are: high combined stress margin of safety, low radial expansion, high nuclear fuel loading margin and high over-temperature capability.

(U) The disadvantages of the Alternate 9 fuel design are high sensitivity to stresses due to bundling loads and highest reactor weight of all alternates considered.

3.0 COMPARISON MATRIX (U)

(U) This section describes the method used to construct the Comparison Matrix for selection of the Fuel Element design concept.

(U) The selection method includes the following steps:

- (1) Determination of functional, technical, and non-technical requirements.
- (2) Evaluation of each design concept against each of the requirements.
- (3) Construction of the Comparison Matrix.

(U) The requirements used for evaluation are based on the current ECC specification and are summarized in Section 1 as the Functional, Technical, and Non-Technical Requirements. Three major requirements groups were selected: Reliability, Performance, and Weight.

(U) Safety is given a low weighting factor for this study. The Reliability evaluation considers, among the total failure modes identified, the effect and probability of those failure modes leading to loss of the Emergency Operating Mode capability. The safety factor emphasizes those failures leading to loss of the Emergency Operating Mode capability.

(U) For the development of the Fuel Elements in a reasonable time, consideration must be given to the development and qualification difficulty and its effect on schedule. For this reason, Development and Qualification and Schedule are included as additional evaluation requirements.

(U) In summary, the requirements selected as a basis for evaluating the design alternates are: Reliability, Performance, Weight, Safety, Development and Qualification, and Schedule. The Comparison Matrix on each of these requirements is presented in Tables 3-1 through Table 3-6.

(U) The second step of this procedure is to evaluate the design alternates against the requirements. Each requirement is reduced to individual considerations deemed important in judging one concept against another. Based upon the standing of alternates in each

consideration, the "emphasis factors" are established. The emphasis factors of the concepts, for each consideration, are identified on Tables 3-1 through 3-6 by the circled numbers.

The emphasis factors meet the following criteria:

Emphasis Factor Criteria		
<u>Selection</u>	<u>Value</u>	<u>Rule</u>
First Selection, a	0 to 1.0	
Second Selection, b	0 to 1.0	$b < a$
Third Selection, c	0 to 1.0	$c < b$
$\sum =$		1.0

(U) The emphasis factors, for each requirement, are summed by alternates and divided by the number of considerations in a particular requirement. These numbers are the Figures of Merit for a Requirement and are shown at the bottom of Tables 3-1 through 3-6. The largest Figure of Merit represents the best alternate relative to the Requirement. The summary evaluation is presented in Table 3-7. Figures of Merit previously developed in Tables 3-1 through 3-6 are multiplied by a weighting factor for that Requirement, and this value is entered into the matrix. The sum of these columns is the overall Figure of Merit for each concept. The following weighting factors were used:

<u>Requirement</u>	<u>Weighting Factor</u>
Reliability	.30
Performance	.25
Weight	.25
Safety	.10
Development and Qualification	.05
Schedule	.05

NOMENCLATURE TABLE 3-1
RELIABILITY CONSIDERATIONS
FUEL ELEMENTS
CP-677566

ASTRONUCLEAR LABORATORY
COMPARISON MATRIX OF DESIGN APPROACHES

SELECTION

FUNCTIONAL & TECHNICAL DESIGN REQUIREMENTS	1	2	3	4	5	6	7	8	9
	Standard Geometry Intermediate Expansion Graphite	Standard Geometry High Expansion Graphite	Standard Geometry Hybrid	37 Hole Hybrid	Standard Geometry Composite	Single Channel Carbide	Zigzag Composite	High Void Composite Long Core	Standard Geometry Graphite Large Core
FAILURE MODES (Objective: Minimum failure probability)	(.057)	(.098)	(.112)	(.134)	(.113)	(.119)	(.113)	(.141)	(.112)
Overall Failure Estimates	1120×10^{-4}	230×10^{-4}	132×10^{-4}	55×10^{-4}	125×10^{-4}	100×10^{-4}	126×10^{-4}	43×10^{-4}	132×10^{-4}
Most Probable Failure Modes and Their Estimated Probability	Corrosion at cracks in channel coating (10^{-1}).	Reactivity loss via excessive hot end corrosion, due to coating degradation from thermal cycling (10^{-2}). Reactivity loss via ex- cessive corrosion at interfaces of matrix cracks developed by thermal stresses (10^{-2}).	Reactivity loss via ex- cessive corrosion at interfaces of matrix cracks developed by thermal stresses (10^{-2}).	Reactivity loss via ex- cessive corrosion through hot end dif- fusion (2×10^{-3}). Reactivity loss via ex- cessive corrosion of intergranular binder carbon, abetted by porosity (2×10^{-3}).	Reactivity loss via ex- cessive corrosion at interfaces of matrix cracks developed by thermal stresses (10^{-2}).	Thermal stress cracking of the carbide (10^{-2}).	Reactivity loss via ex- cessive corrosion, at interfaces of matrix cracks developed by thermal stresses (10^{-2}).	Reactivity loss via ex- cessive corrosion through hot end dif- fusion (1.5×10^{-3}). Reactivity loss via ex- cessive corrosion of intergranular binder carbon, abetted by porosity (1.5×10^{-3}).	Reactivity loss via ex- cessive hot end cor- rosion, due to coat- ing degradation from thermal cycling (10^{-2}).
SEVERITY OF FAILURE MODES (Objective: Minimize severity of failures)	(.058)	(.101)	(.115)	(.139)	(.117)	(.093)	(.117)	(.145)	(.115)
Overall Severity Estimates	112×10^{-4}	23×10^{-4}	13.2×10^{-4}	5.5×10^{-4}	12.5×10^{-4}	30×10^{-4}	12.6×10^{-4}	4.3×10^{-4}	13.2×10^{-4}
Most Severe Failure Modes Their Estimated Probability and Severity of Effects*.	Corrosion at cracks in channel coating (10^{-1}).	Corrosion at inter- faces of matrix cracks opened by thermal stresses (10^{-2}).	Corrosion at interfaces of matrix cracks open- ed by thermal stresses (10^{-2}).	Reactivity loss via ex- cessive corrosion through hot end dif- fusion (2×10^{-3}). Reactivity loss via ex- cessive corrosion of intergranular binder carbon, abetted by porosity (2×10^{-3}).	Corrosion at interfaces of matrix cracks opened by thermal stresses (10^{-2}).	Thermal stress cracking of the carbide (10^{-2}).	Corrosion at interfaces of matrix cracks open- ed by thermal stresses (10^{-2}).	Reactivity loss via ex- cessive corrosion through hot end dif- fusion (1.5×10^{-3}). Reactivity loss via ex- cessive corrosion of intergranular binder carbon, abetted by porosity (1.5×10^{-3}).	Reactivity loss via ex- cessive hot end cor- rosion, due to coat- ing degradation from thermal cycling (10^{-2}).

NOTE:
* Severity of effects for all failure modes, in all cases listed,
entails: limited mission capability, reduced redundancy and
increased risk of other failures. For Concept 6, additionally,
partial obstruction of flow channels likely to result from
thermal stress cracking can lead to major NSS degradation
necessitating emergency action.

NOMENCLATURE		TABLE 3-1 (Cont'd.)		<div><div><div>W</div></div><div>ASTRONUCLEAR LABORATORY</div><div>COMPARISON MATRIX OF DESIGN APPROACHES</div></div>							SELECTION
		1	2	3	4	5	6	7	8	9	
FUNCTIONAL & TECHNICAL DESIGN REQUIREMENTS		Standard Geometry Intermediate Expansion Graphite	Standard Geometry High Expansion Graphite	Standard Geometry Hybrid	37 Hole Hybrid	Standard Geometry Composite	Single Channelled Carbide	Zigzag Composite	High Void Composite Long Core	Standard Geometry Graphite Large Core	
Effect of Most Severe Failure Modes		Reactivity loss to an extent that prevents completion of mission. (Eventual inability to restart.)	Reactivity loss to an extent that prevents completion of mission. (Eventual inability to restart.) Also, eventual extension of cracks to element surfaces so as to allow access of unsaturated hydrogen to core interstices, further accelerating corrosion of F/E surfaces and of core periphery.	Reactivity loss to an extent that prevents completion of mission. (Eventual inability to restart.) Also, eventual extension of cracks to element surfaces so as to allow access of unsaturated hydrogen to core interstices, further accelerating corrosion of F/E surfaces and of core periphery.	Reactivity loss to an extent that prevents completion of mission. (Eventual inability to restart.) Also, eventual weakening of matrix web so as to allow interchannel, or channel to surface, penetrations, causing flow irregularities and more rapid corrosion.	Reactivity loss to an extent that prevents completion of mission. (Eventual inability to restart.) Also, eventual extension of cracks to element surfaces so as to allow access of unsaturated hydrogen to core interstices, further accelerating corrosion of F/E surfaces and of core periphery.	Pieces loosened by thermal stress cracking partially obstruct the flow channels, causing perturbation of flow, pressure, and temperature, which could lead to serious degradation of the core. Increased leakage of hydrogen to core interstices may cause serious corrosion of graphites in periphery. (10 ⁻²)	Reactivity loss to an extent that prevents completion of mission. (Eventual inability to restart.) Also, eventual extension of cracks to element surfaces so as to allow access of unsaturated hydrogen to core interstices, further accelerating corrosion of F/E surfaces and of core periphery.	Reactivity loss to an extent that prevents completion of mission. (Eventual inability to restart.) Also, eventual weakening of matrix web so as to allow interchannel, or channel to surface, penetrations, causing flow irregularities and more rapid corrosion.	Reactivity loss to an extent that prevents completion of mission. (Eventual inability to restart.)	
	RELIABILITY FIGURE OF MERIT	.058	.099	.114	.136	.115	.106	.115	.143	.114	8; 4; 5; 7; 3 & 9; 6; 2; 1

NOMENCLATURE TABLE 3-2

PERFORMANCE CONSIDERATIONS
FUEL ELEMENTS
CP-677566

ASTRONUCLEAR LABORATORY
COMPARISON MATRIX OF DESIGN APPROACHES

SELECTION

FUNCTIONAL & TECHNICAL DESIGN REQUIREMENTS		1 Standard Geometry Intermediate Expansion Graphite	2 Standard Geometry High Expansion Graphite	3 Standard Geometry Hybrid	4 37 Hole Hybrid	5 Standard Geometry Composite	6 Single Channel Carbide	7 Zigzag Composite	8 High Void Composite Long Core	9 Standard Geometry Graphite Large Core
FLOW & THERMAL REQUIREMENTS	FUEL TEMPERATURE MARGIN (Objective: Maximum margin between fuel operating temperature and fuel temperature capability.)	(.086)	(.086)	(.119)	(.121)	(.098)	(.172)	(.098)	(.112)	(.108)
	36Fuel Temperature	4811°R at Sta. 46.0	4811°R at Sta. 46.0	4822°R at Sta. 46.0	4726°R at Sta. 50.0	4822°R at Sta. 46.0	4797°R at Sta. 46.0	4822°R at Sta. 46.0	4782°R at Sta. 57.0	4721°R at Sta. 73.0
	36Wall Temperature	4737°R at Sta. 48.0	4737°R at Sta. 48.0	4737°R at Sta. 48.0	4701°R at Sta. 50.0	4737°R at Sta. 48.0	4765°R at Sta. 46-48	4737°R at Sta. 46-48	4719°R at Sta. 59.5	4680°R at Sta. 73.0
	Temperature of Local Melting of NbC-ZrC Coating	4990°R	4990°R	----	----	----	No coating thus, no local coating melting.	----	----	4990°R
	Temperature of 2% of Fuel Migration through Bead Shell after 1 Hour	4630°R	4630°R	No fuel beads in aft (hot) end of element. Thus, no migration.	No fuel beads in aft (hot) end of element. Thus, no migration.	No fuel beads in element. Thus, no migration.	No fuel beads in element. Thus, no migration.	No fuel beads in element. Thus, no migration.	No fuel beads in element. Thus, no migration.	4630°R
	Temperature of 100% of Fuel Migration through Bead Shell after 1 Hour	4920°R	4920°R	No fuel beads in aft (hot) end of element. Thus, no migration.	No fuel beads in aft (hot) end of element. Thus, no migration.	No fuel beads in element. Thus, no migration.	No fuel beads in element. Thus, no migration.	No fuel beads in element. Thus, no migration.	No fuel beads in element. Thus, no migration.	4920°R
	Temperature of Local Matrix Phase Incipient Melting	----	----	5170°R on max. 520 mg/cc fuel loading.	5080°R on max. 618 mg/cc fuel loading.	5110°R on max. 579 mg/cc fuel loading.	5300°R on max. 1200 mg/cc fuel loading.	5110°R on max. 579 mg/cc fuel loading.	5110°R on max. 575 mg/cc fuel loading.	----
	SUMMARY OF 36MARGINS									
	2% Fuel Bead Migration After 1 Hour	-181°R	-181°R	----	----	----	----	----	----	-91°R
	100% Fuel Bead Migration After 1 Hour	+109°R	+109°R	----	----	----	----	----	----	+199°R
	Coating, Local Melting Zr-U-C Eutectic	+253°R	+253°R	----	----	----	----	----	----	+310°R
	Matrix, Local Phase Change	----	----	+348°R	+354°R	+288°R	+503°R	+288°R	+328°R	----

NOMENCLATURE TABLE 3-2 (Cont'd.) PERFORMANCE CONSIDERATIONS FUEL ELEMENTS CP-677566		<div>ASTRONUCLEAR LABORATORY</div> <div>COMPARISON MATRIX OF DESIGN APPROACHES</div> <div>123456789</div>									SELECTION
FUNCTIONAL & TECHNICAL DESIGN REQUIREMENTS		Standard Geometry Intermediate Expansion Graphite	Standard Geometry High Expansion	Standard Geometry Hybrid	37 Hole Hybrid	Standard Geometry Composite	Single Channel Carbide	Zigzag Composite	High Void Composite Long Core	Standard Geometry Graphite Large Core	
FLOW & THERMAL REQUIREMENTS (Cont'd.)	NUCLEAR LOADING (Objective: Flat radial exit gas temperature profile within orifice capability at design operating drum angle.) Ability to Match Fuel Loading To Neutron Flux	(.100) Min. Loading 269 mg/cc	(.100) Min. Loading 268 mg/cc	(.100) Min. Loading 221 mg/cc	(.100) Min. Loading 262 mg/cc	(.100) Min. Loading 247 mg/cc	(.200) Min. Loading 511 mg/cc. Very small across flats di- mension results in ca- pability of fine load- ing if required.	(.100) Min. Loading 247 mg/cc	(.100) Min. Loading 245 mg/cc	(.100) Min. Loading 181 mg/cc	
	CORROSION RESISTANCE (Objective: Meeting required life of 10 hrs and 60 cycles.) 1. Fuel and Core Geometry Effects*	(.023)	(.046)	(.116)	(.093)	(.162)	(.189)	(.162)	(.139)	(.070)	
	Reactivity (Hot End Diff. Loss (\$))	0.70	0.70	0.76	1.02	0.78	No free carbon available	0.78	0.86	0.68	
	Mid-Band	0.78	0.78	0.78	1.17	0.85		0.85	1.08	1.12	
	Material Loss Due to Hot End Diffusion Based on De- fect Free Coating						No free carbon available				
	Incremental Loss gm/inch	1.17	1.17	1.17	1.66	1.17		1.17	1.13	.96	
	Relative Carbon Loss (gm) per Unit Bore Area	1.0	1.0	1.0	0.87	1.0		1.0	0.92	0.84	
	Percent Carbon Loss from Minimum Web**	17.2	17.5	19.7	26.0	19.7		19.7	19.7	14.4	
	Relative Interelement Hydro- gen Leakage	1.0	1.0	1.0	0.94	1.0	1.16	1.153	0.70	0.78	
	NOTES: * Pewee-2 development coating profile, i.e., 3.1 mils thickness at hot end. Defect free hot end corrosion based on carbon diffusion loss only. Mid-band corrosion proportional to bore surface area and fuel length with a ten gram loss assumed for the reference element. ** Assuming carbon is removed equally from the circumference of the bore, then in- cremental grams lost can be related to an annular region from which carbon has been removed to a certain depth dimension from the bore wall. The per- cent carbon removed from interior web is defined as the ratio of this distance to the minimum web thickness.										

W59778-1

NOMENCLATURE TABLE 3-2 (Cont'd.) PERFORMANCE CONSIDERATIONS FUEL ELEMENTS CP-677566		ASTRONUCLEAR LABORATORY COMPARISON MATRIX OF DESIGN APPROACHES									SELECTION
FUNCTIONAL & TECHNICAL DESIGN REQUIREMENTS		1 Standard Geometry Intermediate Expansion Graphite	2 Standard Geometry High Expansion	3 Standard Geometry Hybrid	4 37 Hole Hybrid	5 Standard Geometry Composite	6 Single Channel Carbide	7 Zigzag Composite	8 High Void Composite Long Core	9 Standard Geometry Graphite Large Core	
STRUCTURAL INTEGRITY	2. Potential to Attain High Resistance to Cyclic Effects	LOW Minimum number of element tests indicate most prone to accelerated deterioration due to cyclic effects. High Graphite-Hydrogen reaction rate degrades structure.	LOW/MODERATE Prone to accelerated deterioration due to cyclic effects. High Graphite-Hydrogen reaction rate degrades structure.	MODERATE Because defects/cycles induce accelerated deterioration. High Graphite-Hydrogen reaction rate degrades primary structure in forward end. Aft end deterioration rate is moderated by presence of carbide in the structure.	MODERATE/LOW Because defects/cycles induce accelerated deterioration. High Graphite-Hydrogen reaction rate degrades primary structure in forward end. Aft end deterioration rate is moderated by presence of carbide in the structure.	HIGH/MODERATE Because defects/cycles induce moderate deterioration. Deterioration rate due to Graphite-Hydrogen reaction is moderated because of carbide in the structure.	HIGH Because no coating required therefore no defects. No free carbon available.	HIGH/MODERATE Because defects/cycles induce moderate deterioration. Deterioration rate due to Graphite-Hydrogen reaction is moderated because of carbide in the structure.	HIGH/MODERATE Because defects/cycles induce moderate deterioration. Deterioration rate due to Graphite-Hydrogen reaction is moderated because of carbide in the structure.	LOW/MODERATE Prone to accelerated deterioration due to cyclic effects. High Graphite-Hydrogen reaction rate degrades structure.	
	COMBINED STRESS (Objective: Minimize combined stresses consistent with thermal, friction and coating-matrix interaction effects)	(.055)	(.091)	(.091)	(.164)	(.127)	(.018)	(.145)	(.182)	(.127)	
	2.36 Estimated Maximum Stress*	2543 psi at Sta. 36	2042 psi at Sta. 36	2005 psi at Sta. 35	1777 psi at Sta. 35	4272 psi at Sta. 36	16040 psi at Sta. 36	4199 psi at Sta. 36	3630 psi at Sta. 41	1867 psi at Sta. 61	
	Allowable Stress (.8 x 2.36 Min. Tensile Stress)	2200 psi	1950 psi	1950 psi	1940 psi	4350 psi	5300 psi	4350 psi	4530 psi	1970 psi	
	Q Maximum Nominal (Required)	66.4 BTU/Inch ³	66.4 BTU/Inch ³	66.4 BTU/Inch ³	72.5 BTU/Inch ³	66.4 BTU/Inch ³	75.3 BTU/Inch ³ No coating, no friction lockup.	66.4 BTU/Inch ³	58.7 BTU/Inch ³	45.4 BTU/Inch ³	
	Margin = $\frac{\text{Allowable stress}}{2.3 \text{ Max. stress}} - 1$ of Safety	-.13	-.045	-.032	+.092	+.018	-.670	+.037	+.125	+.053	
	SENSITIVITY TO BUNDLING FRICTION** (Objective: Minimize stress contribution of bundling)	(.128)	(.107)	(.085)	(.085)	(.170)	(.043)	(.170)	(.148)	(.064)	
	30 psi bundling and 81°R (2.3 ΔT element to element) at maximum combined stress. Peak friction/allowable stress at point of maximum combined stress.	.227	.256	.280	.280	.103	.623	.103	.126	.370	

NOTES:

* Coating assumed stress free at ZrC deposition temperature

** Also a measure of sensitivity to transverse temperature difference

NOMENCLATURE TABLE 3-2 (Cont'd.) PERFORMANCE CONSIDERATIONS FUEL ELEMENTS CP-677566		<div> ASTRONUCLEAR LABORATORY COMPARISON MATRIX OF DESIGN APPROACHES </div>									SELECTION
FUNCTIONAL & TECHNICAL DESIGN REQUIREMENTS		1 Standard Geometry Intermediate Expansion Graphite	2 Standard Geometry High Expansion Graphite	3 Standard Geometry Hybrid	37 Hole Hybrid	Standard Geometry Composite	Single Channel Carbide	7 Zigzag Composite	8 High Void Composite Long Core	9 Standard Geometry Graphite Large Core	
STRUCTURAL INTEGRITY (Cont'd.)	RADIAL EXPANSION (Objective: Maximum expansion equal to or less than 1.66%)	.134	.110	.110	.110	.110	.096	.110	.110	.110	8; 7; 5; 6; 3 9; 4; 2; 1
	Nominal Maximum Expansion At Position of Maximum Fuel Temperature	1.146%	1.48%	1.54%	1.505%	1.54%	1.62%	1.54%	1.53%	1.44%	
	Margin	.519%	.185%	.125%	.160%	.125%	.045%	.125%	.135%	.225%	
	NUCLEAR CRITICALITY (Objective: Sufficient fuel loading to sustain controlled nuclear fission with the required +\$.00 reactivity shimming.)	.069	.069	.138	.034	.138	.104	.138	.138	.172	
	Maximum Loading Required mg/cc	621	630	520	618	579	1200	579	575	425	
	Maximum Loading Based on Present Processes mg/cc	630	630	630	630	700	~ 2000	700	700	630	
	MARGIN (without uncertainties) Loading Limit - Max. Loading Max. Loading	1.4%	0%	21%	1.9%	38%	67%	21%	22%	48%	
	Uncertainties	LOW	LOW	INTERMEDIATE	INTERMEDIATE	INTERMEDIATE	HIGH	INTERMEDIATE	INTERMEDIATE	LOW	
	PERFORMANCE FIGURE OF MERIT	.085	.087	.108	.101	.129	.118	.132	.133	.107	

NOMENCLATURE		TABLE 3-3		<div><div><div>W</div></div></div> <div>ASTRONUCLEAR LABORATORY</div> <div>COMPARISON MATRIX OF DESIGN APPROACHES</div>							SELECTION
		1	2	3	4	5	6	7	8	9	
FUNCTIONAL & TECHNICAL DESIGN REQUIREMENTS		Standard Geometry Intermediate Expansion Graphite	Standard Geometry High Expansion Graphite	Standard Geometry Hybrid	37 Hole Hybrid	Standard Geometry Composite	Single Channel Carbide	Zigzag Composite	High Void Composite Long Core	Standard Geometry Graphite Large Core	
OBJECTIVE: THE WEIGHT SHALL BE MINIMIZED											
	Core Weight (Fuel Envelope)	3,660 lbs.	3,645 lbs.	4,125 lbs.	4,060 lbs.	5,515 lbs.	8,445 lbs.	5,515 lbs.	6,300 lbs.	6,085 lbs.	
	Shield Weight	3,850 lbs.	3,897 lbs.	3,897 lbs.	3,897 lbs.	3,638 lbs.	3,630 lbs.	3,638 lbs.	3,544 lbs.	3,819 lbs.	
	Reflector Weight	3,790 lbs.	3,790 lbs.	3,790 lbs.	3,790 lbs.	3,790 lbs.	3,790 lbs.	3,790 lbs.	4,520 lbs.	6,060 lbs.	
	Core, Shield & Reflector Weight	11,300 lbs.	11,332 lbs.	11,812 lbs.	11,747 lbs.	12,943 lbs.	15,865 lbs.	12,943 lbs.	14,364 lbs.	15,964 lbs.	
	Weight Above Alternate 1	Reference	32 lbs.	512 lbs.	447 lbs.	1,643 lbs.	4,565 lbs.	1,643 lbs.	3,064 lbs.	4,664 lbs.	

NOMENCLATURE TABLE 3-4

SAFETY CONSIDERATIONS
FUEL ELEMENTS
CP-677566

ASTRONUCLEAR LABORATORY
COMPARISON MATRIX OF DESIGN APPROACHES

SELECTION

FUNCTIONAL & TECHNICAL DESIGN REQUIREMENTS	1 Standard Geometry Intermediate Expansion	2 Standard Geometry High Expansion	3 Standard Geometry Hybrid	4 37 Hole Hybrid	5 Standard Geometry Composite	6 Single Channel Carbide	7 Zigzag Composite	8 High Void Composite Long Core	9 Standard Geometry Graphite Large Core
<p>OBJECTIVE: TO INSURE SAFETY CAPABILITY THROUGHOUT THE LIFE OF FUEL ELEMENTS</p> <p>MANUFACTURING, HANDLING & STORAGE SAFETY (Objective: Provide for personnel safety by precluding accidental criticality through ease of poisoning.)</p> <p>STEADY STATE OPERATION SAFETY (Objective: Minimize fission product release.)</p> <p>SAFETY UNDER ACCIDENT CONDITIONS (Objective: Maximize temperature capability.)</p>	<p>(.175)</p> <p>Ease of poisoning of .102 diameter flow channels proven in NRX-A program. Reference case.</p> <p>(.223)</p> <p>In first 50 minutes of operation .3% of gross gamma and 1% of I_{131} are released. Double the above amount is released at last 50 minutes due to corrosion. Reference case.</p> <p>(.146)</p> <p>Sublimation at 5800°R Reference time at temperature.</p>	<p>(.175)</p> <p>Ease of poisoning of .102 diameter flow channels proven in NRX-A program. Reference case.</p> <p>(.223)</p> <p>In first 50 minutes of operation .3% of gross gamma and 1% of I_{131} are released. Double the above amount is released at last 50 minutes due to corrosion. Reference case.</p> <p>(.146)</p> <p>Sublimation at 5800°R Reference time at temperature.</p>	<p>(.150)</p> <p>Ease of poisoning diminished by stack-up tolerance at Sta. 35.0. Increased quantity of poison wires required.</p> <p>(.044)</p> <p>10 times the amount of reference case are released due to absence of pyro coating surrounding fuel particles. No change with time.</p> <p>(.098)</p> <p>Sublimation at 5800°R Eutectic 5400°R. Increased time at temperature.</p>	<p>(.075)</p> <p>Ease of poisoning diminished by stack-up tolerance at Sta. 35.0. Increased quantity of poison wires required.</p> <p>(.044)</p> <p>10 times the amount of the reference case are released due to absence of pyro coating surrounding fuel particles. No change with time.</p> <p>(.098)</p> <p>Sublimation at 5800°R. Eutectic 5400°R. Increased time at temperature.</p>	<p>(.125)</p> <p>Increased quantity of poison wires required because of harder spectrum.</p> <p>(.044)</p> <p>10 times the amount of reference case are released due to absence of pyro coating surrounding fuel particles. No change with time.</p> <p>(.098)</p> <p>Sublimation at 5800°R Eutectic 5400°R. Increased time at temperature.</p>	<p>(.025)</p> <p>Ease of poisoning diminished by stack-up tolerance of many small pieces and smaller diameter of flow channels. Increased quantity of poison wires required because of harder spectrum.</p> <p>(.111)</p> <p>About 5 times the amount of the reference case. Recoil path length ends in carbide which has slower diffusion. No change with time.</p> <p>(.024)</p> <p>Eutectic 5400°R Min. time at temperature.</p>	<p>(.125)</p> <p>Increased quantity of poison wires required because of harder spectrum.</p> <p>(.044)</p> <p>10 times the amount of reference case are released due to absence of pyro coating surrounding fuel particles. No change with time.</p> <p>(.098)</p> <p>Sublimation 5800°R Eutectic 5400°R Increased time at temperature.</p>	<p>(.050)</p> <p>Ease of poisoning increased by 5% larger flow channels diameter and diminished by increased fuel length. Increased quantity of poison wires required because of harder spectrum.</p> <p>(.044)</p> <p>10 times the amount of reference case are released due to absence of pyro coating surrounding fuel particles. No change with time.</p> <p>(.122)</p> <p>Sublimation at 5800°R Eutectic 5400°R Increased time at temperature.</p>	<p>(.100)</p> <p>Ease of poisoning diminished by increased fuel length.</p> <p>(.223)</p> <p>In first 50 minutes of operation .3% of gross gamma and 1% of I_{131} are released. Double the above amount is released at last 50 minutes due to corrosion. Reference case.</p> <p>(.170)</p> <p>Sublimation at 5800°R with increased time at temperature.</p>

W59778-1

NOMENCLATURE		TABLE 3-4 (Cont'd.)									SELECTION
SAFETY CONSIDERATIONS FUEL ELEMENTS CP-677566		ASTRONUCLEAR LABORATORY COMPARISON MATRIX OF DESIGN APPROACHES									
FUNCTIONAL & TECHNICAL DESIGN REQUIREMENTS		1 Standard Geometry Intermediate Expansion Graphite	2 Standard Geometry High Expansion Graphite	3 Standard Geometry Hybrid	4 37 Hole Hybrid	5 Standard Geometry Composite	6 Single Channel Carbide	7 Zigzag Composite	8 High Void Composite Long Core	9 Standard Geometry Graphite Large Core	
SAFETY UNDER EMERGENCY MODE OF OPERATION (Objective: Maximize tempera- ture margin to provide latitude for avoiding emergency mode operation.)		.176	.176	.059	.118	.059	.059	.059	.088	.206	
36 Fuel Temperature		4811°R	4811°R	4822°R	4726°R	4822°R	4817°R	4822°R	4782°R	4721°R	
Limiting Temperature		Sublimation Tempera- ture 5800°R.	Sublimation Temper- ature 5800°R.	Eutectic Temperature 5400°R.	Eutectic Temperature 5400°R	Eutectic Temperature 5400°R.	Eutectic Temperature 5400°R.	Eutectic Temperature 5400°R.	Eutectic Temperature 5400°R.	Sublimation Tempera- ture 5800°R.	
MARGIN		989°R	989°R	578°R	674°R	578°R	583°R	578°R	618°R	1079°R	
SAFETY FIGURE OF MERIT		.181	.181	.087	.084	.081	.055	.081	.076	.174	1 & 2; 9; 3; 4; 5 & 7; 8;6

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NOMENCLATURE		TABLE 3-5 (Cont'd.) DEVELOPMENT AND QUALIFICATION CONSIDERATIONS FUEL ELEMENTS CP-677566								SELECTION	
		<div>ASTRONUCLEAR LABORATORY</div> <div>COMPARISON MATRIX OF DESIGN APPROACHES</div>									
FUNCTIONAL & TECHNICAL DESIGN REQUIREMENTS		1 Standard Geometry Intermediate Expansion Graphite	2 Standard Geometry High Expansion Graphite	3 Standard Geometry Hybrid	37 Hole Hybrid	Standard Geometry Composite	Single Channel Carbide	7 Zigzag Composite	8 High Void Composite Long Core	9 Standard Geometry Graphite Large Core	
DEVELOPMENT OF SURVEIL- LANCE TECHNIQUES (NDT AND CORROSION TESTS)		(.167) Uses existing techniques on graphites.	(.167) Uses existing tech- niques on graphites.	(.100) Development of cor- rosion testing tech- niques required for hybrid portion of the element.	(.033) Development of corro- sion testing techniques required for hybrid portion of the element. Additional corrosion testing development required for geometry change.	(.133) Development of corro- sion testing techniques for composite required.	(.083) Requires completely new testing and sur- veillance techniques due to geometry and material.	(.050) Handling and corrosion testing require develop- ment due to geometry.	(.117) Handling and corrosion testing require develop- ment due to length.	(.150) Handling and corro- sion testing require development due to length.	
JOINT INTEGRITY (Objective: Joint leaks equal or less than matrix and maximized joint strength.)		(.088)	(.088)	(.112)	(.112)	(.125)	(.125)	(.125)	(.125)	(.100)	
36 Joint Temperature		~ 4710°R	~ 4710°R	~ 4480°R	4400°R					4620°R	
Joint Temperature Capability		4800°R after 10 x 7.5 minute cycles.	4800°R after 10 x 7.5 minute cycles.	4800°R after 10 x 7.5 minute cycles.	4800°R after 10 x 7.5 minute cycles.	NO JOINT	NO JOINT	NO JOINT	NO JOINT	4800°R after 10 x 7.5 minute cycles.	
MARGIN		+90°R	+90°R	+320°R	+400°R					+180°R	
DEVELOPMENT AND QUALI- FICATION FIGURE OF MERIT		.141	.141	.103	.079	.126	.070	.088	.117	.135	1 & 2; 9; 5; 8; 3; 7; 4; 6

NOMENCLATURE TABLE 3-6 SCHEDULE CONSIDERATIONS FUEL ELEMENTS CP-677566		<div>W</div> ASTRONUCLEAR LABORATORY COMPARISON MATRIX OF DESIGN APPROACHES									SELECTION
FUNCTIONAL & TECHNICAL DESIGN REQUIREMENTS		1 Standard Geometry Intermediate Expansion Graphite	2 Standard Geometry High Expansion Graphite	3 Standard Geometry Hybrid	4 37 Hole Hybrid	5 Standard Geometry Composite	6 Single Channel Carbide	7 Zigzag Composite	8 High Void Composite Long Core	9 Standard Geometry Graphite Large Core	
OBJECTIVE: MEET R-1 TEST DATE OF 12/31/72											
FACILITIES AND CAPACITY		Available (including Y-12) (In process of change over to composite)	Available (including Y-12) (In process of change over to composite)	Side by side facilities of graphite and composite required.	Side by side facilities of graphite and composite required.	Change to facilities required (6-9 months away).	No facilities available. Different type facilities required. Major changes required. LASL doesn't plan to go directly to carbide.	Tooling change required. Change in facilities required. Similar to requirements for design concept 5.	Facilities change as required for design concept 5. Major changes required in furnaces.	Similar changes as required for design concept 5. Increased length requires major changes in furnaces.	
MATERIALS		Qualified Sources Available	Qualified Sources Available	Combination of No. 1 and No. 5.	Combination of No. 1 and No. 5.	Order placed for graphite. ZrC-UO ₂ procurement under negotiation. Must develop and qualify supplier. Planned by March or April 1970.	Use of and extension of No. 5.	Use of and extension of No. 5.	Similar to No. 5.	Qualified Sources Available	
PROCESSES		Qualified Processes Exist	Qualified Processes Exist	Qualified process exists for the forward end of element. For the aft end of the element some problem expected in heat treatment and die procurement.	New dies required. Geometry difficult to extrude (some extrusion development required). New fixtures of long lead times required.	Some problem expected in heat treatment and die procurement.	Wet machining requires major development. Difference in extrusion requires development.	Some problem expected in heat treatment and die procurement. New trays and handling equipment required. Machining of outside is of major concern.	Some problem expected on heat treatment and die procurement. New fixtures required.	Qualific Processes Exist	
QUALITY CONTROL		Demonstrated capability exists.	Demonstrated capability exists.	Demonstrated capability exist for the forward end of element. New standards are required. Transfer of LASL technology expected.	Problem expected because new processes need to be developed.	New standards required. Transfer of LASL technology expected.	Completely new techniques required.	No. 5 and some handling problems such as chipping edges expected.	New standards required. Transfer of LASL technology expected.	Demonstrated capability exists. New tools required.	
SCHEDULE FIGURE OF MERIT		.167	.167	.100	.067	.132	.017	.083	.117	.150	1 & 2; 9; 5; 8; 3; 7; 4; 6

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NOMENCLATURE		TABLE 3-7		<div><div> </div><div>ASTRONUCLEAR LABORATORY</div><div>COMPARISON MATRIX OF DESIGN APPROACHES</div></div>								SELECTION
FUNCTIONAL & TECHNICAL DESIGN REQUIREMENTS		WEIGHTING FACTOR	1	2	3	4	5	6	7	8	9	
			Standard Geometry Intermediate Expansion Graphite	Standard Geometry High Expansion Graphite	Standard Geometry Hybrid	37 Hole Hybrid	Standard Geometry Composite	Single Channel Carbide	Zigzag Composite	High Void Composite Long Core	Standard Geometry Graphite Large Core	
RELIABILITY		.30	.01740	.02970	.03420	.04080	.03450	.03180	.03450	.04290	.03420	2; 4; 3; 5; 1 7; 8; 9; 6
PERFORMANCE		.25	.02125	.02175	.02700	.02525	.03225	.02950	.03300	.03325	.02675	
WEIGHT		.25	.04300	.04300	.03875	.03875	.03050	.00425	.03090	.01700	.00425	
SAFETY		.10	.01810	.01810	.00870	.00840	.00810	.00550	.00810	.00760	.01740	
DEVELOPMENT AND QUALIFICATION		.05	.00705	.00705	.00515	.00395	.00630	.00350	.00440	.00585	.00675	
SCHEDULE		.05	.00835	.00835	.00500	.00335	.00660	.00085	.00415	.00585	.00750	
	FIGURE OF MERIT		.11515	.12795	.11880	.12050	.11825	.07540	.11465	.11245	.09685

4.0 SELECTION OF DESIGN (U)

- (U) The numerical results of evaluating the design concepts of Fuel Element are inconclusive because of small differences between the values of Summary Figures of Merit.
- (U) The first four design concepts vary only by approximately 8%. Therefore, the selection of concepts for continued design and development cannot be based purely on numerical ranking.
- (U) The numerical results of this trade study do not run counter to the choice of Fuel Element as directed by SNPO; namely, Standard Geometry, High Expansion Graphite Fuel Element and Standard Geometry, Composite Fuel Element.
- (U) These two design concepts have Figure of Merit which place them in the first and fourth place among the evaluated designs.
- (U) The choice of design concepts for continued design and development are: Standard Geometry, High Expansion Graphite (Concept 2) and Standard Geometry, Composite (Concept 5).
- (C-RD) The reasons for choosing the Standard Geometry, High Expansion Graphite design were high degree of certainties of the properties and behavior and high probability of meeting the schedular requirements. The reasons for preference of the Standard Geometry, Composite over the 37 Hole Hybrid (which has a composite aft end) was the schedular and developmental penalty combined with the larger uncertainty of both the structural and corrosion performance of this element. This uncertainty results from possible, but unknown, effects of the new geometry extrusion process on the strength and/or other properties especially of the 37 hole composite hot end of the fuel, as well as the uncertainty of extrapolating 19 hole element corrosion test performance to a 37 hole element on an analytical basis. An additional reason for the choice is the flexibility the Standard Geometry Graphite element and Composite element by providing the components for the development of the Standard, Hybrid Element, should future development results dictate.

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APPENDIX A-1

FUEL ELEMENT AND CORE SIZE PARAMETRIC STUDY TRADE STUDY FOR NERVA FUEL ELEMENT DESIGN CONCEPTS TRADE STUDY NO. 759

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A-1.1 (U) INTRODUCTION (U)

This appendix presents the parametric study leading to selection of the alternate fuel element and core designs in the trade study. The parametric study of the design variables affecting fuel performance led to the selection of new element geometries and core sizes for the more detailed comparisons in the body of this report. Although one of the objectives of this trade study was to explore the effects of changes in fuel element geometry, fuel element materials and core size on the thermal stress due to internal heat generation, Section A-1.3 also considers the effect of these changes on other stress components. This was necessary because internal heat generation is only one of the several stress producing mechanisms which are affected by these changes. The section also covers the influence of modification of the fueled core design on central element, support stem and hot end hardware stresses. Section A-1.4 contains, in addition to the parametric curves, details on the fuel element temperature uncertainties, maximum material temperature limits, effects of axially varying void fraction and coating thickness and the effect of changes in core diameter and length on support stem flow control.

In this introduction to Appendix A-1, the objectives, requirements, assumptions and variables of the parametric study are given. Results are summarized in Section A-1.2.0. Supporting technical information is contained in the next four subsections. Section A-1.3.0 discusses structural capability, Section A-1.4.0 contains thermal and corrosion information, the nuclear evaluation is contained in Section A-1.5.0 and mechanical design considerations are given in Section A-1.6.0. A brief evaluation of the feasibility of using an increased number of support stems is contained in Section A-1.7.0. Section A-1.8.0 presents the effect of increasing core size on the weight of the nuclear subsystem. The parametric data given in Sections A-1.3.0 through A-1.6.0 is combined into a series of cross plots in Section A-1.9.0 from which some of the fuel alternates were selected.

(U) A-1.1.1 Objective of the Parametric Study (U)

The objective of this study was to parametrically develop the factors which affect fuel performance as a function of fuel and central element geometry and core length and

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radius. Alternate designs were then selected on the basis of thermal stress, core pressure drop, fuel corrosion, fuel uranium loading, NSS weight, and other considerations. Under the assumptions of the analysis, some of the fuel element designs have significantly reduced thermal stress in comparison to the reference R-1 fuel.

A-1.1.2 (CRD) Assumptions, Requirements or Fixed Values for the Parametric Study (U)

1. The basic reactor arrangement for R-1 is unchanged; that is, a full flow cycle reactor with R-1 type support plate, support stems and hot end hardware, lateral support system and reflector.
2. A support stem reactivity flow control requirement of \$2.00 with a goal of \$3.00 at rated conditions.
3. Basic material properties and residual stresses from manufacturing are not affected by changes in fuel element geometry. (This is a questionable assumption, but there is no alternative.)
4. A 10-hour, sixty cycle goal at design conditions.
5. Nominal state point conditions as follows: $T_c = 4250^\circ R$, $P_c = 450$ psia, thrust = 75,000 lbs, core flow = 89.9 lb/second.
6. The fuel element is a hexagonal solid matrix with cylindrical axial flow paths (i.e., pebble bed, rod bundle, and other core geometries were not considered).
7. The minimum effective hydraulic impedance diameter for coated bores was assumed to be the nominal reamed diameter minus twice the average nominal coating thickness minus an additional 4.1 mils. The 4.1 mils is intended to account for dimensional tolerances in reamed diameter and coating thickness and also roughness effects of the coating.

A-1.1.3 (CRD) Independent Variables of the Parametric Study (U)

(CRD) The following were varied independently (as far as possible) over the range indicated.

1. Core volume varied from the reference R-1 volume to twice the reference R-1 volume.
2. Core length varied from 40 to 100 inches.

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3. Core fueled diameter varied from reference R-1 diameter of 37.3* inches to 54 inches.
 4. Fuel channel uncoated diameter varied from 0.060 inches to 0.120 inches.
 5. Centerline - centerline hole pitch varied from 0.130 inches to 0.180 inches.
 6. Number of holes per element varied from 1 to 19 to 37, depending on dimensional limits and other requirements.
 7. Fuel element across flats dimension varied from 0.60 inch to 1.5 inches.
 8. Two support stem arrangements were considered, the reference R-1 arrangement of one unfueled element and stem for every six fuel elements (excluding periphery) and an arrangement with one unfueled element and stem for every three fuel elements.
 9. The average coating thickness profile was varied from 2.2**mils to 8 mils.
 10. Matrix materials considered were graphite (both high and intermediate expansion) 30 v/o composite (UC • ZrC - C) and solid solution UC • ZrC carbide.

A-1.1.4 (CRD) Dependent Variables (U)

(CRD) The following dependent variables were calculated as functions of the independent variables listed in Section A-1.1.3. All of these are discussed in detail in the respective technical sections. Where applicable, region boundaries, or limits for the parametric cross plots were established in the technical section.

1. Core pressure drop limit with six fuel elements per stem was established in Section A-1.6.
2. Core pressure drop limit with three fuel elements per stem was established in Section A-1.7.

* This diameter is the reference R-1 size including 1878 fuel elements and 349 unfueled elements. The ratio of fueled-to-unfueled is 5.38. Using a fueled-to-unfueled ratio of 6.0 with 1878 fueled elements results in a core fueled diameter of 37.0 inches.

** Corresponds to the average thickness obtained on 72 PW-2 GEM NbC - HED ZrC coated elements.

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3. Maximum uranium loading limits for 30 v/o composite and graphite fuel were established in Section A-1.4.

4. Fuel element cross flats limits were established in Section A-1.6.

5. The fuel element minimum web thickness limit for both graphite and composite fuel was established in Section A-1.6.

6. The thermal stress due to internal heat generation is discussed in Section A-1.3 and A-1.9 and is to be minimized.*

7. Fuel temperatures are discussed in Section A-1.4 and are to be minimized.

8. The reactivity loss due to corrosion is discussed in Section A-1.4 and is to be minimized.

9. The nuclear subsystem weight is discussed in Section A-1.8 and is to be minimized.

* Other stress components are analyzed in Section A-1.3 but were not put on a parametric basis. In selecting fuel element alternates to be included in the trade study, only the stress component due to internal heating was considered.

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A-1.2 (CRD) SUMMARY AND CONCLUSIONS (U)

(U) Within the requirements, assumptions and ground rules of this parametric study, as given in the introduction, alternate fuel element designs and core sizes were selected leading to a total of nine fuel element design alternates for comparison. These are listed in Table A-1.2-1. In selecting the alternates, the core pressure drop, maximum uranium loading and element dimensions were held within limits established by the technical sections.

(CRD) Maximum uranium loadings of graphite and 30 volume percent composite fuel were limited to less than 630 mg/cc and 700 mg/cc respectively. These limits were established based on previous experience and projected loading effects on material properties.

(CRD) A minimum allowable across flats dimension of 0.69 inches was set for both composite and graphite 19 and 37 hole hexagonal fuel elements. Consideration of the minimum spacing between support stems in the core support plate led to this limit.

(CRD) A maximum core pressure drop of 170 psid was established for the parametric study with both composite and graphite fuel elements having an average carbide coating thickness equal to that obtained on 72 PW-II elements with GEM NbC - HED ZrC coatings. This pressure drop limit permits some flexibility in choosing coating thickness profiles and is shown to allow some additional flexibility by tapering the reamed diameter larger at the hot end. The 170 psid limit was set for the R-1 cluster arrangement with six fuel elements per central element and support stem. Although use of a support element for every three fuel elements would appear to increase the core pressure drop capability by 75%, it was found that an R-1 type support plate would have an excessively high pressure drop and this arrangement was not considered further.

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(U) Selection of the alternate fuel element designs was made within these limits primarily on the basis of minimizing the thermal stress component caused by the temperatures resulting from internal heat generation within the element. For a given fuel matrix material this stress component is directly proportional to a "stress function", which was, therefore, minimized.

(CRD) Figure A-1.2-1 gives this stress function versus the 19 hole fuel element across flats dimension for four reamed diameters from 0.097" to 0.112 inches. The region in which the core pressure drop limit is exceeded is indicated as is the region in which the graphite element loading limit is exceeded. It can be seen that the standard geometry for alternates 1 and 2 gives the minimum stress function within the limits established. This is a 0.754" 19 hole graphite element with a 0.102" reamed diameter.

(CRD) The next figure, A-1.2-2, is the same as the previous figure but with the core pressure drop and loading limits of the 19 hole 30 v/o composite element. There is more flexibility available in the selection of the fuel geometry for this matrix material. The square on the figure shows that Alternate 3, a hybrid composite/graphite element; Alternate 5, a composite element; and Alternate 7, a variation of Alternate 5 with a zig-zag boundary have the same 0.754" across flats and 0.102" reamed diameters. These 19 hole elements were selected with these dimensions to obtain a direct comparison to the standard graphite element and because of the manufacturing and test experience which has been obtained with these element dimensions.

(CRD) Examination of the figure shows that the stress function could be reduced to a minimum of about 0.051 with a 0.69" element size. However, approximately 50% of this gain could be made without changing the element across flats dimension from the standard 0.754", as indicated by the circle on the figure. It was considered advisable to forego this additional reduction in stress function in order to retain the 0.754" element size. Alternate 8 is a 19 hole composite element with a 0.108 inch reamed diameter. Since this alternate

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is not close to the core pressure drop limit, the core length was increased to 62 inches.

This increase in core length further reduced the stress function by approximately 16%.*

(CRD) Selection of the 37 hole element is shown by Figure A-1.2-3. The loading limit will be exceeded by a small amount with a full length composite element. The selected Alternate 5 is a 0.799" hybrid composite/graphite 37 hole element with a 0.084 inch reamed diameter. This geometry was selected to maintain the possibility of using a full length composite.

(CRD) Figure A-1.2-4 gives the stress function versus fueled radius with constant length for a standard cross-section graphite 19 hole fuel element. The parametric evaluation showed that both the weight penalty and the lateral support design problems are less when core volume is increased by increasing length rather than radius. Therefore, the length change was maximized in the selection of Alternate 9 core dimensions of 76" length and 20.3" fueled radius.

(U) Selection of the single channel carbide element (Alternate 6) is described in Section A-1.9.4, and Figure A-1.9-11 shows the limits which affect this design (similar to Figures A-1.2-1 through A-1.2-4).

* The increase in core length results in reduced fuel uranium loading. Therefore, it would have been possible to select an even larger reamed diameter and longer core.

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Figure A-1.2-1: (CRD) SELECTION OF GRAPHITE 19 HOLE ALTERNATES IN REFERENCE CORE SIZE (U)

FIGURE A-1.2-2: (CRD) SELECTION OF COMPOSITE 19 HOLE ALTERNATE (U)

FIGURE A-1.2-3: (CRD) SELECTION OF 37 HOLE ELEMENT REFERENCE CORE SIZE

FIGURE A-1.2-4: (CRD) SELECTION OF LARGE CORE ALTERNATE (U)

TABLE A-1.2-1

(CRD) LIST OF ALTERNATES FOR FUEL ELEMENT CONCEPTUAL DESIGN TRADE STUDY (U)

Alternate	No. Holes	Across Flats	Ream. Dia.	Matrix Material	Coating*		Core Dimensions		Hole Pitch
					Thickness Mils.	Material	Dia. in.	Length in.	
1. Std. Geom.-Int. Exp. Graph	19	0.753	0.102	GLFI	2.2	75gNb, ZrC	37.1	52	0.173
2. Std. Geom.-Hi Exp. Graph.	19	0.753	0.102	AXM	2.2	75gNb, ZrC	37.1	52	0.173
3. Std. Geom.-Hybrid	19	0.753	0.102	35"AXM, 17" Comp.	2.2	ZrC	37.1	52	0.173
4. 37 Hole - Hybrid	37	0.799	0.084	35" AXM, 17" Comp.	2.2	ZrC	37.1	52	0.1309
5. Std. Geom.-Composite	19	0.753	0.102	Composite	2.2	ZrC	37.1	52	0.173
6. Single Channel-Carbide	1	0.170	0.120	ZrC-UC		None	37.1	52	0.170
7. Zig Zag-Composite	19	0.753**	0.102	Composite	2.2	ZrC	37.1	52	0.173
8. Hi Void-Comp.-Long Core	19	0.753	0.108	Composite	2.2	ZrC	37.1	62	0.173
9. Std. Geom.-Graph-Large Core	19	0.753	0.102	AXM	2.2	84%NbC	40.6	76	0.173

* The thickness given is the mean thickness over the entire length.

** This is the distance between the "Mean Line" of the two opposite sides.

A-1.3 STRUCTURAL DESIGN CONSIDERATIONS

A-1.3.0 INTRODUCTION

Section A-1.3 begins with a discussion of the stress mechanisms affecting the NERVA fuel elements and the factors, such as geometry, material types and their properties, nuclear heating, hydraulic loads, etc., which influence these stress mechanisms. Combined stress analyses under the NERVA full power reactor environment are presented for a number of fuel materials in several geometries. This section concludes with support considerations, involving central element, support stem, and hot end hardware structural implications. Numerical data presented in Appendix 5 are drawn from analyses discussed in some detail in this section.

A-1.3.1 Fuel

A-1.3.1.1 Fuel Element Stress Mechanisms

Transverse Stress - The transverse stress in a reactor fuel element is obtained from the summation of stresses caused by internal heat generation, external bundling pressure on the element, the mechanical interaction between the bore coating and the matrix, and the mechanical interaction between the fuel bead and the matrix. Optimizing the transverse fuel element stress to meet a specified minimum strength involves optimizing each stress mechanism separately.

1) Internal Heat Generation Stress

The maximum steady state heat generation stress can be written as

$$\sigma_{q''' } = \frac{E\alpha}{K} \cdot q''' \cdot \psi$$

where

$$\begin{aligned} \psi &= F(L^2) \text{ Geometric shape factor} \\ \sigma_{q''' } &= \text{Maximum thermal stress} \\ E &= \text{Modulus of elasticity} \\ \alpha &= \text{Coefficient of thermal expansion} \\ K &= \text{Thermal conductivity} \\ q''' &= \text{Heat generation rate} \\ L &= \text{Characteristic length} \end{aligned}$$

Optimizing the heat generation stress for a given q''' involves selecting the optimal material (minimal $E\alpha/K\sigma_{ult}$) and selecting a geometry with a minimal shape factor " ψ ".

2) Bundling Pressure Stress

The transverse stresses caused by bundling pressure are compressive and negligible when compared to heat generation stress.

3) Coating-Matrix Interaction Stress

The coating-matrix interaction stress can be written as

$$\sigma_{c \cdot m} = E_m \left[-\alpha_m \Delta T + \Delta \alpha T_c - \Delta \alpha_d T_d \right] \cdot \phi$$

where

$\sigma_{c \cdot m}$	=	Coating-matrix interaction stress in the matrix
E_m	=	Matrix modulus of elasticity at the matrix average temperature T_m
α_m	=	Matrix coefficient of thermal expansion at temperature T_m
ΔT	=	Temperature difference between the matrix and the coating = $T_m - T_c$
T_m	=	Average matrix temperature
T_c	=	Average coating temperature
$\Delta \alpha$	=	Difference in coefficient of thermal expansion between matrix and coating = $\alpha_c - \alpha_m$
α_c	=	Coating coefficient of thermal expansion at temperature T_c
$\Delta \alpha_d$	=	Difference in coefficient of thermal expansion between matrix and coating at coating deposition temperature $T_d = \alpha_{cd} - \alpha_{md}$
T_d	=	Coating deposition temperature
$\emptyset$	=	Geometry and relative stiffness factor between coating and matrix

Optimizing the coating-matrix interaction stress involves both selecting compatible materials and a coating deposition temperature to minimize $\sigma_{c \cdot m}$. In fact, $\sigma_{c \cdot m}$ can be theoretically reduced to zero by a proper selection of coating deposition temperature.

4) Fuel Bead-Matrix Interaction Stress

The fuel bead-matrix interaction stress can be written as

$$\sigma_{fb} = E_m \left[\Delta \alpha (T_m - T_g) + \alpha_b \cdot \Delta T \right] \cdot \emptyset \text{ for } T_m \geq T_g$$

where

σ_{fb}	=	Fuel bead-matrix interaction stress in the matrix
E_m	=	Matrix modulus of elasticity at the matrix temperature T_m
$\Delta\alpha$	=	Difference in coefficient of expansion between bead and matrix at temperature $T_g = \alpha_b - \alpha_m$
T_g	=	Graphitization temperature
α_b	=	Bead coefficient of thermal expansion
ΔT	=	Difference between bead temperature and matrix temperature = $T_b - T_m$
θ	=	Geometry and stiffness factor

The fuel bead-matrix interaction stress in the matrix could be reduced to zero if it were possible, without damaging the beads, to graphitize at a temperature that is higher than the maximum bead temperature during reactor operation. The graphitization temperature is a stress-free temperature for fuel bead-matrix interaction. No effective fuel bead-matrix interaction occurs below the graphitization temperature.

Axial Stress - The axial stress in a reactor fuel element is obtained from the summation of stresses caused by frictional interaction between elements, thermal gradient across an element, internal heat generation, core pressure drop, coating-matrix interaction, and fuel bead-matrix interaction. Optimizing the axial stress in a reactor fuel element to meet a specified minimum strength involves optimizing each stress mechanism separately.

1) Friction Stress

The maximum (with 99% probability) axial friction stress is based on the interaction of a thermal stress caused by a 2.3σ interelement temperature difference and the maximum frictional stress that can be produced by a given bundling pressure and coefficient of friction. If no slipping occurs between elements, then the axial stress in a

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fuel element is $E\alpha\Delta T$ where ΔT is the 2.3σ interelement temperature difference. If slipping occurs and if the fuel element is unsupported at the hot end, then the axial stress in the fuel element is

$$\sigma_f = \int_x^{52} \mu P_b \cdot \frac{P}{A} dx$$

where

x = Element station

μ = Coefficient of friction

P_b = Bundling pressure

$\frac{P}{A}$ = Perimeter to cross-sectioned area ratio of fuel element

σ_f = Axial friction stress

The net axial stress is the smaller of these two functions.

2) Axial Thermal Gradient Stress

The axial fuel element stress that is caused by a temperature difference across the flats of a fuel element, assuming a linearly varying temperature, is $1/2 E\alpha\Delta T$ where ΔT = temperature difference across the flats.

3) Axial Coating, Fuel Bead and Matrix Interaction Stress

The axial stress mechanism for coating-matrix interaction and fuel bead-matrix interaction is the same as the transverse mechanism.

4) Core Pressure Drop Stress

The axial stress caused by core pressure drop is compressive in most cases and can be neglected for a conservative analysis of axial stress mechanisms.

5) Internal Heat Generation Stress

The axial stress caused by internal heat generation can be evaluated by using the same equations that are used in evaluation of transverse heat generation stresses.

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A-1.3.1.2 Variables Affecting Fuel Stress Mechanisms (U)

(U) Geometry - The geometric parameters that affect heat generation stress are hole diameter, hole pitch, external geometric shape and across flats dimension.

1) Hole Diameter and Pitch (U)

(U) The ratio of pitch to diameter and the absolute pitch affects the heat generation stress by its effect on the geometric shape factor ψ . Figures 3-1 and 3-2 show the relationship between pitch-diameter ratio (S/D), pitch (S) and ψ . For constant S/D ratios, the thermal stress is decreased by the square of the decrease in hole pitch.

2) External Geometric Shape (U)

(C-RD) The external geometric shape of a fuel element will probably not affect the thermal stresses. Elemental shapes such as the castellated, zig-zag, or spaghetti element do nothing more than replace an adiabatic line in a 19 hole element with a solid surface. The temperature and stress distribution will probably remain unchanged from that of a 19 hole element. There is relatively little difference in temperature and stress distribution between the 19 hole hexagonal elements and the true triangular pitch. The pitch on the 19 hole elements is purposely irregular so that the temperature and stress distribution is optimized at the level of the true triangular pitch. The 19 hole element has an assumed adiabatic external surface while the true triangular pitch is assumed to extend to infinity.

3) Across Flats Dimensions (U)

(C-RD) The thermal stress is affected by across flats dimension of a fuel element as shown in Figure 3-3. Thermal stress is plotted against across flats dimension for a 19 hole fuel element. In this plot three void fractions .27, .345, .47 corresponding to three pitch diameter ratios 1.84, 1.62, 1.39, respectively, were plotted. The plot shows that the fuel element thermal stress increases with across flats dimension for a constant void fraction and a constant pitch to diameter ratio.

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$$(1) \quad \text{THERMAL STRESS} = E \alpha (\bar{T} - T_{\text{MINIMUM}})$$

$$(2) \quad T_{\text{MAX}} - T_{\text{SURF}} = \frac{Q}{K} \Psi$$

$$\Psi = \left(\frac{S}{2}\right)^2 \sqrt{0.55133} \ln \left(\frac{S}{D}\right) + 0.25 \left(\frac{D}{S}\right)^2 - 0.234467$$

$$(3) \quad Q = \frac{Q_{\text{CORE TOTAL}} f_R f_A}{\text{CORE FUEL VOLUME}} = \frac{Q_{\text{CORE TOTAL}} f_R f_A}{\text{CORE SIZE} \times \text{FUEL FRACTION}}$$

$$(4) \quad \text{MAXIMUM THERMAL STRESS} = \left(\frac{E\alpha}{K}\right) \frac{Q_{\text{CORE TOTAL}}}{\text{CORE SIZE}} f_R f_A S^2 f\left(\frac{S}{D}\right)$$

$$(5) \quad \text{THERMAL STRESS PARAMETER} = Q\Psi$$

E	= MATRIX MODULUS OF ELASTICITY*
α	= MATRIX CTE*
K	= MATRIX THERMAL CONDUCTIVITY*
Q	= VOLUMETRIC HEAT GENERATION RATE
$Q_{\text{CORE TOTAL}}$	= TOTAL POWER GENERATED IN CORE FUEL
f_R	= RADIAL PEAK TO AVERAGE HEAT GENERATION RATE
f_A	= AXIAL PEAK TO AVERAGE HEAT GENERATION RATE
$f(S/D)$	= FUNCTION OF (S/D)

* $\frac{E\alpha}{K}$ varies with temperature level, but is treated as a constant for purposes of comparison.

Figure A-1.3-1. Thermal Stress Due to Steady State Heat Generation (U)

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Figure A-1.3-2 (CRD) Thermal Stress Function Versus Fuel Element Geometry (U)

Figure A-1.3-3. Effect of Fuel Element Parameters on Thermal Stress (Vs. Across-Flat Dimensions)

Material Properties (U)

1) Matrix (U)

(C-RD) The fueled matrices being considered for this parametric study are GLFI, AXM, 30 V/O UC-ZrC-C, 100 V/O UC-ZrC, and Figures A-1.3-4 through A-1.3-7, respectively, show the mechanical properties for these materials.

2) Fuel Bead (U)

(C-RD) The fuel bead properties are

	UC ₂	PyC Radial	PyC Longitudinal
$\alpha \frac{1}{R}$	7.8×10^{-6}	3.0×10^{-6}	$.83 \times 10^{-6}$
E psi	40×10^6	1.5×10^6	4.0×10^6
μ	0.3	0.3	-0.16
$K \frac{BTU}{in \ sec^{\circ}R}$	3.85×10^{-4}	$.0531 \times 10^{-4}$	

3) Coating (U)

(U) The coating is both NbC and ZrC. The properties for these two materials are shown in Figures 3-8 and 3-9.

(U) Coating Thickness and Deposition Temperature - The coating thicknesses and deposition temperatures are shown on Figure 3-10. The coating thickness does not influence matrix stress as severely as the coating deposition temperature (Reference DRM 52064 and DRM 51557).

(C-RD) Heat Generation Rate - The R-1 referenced heat generation rate at steady state for a 1.1 power factor is shown plotted in Figure 3-11. The R-1 referenced heat generation rate is for a 37.1 inch diameter and 52 inch length core.

(U) Drum Motion - There will be, nominally, no drum motion between the alternates in this trade study. Reactivity control will be about \$3.

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Figure A-1.3-4 Mechanical Properties - GLFI-550

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Figure A-1.3-5 Mechanical Properties - AXM-630 MGM/CC Uranium

(Reference DRM 51230A)

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Figure A-1.3-6 Mechanical Properties - 30 V/O UC-ZrC-C

(Reference Material Properties Group Work Sheet, Signed by J. S. Blay 12/12/69)

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Figure A-1.3-7 Mechanical Properties of 100 V/O UC-ZrC

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Figure A-1.3-8 NbC Coating Mechanical Properties
(Reference DRM 51221A)

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Figure A-1.3-10 Coating Thickness and Deposition Temperature
for R-1 Fuel Elements (GEM NbC and HED ZrC)
(Reference Drawing 711J596 & Telecom with C. Hollabaugh WNCO 12/9/69)

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Figure A-1.3-11 Wall Temperature & Heat Generation Rate for R-1 Fuel Elements - 1.1.P.F
(Reference DRM 52102)

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A-1.3.1.3 Operating Limits for Major Fuel Stress Mechanisms Based on
Fuel Type (U)

High Expansion AXM Stress (U)

Transverse Stress (U)

(C-RD) 1) The calculated AXM transverse stresses are plotted on Figure 3-12.

These stresses were calculated using a one-twelfth symmetry finite element model of the 19 hole reference hexagonal element in the finite element computer code FEATS. The stresses on Figure 3-12 are the plotted results of computer listing FDL-22. The flow channels were assumed to be .102 inches reamed diameter. The bores were assumed to be coated with ZrC over NbC as described on Drawing 711J596. The analysis is based on the maximum coating thickness and the coating deposition temperature profiles of Figure 3-10. The input heat generation rate and the thermal boundary temperatures used in the thermal analysis section of FEATS are shown on Figure 3-11. The stresses shown on Figure 3-12 are the maximum transverse tensile stresses. In general, the transverse matrix stresses are substantially less than a strength limit of 80% of the 2.3 minimum ultimate tensile strength. Only between stations 0 to 4 does the matrix transverse stress appear to exceed the strength limit. However, this stress is a radial stress at a flow channel and depends upon coating adherence. A loss of coating adherence is assumed to occur before this stress level is reached. Coating-matrix interaction stresses in the matrix are assumed to begin to occur at the coating deposition temperature of ZrC. It is assumed that the NbC relaxes sufficiently because of the time duration at the ZrC deposition temperature that the NbC and the matrix are at a zero stress state at the ZrC deposition temperature. Calculated LASL Q''' test parameters are shown plotted on Figures 3-13 and 3-14. These parameters are calculated maximum transverse stress and calculated surface temperature. The input parameters were heat generation rates of 50 BTU/in³ sec., 100 BTU/in³ sec. and a flow rate of 200 lb/hr. The model used in the analysis was a one-twelfth symmetry section finite element model of a 19 hole AXM reference element with .002 inches of NbC coating in .099 diameter reamed flow channels. Coating deposition temperature was assumed at 2500°R. The

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Figure A-1.3-12 R-1 10 Hour Fuel AXM-630 - Transverse Matrix Stress and Coating Stress
 $Q'''_{max} = 66.4 \text{ BTU/IN}^3\text{-SEC}$ 19 Hole Reference Hexagon

Figure A-1.3.13 Calculated Thermal Stress and Temperatures from LASL Thermal Stress Furnace Parameters

Figure A-1.3.14 Calculated Thermal Stress and Temperatures from LASL Thermal Stress Furnace Parameters

analysis indicates that the Q''' capability of AXM fuel is less than 50 BTU/in³ sec.

However, the thermal boundary conditions between the LASL Q''' test and reactor operation are sufficiently different from each other than LASL Q''' test results cannot be directly applied to reactor operation.

(C-RD) 2) The physical property analysis of AXM fuel indicates that a one sigma variation of 27.5% in heat generation thermal stress can be expected as a result of the variances in nuclear heating and physical properties and geometry. The mean values and geometric dimensions used in the analysis are

	<u>Mean Value</u>	<u>Std. Dev.</u>	<u>Std. Dev. % Mean</u>
Pitch	.1739 in.	.00782	4.5
Dia.	.102 in.	.0004347	.2
Q'''	53.6 BTU/IN ³ -SEC	1.63	3.0
$\bar{\alpha}$	$3.1 \times 10^{-6} / ^\circ R$	$.15 \times 10^{-6}$	4.8
E	2.57×10^6 psi	$.4 \times 10^6$	15.5
K	6.5×10^{-4} BTU/IN/SEC/ $^\circ R$	$.51 \times 10^{-4}$	7.8

The mean values of the properties are evaluated at a material temperature of 4062 $^\circ R$, the minimum fueled matrix temperature at station 36 where the maximum thermal stress occurs. The properties uncertainty analysis was done by computer code NORFO2.

(C-RD) 3) A comparison of the electrical corrosion test and the R-1 reactor indicates that the heating rate profiles are dissimilar. Figure 3-15 shows the comparison between the heating rates for two typical electrical corrosion tests and the heating rate for the R-1 ten hour reactor. Figure 3-16 shows the same comparison except that heating rate is plotted against element surface temperature. Figure 3-17 compares the channel wall temperature between the electrical test and the reactor test. The electrical test channel wall temperatures are significantly different than the reactor channel wall temperatures except at the mid-ban region. Figure 3-18 shows Q''' versus surface temperature for AXM fuel for a reactor and for a LASL Q''' test at reactor flow. The

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Figure A-1.3-15 Comparison of Heat Generation in Fuel Elements

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Figure A-1.3-16 Volumetric Heating Comparison Reactor and Electrical Test
Heating Rate Versus Temperature - 102-Mil Diameter Bores)

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Figure A-1.3-17 Calculated Channel Wall Temperatures

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calculated maximum axial stresses are shown at selected points along each curve. The basic dissimilarity between the LASL Q''' test and the reactor is further amplified by the different stress values for approximately the same Q''' in the two tests.

(C-RD) Axial Stress - The calculated maximum AXM axial stresses are plotted on Figures 3-19 through the 3-22. Figure 3-19 shows a calculated axial stress for a 2.3σ interelement temperature difference of 81.4°R at station 52 for a 19 hole reference element. Also shown on Figure 3-19 are the assumed bundling pressure distribution and the assumed interelement temperature difference distribution. The calculation of axial stress due to interelement temperature difference is performed as discussed in Section A-1.3.1.1. In this calculation the coefficient of friction was assumed as .2, the maximum bundling pressures were assumed to be 20 psi and 30 psi, and the maximum interelement temperature difference was assumed to be 81.4°R .

(U) Figure 3-20 shows the calculated maximum axial stresses due to heat generation and coating-matrix interaction for a 19 hole reference AXM fuel element with the maximum coating profiles and coating deposition temperatures as described on Figure 3-10. The calculations were performed with the finite element code FEATS using the heat generation rates and thermal boundary conditions of Figure 3-11. Figure 3-20 is a plot of the results of computer listing FDL-22.

(C-RD) Figure 3-21 shows the maximum combined axial stress in an AXM 19 hole element. In this figure, the axial stress due to heat generation and coating-matrix interaction (Figure 3-20) is statistically combined with the axial stress due to interelement temperature differences (Figure 3-19) and with a variational axial stress of 63% of the axial heat generation stress of 2042 psi at station 36 where the stress limit (80% of the 2.3σ minimum ultimate tensile strength) is approximately 1950 psi. The AXM 37 hole element (geometry as per Drawing 947C816) was analyzed using the finite element computer code FEATS. The results (computer output FDL-24) are plotted on Figure 3-22. The results indicate that the heat generation axial stress for a Q''' max of $66.4 \text{ BTU/in}^3 \text{ sec.}$ for the 37 hole element is less than the heat generation axial stress for the 19 hole element. This

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Figure A-1.3-18 AXM-630 Fuel Element R-1 10 Hour Reactor 1.1 PF
Maximum Axial Tensile Stress Shown

Figure A-1.3-19 Axial Stress AXM-630 for a 2.30' Interelement ΔT of 81.4°R at Station 52 with Bundling Pressures of 20 psi and 30 psi

Figure A-1.3-20 R-1 10 Hour Fuel AXM-630 19 Hole Model Axial Matrix Stress and Coating Stress for Maximum Coating Thickness Profile $Q'''_{\text{max}} = 66.4 \text{ BTU/IN}^3$

Figure A-1.3-21 Maximum Combined Axial Stress for AXM-630 Fuel 19 Hole Model $Q''' = 66.4 \text{ BTU/IN}^3 \text{ SEC}$

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improvement is due to the decreased pitch (.1309 versus .1740) and increased void ratio. However, the axial stress due to heat generation and coating-matrix interaction is approximately the same for both the AXM 37 hole and the AXM 19 hole element.

(C-RD) 30 V/O Composite Fuel - The transverse stress for the 30 V/O UC-ZrC-C fuel elements has been calculated and is small compared to the axial stress. The effect of axial friction and the high axial modulus of elasticity as compared to the transverse modulus of elasticity (see Figure 3-6) will cause axial stresses that will be more critical than the transverse stresses. The calculated axial stress for a 30 V/O UC-ZrC-C fuel element that is caused by a 2.3 σ interelement temperature difference of 81.4 $^{\circ}$ R is plotted on Figure 3-23. The frictional coefficient is assumed to be .2. The bundling pressure distribution and the temperature difference distribution are assumed to be as shown on Figure 3-19. The calculated axial stress for heat generation and coating-matrix interaction is shown on Figure 3-24. The comparison between the 80% of the 2.3 σ minimum ultimate tensile stress for 30 V/O UC-ZrC-C and the maximum combined axial stress is shown on Figure 3-25. The maximum calculated axial stress includes a 2.3 σ variation in stress of 43% caused by uncertainties in material properties and an axial stress caused by a 2.3 σ interelement temperature difference of 81.4 $^{\circ}$ R. The maximum combined axial stress for 30 V/O UC-ZrC-C is 4272 psi at station 36. The limit on tensile stress at station 36 is 4350 psi. The calculated maximum axial thermal stress for a castellated or Zig-Zag 30 V/O UC-ZrC element is shown plotted on Figure 3-26. A comparison between Figure 3-24 and Figure 3-26 shows that there is no apparent difference in thermal stress between a castellated or Zig-Zag element and a 19 hole element. The 30 V/O UC-ZrC-C 19 hole element was modeled for a finite element computer code FEATS. The output (computer listing FDL-25) is the basis for the plots on Figure 3-24. The composite Zig-Zag element was also modeled for FEATS. The output (computer listing FDL-27) is the basis for the plots on Figure 3-26.

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Figure A-1.3-22 AXM-630 37 Hole Model Matrix₃ and Coating Axial Stress, Maximum Coating Profile, $Q'''_{max} = 66.5$ BTU/in. sec.

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Figure A-1.3-23 Calculated Axial Stress 30 V/O UC-ZrC-C for a 2.3 σ Interelement ΔT of 81.4 $^{\circ}$ R at Station 52 with Bundling Pressure of 30 psi

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Figure A-1.3-24 30 V/O Composite Fuel (19 Hole Hex Model) Axial Matrix Stress and Coating Stress for Maximum Coating Thickness Profile

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Figure A-1.3-25 30 V/O Composite Fuel Axial Stress (19 Hole Element) Includes 2.3 Variation of Stress = 43% due to Variations in Material Properties; and 2.3 Interelement ΔT of 81.4°R with Bundling Pressure of 30 psi

Figure A-1.3-26 Composite (30 V/O) Fuel (Zig Zag Model) Axial Matrix Stress and Coating Stress for Maximum Coating Thickness Profile

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Carbide Single Hole Elements (C-RD)

(C-RD) Heat Generation Stresses - A preliminary study of the heat generation stresses in single hole fuel elements has been completed. Two basic shapes, tubes and hexes, were considered over a wide range of sizes (bore and outer diameters) and void fractions. The element material was UC-ZrC. Estimated material properties were used.

(C-RD) In this preliminary study, the outer surfaces were assumed to be adiabatic, and the inner surfaces were assumed to be at 4000°R , the approximate wall temperature level at station 38. This station was selected as the location of minimum structural margin using the curves shown on Figure 3-27. It can be seen that q''' , the volumetric heat generation rate, peaks near the mid-station. If the material properties were independent of temperature, this would be the station of maximum stress. However, the calculated stresses are proportional to $E\alpha Q''' D^2/k$ which peaks at station 19. Since structural margin also depends on strength, the calculated stress curve should be divided by the ultimate tensile strength σ_{ult} . This relationship peaks at station 38, which is the location of the maximum calculated stress to strength ratio or minimum structural margin. The shape of the curves in Figure 3-27 are not expected to change significantly with geometry changes. The temperature levels, conductivities, and heat generation rates were computed using R. Holman's data. At station 38, for a void fraction of 38%, the heat generation rate used was 52 BTU/sec./in.³. All heat generation rates were adjusted for void fraction changes using this data.

(C-RD) The maximum stress in either the tube or hex shaped elements can be expressed as:

$$\sigma_{\max} = C \frac{E\alpha Q''' (OD)^2}{16K (1-\nu)}$$

where

- $\sigma_{\max}$ = Maximum calculated thermal stress (psi)
- C = Shape factor (dimensionless)
- Q''' = Volumetric heat generation rate (BTU/sec./in.³)
- E = Modulus of elasticity (psi)
- ν = Poisson's ratio (dimensionless)
- α = Thermal expansion coefficient (1/°R)

- k = Thermal conductivity (BTU/sec./in./°R)
- OD = Outer diameter (tubes) or across flats dimension (hexes) (inches)

(C-RD) In this expression, the shape factor C is a function of the ratio of ID/OD, where ID is the bore diameter of the single hole element. Figure 3-28 shows a plot of shape factor C versus ID/OD. The tube relationship was prepared using the closed form solution available in the literature:

$$C = .5(ID/OD)^2 - 1.5 + 2 \ln (OD/ID)/1-(ID/OD)^2.$$

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The hex shape factor relationship was developed numerically using the FEATS finite element code. Sixteen cases were modeled and run for ID/OD ratios ranging from 0.364 to 0.982. Note that for the tubes, the stress and, therefore, the shape factor go to zero as the wall thickness goes to zero (ID/OD = 1). Also note that the shape factors for the hexes do not go to zero as the minimum web thickness goes to zero since there is material remaining at the corners of the hexes. The maximum stresses for any size tube or hex can be computed using the equation given earlier in this section and adjusting the heat generation rate for the void fraction. The material properties (E , ν , α , k) were evaluated at 4000°R.

(C-RD) The terms in the maximum stress equation are not deterministic, that is, the calculated stress must be considered statistically. It was shown by a statistical analysis, that the standard deviation of the calculated stress is about 17.5% of the mean or nominal stress. In order to account for this uncertainty in the calculations, 2.3 standard deviations were added to the calculated nominal stresses. These "maximum" stresses are plotted typically in Figures 3-29 and 3-30 for tubes and hexes, respectively, against outer diameter (tubes) and across flats dimension (hexes) with the bore diameter as the field parameter. Also shown in Figures 3-28 and 3-29 is the assumed allowable stress line evaluated at the 4000°R wall temperature. The allowable stresses were assumed to be:

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Figure A-1.3-27 Structural Parameters versus Core Station

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Figure A-1.3-28 Size Factor "C" versus Wall or Minimum Web Thickness

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Figure A-1.3-29 Maximum Heat Generation Stresses Single Hole Carbide Fuel Tubes

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Figure A-1.3-30 Maximum Heat Generation Stresses Single Hole Carbide Fuel

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Figure A-1.3-31 Single Hole Element Sizes with Acceptable Structural Margin

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Figure A-1.3-32 GLFI Fuel Axial Matrix Stress and Coating Stress for Maximum Coating Thickness Profile

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Figure A-1.3-33 GLFI Fuel Calculated Axial Stress Includes 2.3 σ Variation of Stress = 63% from Variation in Material Properties (Includes effect of 2.3 σ ΔT Between Elements of 81.4 $^{\circ}\text{R}$)

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$$\begin{aligned}\sigma_{\text{allowable}} &= 0.8 (\text{mean strength} - 2.3 \text{ standard deviation}) \\ &= 0.8 (11000 - 2.3 \times 1100) = 6700 \text{ psi}\end{aligned}$$

Note that the standard deviation was assumed to be 10% of the mean strength.

(C-RD) The intersection of the allowable stress line with the calculated stress curve represents the maximum element size (outer diameter or across flats dimension) for the given bore size in which the calculated 2.3 sigma maximum stress just equals the assumed allowable stress. These loci are plotted in Figure 3-31. Any element sizes which lie below these curves have acceptable structural margins. As the curves approach zero wall (or web) thickness, the hex curve diverges from the tube curve. This behavior results from the fact that the hexes can be stressed when the minimum web becomes zero. The tube curve approaches the zero wall thickness line asymptotically. It can be seen that tubes are slightly better than hexes in that thicker walls are permitted.

(C-RD) Interelement Temperature Difference Stresses - The single hole carbide elements will be subjected to interelement temperature differences. Stresses will be developed in the elements if sufficient bundling pressure and surface friction are available. Since these parameters are not well defined at this time, it is appropriate to look at the interelement temperature difference assuming complete lockup. The actual stress levels developed are slip limited, assuming a friction coefficient of 1.0 (which is probably

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Figure A-1.3-34 37 Hole Fuel Element Maximum Combined Axial Matrix Stress R-1
 $Q''' = 66.4 \text{ BTU/IN}^3\text{-SEC}$

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reasonably for carbide elements), a uniform bundling pressure of 30 psi, and an element length of 5.2 inches, the limiting slip stresses are:

<u>Across Flats Size (Inches)</u>	<u>Bore Diameter (Inches)</u>	<u>Maximum Slip Stress (psi) Frict = 1.0, 30 psi bundling, length=5.2 inches</u>
.170	.120	3300
.110	.080	5500

These slip stresses are directly proportional to the element lengths, friction coefficient, and bundling stresses. It is recommended that the single hole carbide elements be lightly bundled and as short as possible to reduce this stress mechanism.

(C-RD) GLFI Fuel Elements - Intermediate Expansion - The calculated axial stresses for a standard 19 hole GLFI fuel element with .102 inch reamed diameter bores and .753 inches across flats are plotted on Figures 3-32 and 3-33. The assumed element geometry is per drawing 947C812. The assumed bore coating profile and deposition temperatures are per Figure 3-10. The GLFI fuel element was analyzed using the finite element computer program FEATS. The results (computer output FDL-21) are plotted on Figure 3-32. Figure 3-33 compares the maximum combined axial stress for the GLFI fuel element with an assumed strength criteria. The strength criteria is assumed to be 80% of the 2.3 σ minimum ultimate tensile strength. The maximum combined axial stress includes the thermal stress due to heat generation and coating-matrix interaction, the thermal stress due to a 2.3 σ interelement temperature difference of 81.4°R (from Figure 3-19), and the 2.3 σ thermal stress due to the variation in material properties, geometry and nuclear heating (estimated to be 63% of the heat generation stress). Coating-matrix interaction is assumed to begin at the deposition temperature of ZrC. It is assumed that the NbC relaxes sufficiently because of the time duration at the ZrC deposition temperature that the NbC and the matrix are at a zero stress state at the ZrC deposition temperature. The maximum combined axial stress in the GLFI element is 2543 psi at station 36. This stress exceeds the assumed strength criteria of 2200 psi at station 36.

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(C-RD) Thirty-Seven (37) Hole Hybrid Fuel Element - Drawing 947C816 - The thirty-seven (37) hole hexagonal hybrid fuel element has 37 holes of .084 reamed diameter and is .799 inches across flats. The element is composed of AXM high expansion fueled graphite from stations 0 to 35 and 30 V/O UC-ZrC from station 35 to 52. The fuel element was modeled for the finite element computer program FEATS. The model consisted of a 1/6 symmetry section of an equilateral triangle with a 60° sector of an .084 hole at each apex of the triangle. The length of the side of the triangle was .1309 inches (the hole pitch for the 37 hole element). Coating thickness and deposition temperatures for the AXM fuel is assumed to be as described on Figure 3-10. Coating thickness and deposition temperatures for the 30 V/O composite fuel is assumed to be as described on Figure 3-10 except that the coating is entirely ZrC. The results of the analysis are contained in computer output FDL-24. The maximum combined axial matrix stress is plotted on Figure 3-34. This stress peaks at station 35 for both the AXM and the composite fuel. The assumed strength criteria for both AXM and 30 V/O composite is 80% of the 2.3σ minimum ultimate tensile strength. The maximum combined axial stress consists of the thermal stress due to heat generation and coating-matrix interaction, and the 2.3σ stresses due to material properties variation, nuclear heating variations and an interelement temperature difference of 81.4°R. At station 35 the maximum combined axial stress in the AXM is 1777 psi and the AXM assumed strength criterion is 1940 psi. At station 35, the maximum combined axial stress in the 30 V/O composite is 2788 psi and the strength criterion is 4350 psi.

(C-RD) 30 V/O UC-ZrC Fuel Element - 62 Inch Long Core - Drawing 947C814 - The long core composite fuel element is a standard 19 hole fuel element with .108 reamed diameter bores. The station where the maximum combined axial stress occurs was estimated to be station 41. The composite long element was modeled for the finite element computer code FEATS, using a one-twelfth symmetry section fuel element model. The results of the analysis are contained in the computer output FDL-28. The coating was assumed to be HED ZrC to the maximum thickness and deposition temperature as shown on Figure 3-10.

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The temperatures and thicknesses on Figure 3-10 are for a 52 inch long element. The temperature and thicknesses were normalized to a 62 inch long element. The maximum combined matrix axial stress is plotted on Figure 3-35. The assumed strength criterion for the composite is 80% of the 2.3 σ minimum ultimate tensile strength. This assumed strength criterion is also plotted on Figure 3-35. The maximum combined matrix axial stress is 3860 psi at station 41. The strength criterion is 4530 psi at station 41.

(C-RD) AXM Fuel-Large Core 40.6 Inches Diameter by 76 Inches Long - Drawing 947C815 - The large core AXM fuel element is a standard 19 hole element with .102 inch reamed diameter bores and .753 inches across flats. The station where the maximum combined axial stress occurs was estimated to be station 61. The AXM element was modeled for the finite element computer code FEATS, using a one-twelfth symmetry section fuel element model. The results of the analysis are contained in the computer output FDL-29. Coating thickness and deposition temperatures are assumed to be as shown on Figure 3-10 for the maximum coating thickness profile. The temperatures and coating thicknesses on Figure 3-10 are for a 52 inch long fuel element. The temperatures and thicknesses were normalized for an element 76 inches long. The heat generation stress, the heat generation stress with coating-matrix interaction, the maximum combined axial stress and the assumed strength criteria are plotted on Figure 3-36. The maximum combined axial stress is composed of the heat generation stress with coating-matrix interaction, the 2.3 σ stress due to variations in material properties, nuclear heating, and geometry, and the 2.3 σ stress due to an interelement temperature difference of 81.4 $^{\circ}$ R. The maximum combined axial stress peaks at 1954 psi at station 61. The assumed strength criteria is 80% of the 2.3 σ minimum ultimate tensile strength and is 1970 psi at station 61.

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A-1.3.2 Fuel Support (U)

A-1.3.2.1 Central Element (U)

(U) (a) The transverse stresses in the central element are mainly caused by coating-matrix interaction and by the transverse temperature distribution. The coating-matrix interaction stresses are determined using the expression for coating-matrix interaction for fuel elements from Section 3.1.1. The transverse thermal stress resulting from transverse temperature distribution is

$$\sigma_{th} = C_1 E_t \alpha_t (T_o - T_i) \quad \text{where}$$

E_t = transverse modulus of elasticity
 α_t = transverse coefficient of thermal expansion
 T_o = temperature of outer surface of central element
 T_i = temperature of inner surface of central element
 C_1 = shape factor

The temperature difference of $T_o - T_i$ is maximum near the end of a startup ramp. The shape factor varies with the geometry and the physical size of the central element.

(U) The axial stress in the central would be caused by the same mechanisms that cause the transverse stress. In addition, however, a temperature difference between the central element and the fuel element exists that causes additional axial stress in the central. During startup the central element is cooler than the fuel element. Friction between the central element and the fuel element causes axial tensile stress in the central element.

(U) (b) Central elements are not specifically traded in this study because the severity of their structural problems are substantially less than those of fuel problems.

A-1.3.2.2 Stem Assembly (U)

(U) The influence of variations in core size, cluster size, core pressure drop, and fuel element material have been investigated with respect to the support stem stress.

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The maximum stem stress occurs in the axial direction and is considerably more than the transverse (hoop) stress. The allowable axial stress in the support stem, based on yielding, depends on the primary loads. Changes in core width and length will also affect the secondary and bending loads, but to a much lesser degree than the primary loads.

(C-RD) The core pressure drop represents approximately 88% of the primary load with the closure cap pressure drop and core weight representing the remaining 10 and 2%, respectively. The factors mainly influencing the support stem stress are then the core area, the cluster size (which determines the number of support stems) and the core pressure drop. Figure 3-37 shows the core pressure drop as a function of support stem area for two core sizes and three total number of stems. The curves are based on a support stem primary stress of 41,300 psi as in the reference design. Two element to stem ratios, 3 to 1 and 6 to 1, based on the 19 hole and 37 hole elements have been used to determine the total number of support stems. The reference design, corresponding to a core pressure drop of 170 psi, $N = 349$, and a stem cross sectional area of 0.0142 in.^2 , is indicated on the figure.

(C-RD) Stem areas larger than the reference design will be required for core pressure drops greater than 120 psi and a 6 to 1 element to stem ratio. For a required stem area of 3 times the reference area, based on the same inside diameter, the stem thickness must be increased by 28%.

A-1.3.2.3 Hot End Hardware (U)

(C-RD) The parametric effects of fuel element geometric variations and central to outer element ratio variations upon the central element tip creep deformation and the cluster hot end support (CHES) hardware transient thermal stresses are presented in this section. It is assumed that temperatures in the region of the CHES and the CHES geometry is unaltered by the above variations. With these assumptions the parameters that perturb creep and transient thermal stresses are variations in core pressure drop, cluster weight and cluster cross sectional area (area upon which the core pressure drop acts) due to element/cluster geometric changes.

(a) Central Element Tip Creep (U)

(C-RD) Figures 3-38 and 3-39 illustrate the effect of fuel element/cluster geometric variations upon creep of the central element tip. These curves indicate the amount of creep deformation that will occur in ten hours of operation for the given geometry, temperature, and core pressure drop. With a nominal core pressure drop of 170 psi, caution must be exercised in selecting the geometry of composite tips with the six element per central arrangement shown on Figure 3-38. Each figure shows that three elements per central results in significantly reduced creep deformation for a given core pressure drop when compared to six elements per central. A comparison of tip materials shows that AXM creeps less than the composite.

(b) Transient Thermal Stresses (U)

(U) The cluster hot end support hardware considered for parametric stress analysis were the composite standoff with composite retainer (940J128) and the composite standoff with tungsten retainer (982D130). The maximum transient stresses were evaluated using combined axial, friction and thermal gradient loads.

(U) Figures 3-40 and 3-41 show isostresses and the location of maximum and minimum combined stresses in the composite standoff with composite retainer for nominal element/cluster geometry. Figures 3-42 and 3-43 are similar figures for the composite with the tungsten retainer/shield.

(C-RD) Figures 3-44 and 3-45 illustrate the effect of fuel element/cluster geometric variations upon the maximum transient stresses. For a given element/cluster geometric variation, the central element tip load is determined from the sum of core pressure drop at the transient time acting on the cluster area, cluster weight and spring load ($A \Delta P + W + S$). Once the tip load is known the maximum transient stresses can be read directly from these figures along with allowable strengths for different heating ramp rates. The maximum tensile stress for the all composite design shown on Figure 3-41 decreases with element/cluster geometric variations that cause an increased tip load because the mechanical and frictional contributions to the total stress are compressive and tend to cancel the high

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transient, tensile thermal stresses.

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Figure A-1.3-35 30 V/O UC-ZrC-C Fuel Element - 62 Inch Long Core
Maximum Combined Axial Stresses

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Figure A-1.3-36. AXM Fuel - Large Core - 40.6" Diameter x 76" Long - Maximum Axial Matrix Stress

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Figure A-1.3-37 Core Pressure Drop versus Support Stem Cross Sectional Area

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Figure A-1.3-38 65 W/O NbC-C Composite Central Element Tip - Maximum Allowable Core Pressure Drop for the Given Geometry and Tip Creep Deformation for 10 Hours of Operation at a 4250°R Chamber Temp.

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Figure A-1.3-40 All Composite Transient (200°R/sec.) Maximum Isostress (940J128)

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Figure A-1.3-41 All Composite Transient (200°R/sec.) Minimum Isostresses (940J128)

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TIME = 42 SEC.

Figure A-1.3-42 Radial Isostresses in Composite Protection Cup for 200°R/sec. Transient (982D137)

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TIME = 42 SEC.

Figure A-1.3-43 Axial Isostresses in Composite Protection Cup for
200° R/sec. Transient (982D130)

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Figure A-1.3-44 All Composite Components - Maximum Tensile and Compressive Stress versus Heating Ramp Rate and Mechanical Loading (940J128)

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Figure A-1.3-45 Composite Cup Maximum Tensile and Compressive Stresses versus Heating Ramp Rate and Mechanical Loading (982D130)

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A-1.4 THERMAL AND CORROSION CAPABILITY

(U) In this section, the core pressure drop, fuel channel wall temperatures, relative corrosion due to hot end diffusion and relative reactivity loss rates for the fuel element are given as parametric functions of the fuel element hole pitch and reamed diameter. Relative effects of changes in core size are also given in parametric form. Detailed technical information supporting the parametric analysis is given as described below.

(U) Section A-1.4.1 presents the thermal and fluid flow environment of the reference R-1 fuel element and describes the methods and assumptions used. The development of temperature uncertainties due to tolerances and inaccuracies in power prediction is also given. The best estimate of maximum temperature limits for the various matrix and coating materials are discussed using information available at the time of writing. A comparison of thermal conductivity data for fueled graphite and composite material is presented using preliminary data available at time of writing.

(U) The effect on core pressure drop and fuel temperatures of changing the fuel geometry and/or coating thickness is given in Section A-1.4.2. The relationship between hydraulic effective impedance diameter, reamed channel diameter and coating thickness is discussed. The effect of an axially varying fuel void fraction or thicker coatings at the aft end are discussed in relation to core performance.

(U) The parametric curves for core pressure drop and fuel channel wall temperatures as functions of fueled diameter and core length are given in Section A-1.4.3.

(U) Section A-1.4.4 contains parametric relationships between hot end diffusion corrosion and changes in fuel element geometry and/or core size.

(U) The effect of changes in core diameter and length on support stem flow control is given in Section A-1.4.5.

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A-1.4.1 (CRD) Analytical Background (U)

(CRD) A-1.4.1.1 Reference Fuel Element Design (U)

(U) This section gives detailed thermal and hydraulic data for the reference R-1 fuel element as calculated by the analysis methods listed in the references.

(U) Figure A-1.4.1-1 shows the normalized radial power distribution for one traverse from core centerline to periphery. The power is relatively uniform because of the multiregion fuel loading scheme. Variations across a single element of up to 30% are present at the periphery because of the neutron reflector.

(U) The axial power distribution shown in Figure A-1.4.1-2 follows the chopped cosine shape typical of cylindrical nuclear reactors without end reflectors. For a radial maximum to average power factor of 1.1 (from Figure A-1.4.1-1) the peak volumetric heating rate is 66.4 BTU/sec - inch³ at station 24 inches.

Figure A-1.4.1-3 shows the axial temperature distribution in a fuel element with the volumetric heating rate given by Figure A-1.4.1-2. The inlet and exit gas temperatures and fuel element flow rate were specified according to the reactor state point diagram to achieve 4250°R chamber temperature and 450 psia chamber pressure. The calculations were carried out by the MCAP computer program discussed in Reference A-1.5. The fuel hot spot temperature at the center of the uniform triangular array of coolant holes is calculated by assuming that there is no radial conduction.

(CRD) The MCAP model assumes the channel coating generates heat at the same rate as the fueled material. This causes negligible temperature error for thin coating (i.e., ~ 2-3 mils). The volumetric heating rates quoted in this report have been adjusted to the reamed diameter and are correct. Likewise, channel wall temperatures which are used for corrosion calculations are correct. The temperature drop at station 46 inches across a 6 mil coating is estimated to be 22°R, which does affect the maximum coating temperature at the coating matrix interface, based on a correct treatment of the problem.

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(U) The maximum fuel hot spot temperature is 4463°R at station 46 inches. Since the outer boundary of the fuel element is not an equilateral triangular array it is of interest to compare the adiabatic hot spot temperature with a more detailed calculation. Figure A-1.4.1-4 shows a nodal model for one half of a fuel element. The TRACK program was used to calculate the temperature distribution shown below the model. The edges of the model are hotter than near the center where the equilateral triangular array of coolant holes occur. The relative temperature distribution shown in this figure does not appear symmetric because the nodal pattern is not symmetric.

(U) Figure A-1.4.1-5 shows the axial distribution of fuel channel static pressure for power factors of 1.1 and 0.6 for the reference fuel element and core size. The friction and momentum losses are significantly higher for the 1.1 power factor channel, accounting for about 97% of the total pressure drop. Because of the large change axially in gas temperature and viscosity, the Reynolds number changes by nearly a factor of ten from inlet to exit. This is shown in Figure A-1.4.1-6. The pressure drop calculations are, therefore, best carried out by summing the effects over a series of finite axial increments as is done by MCAP. However, since the parametric study is being carried out at fixed gas inlet and exit temperature and fixed chamber pressure, the use of simplified geometric scaling laws to estimate the core pressure drop relative to a reference case is valid.

(CRD) A-1.4.1.2 Temperature Uncertainties (U)

(U) The nominal fuel element temperatures given in Section A-1.4.1.1 represent ideal conditions when all variables affecting the individual fuel element's power generation and flow rate are known exactly. The inaccuracies in the predicted heat generation and flow for each channel cause a channel-to-channel variation from the desired average temperature and a statistical "hot channel" temperature. This section

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discusses the effect of these hot channel factors on the maximum temperatures experienced by the reference element. In general, the same temperature uncertainties are expected to apply to any of the fuel element and core configurations and matrix materials. However, because of the "learning curve" phenomenon, any change from the reference design could be expected to cause at least a temporary increase in the size of all uncertainties. It is believed that in the long run, the overall "hot channel" temperature increase above nominal should be very close to the reference values for any alternate.

(U) Table A-1.4.1-1, Item 1.0, documents the individual hot channel uncertainties which affect an individual element's exit gas* temperature with respect to the average gas temperature of the surrounding elements. The element-to-element temperature uncertainty (3σ) is approximately $\pm 100^\circ\text{R}$ in both core center and periphery (Item 2.0). For comparison, measured temperature data from previous reactor tests indicate somewhat higher temperature variations, element-to-element, based on least-squares curve fitting, after subtracting the instrument accuracy.

(CRD) Uncertainty in the radial power distribution is the largest component in the hot channel analysis and has been estimated to be a "regional" effect, i.e., affects all elements in a local radial region equally. Other regional effects, tabulated in Item 4.0 of the second page of Table A-1.4.1-1 are relatively insignificant. Two estimates for the accuracy of radial power prediction have been given, one value after incorporating the results of a single reactor power test into the nuclear prediction and the second after incorporating a series of power tests on identical reactors into the prediction. The radial region with the highest uncertainty is the core periphery, resulting in an overall

* Any uncertainties affecting the radial transfer of heat within the element which would apply to the temperature differences between gas and the matrix are negligible by comparison and are not included.

combined regional uncertainty in exit gas temperature of 325°R after one power test, Item 5.0. The statistical combination of regional and element effects results in a 3σ uncertainty in exit gas temperature of 342°R at the core periphery. These uncertainties do not include the effects of control drum rotation or accuracy in controlling the reactor mixed-mean (or chamber) temperature. The former is neglected because approximately \$3.00 reactivity compensation by support stem flow control has been assumed for the trade study. The uncertainty in core mixed mean exit gas temperature was estimated* to be $\pm 63^{\circ}\text{R}$ (3σ) using a radial distribution of thirty-six exit gas thermocouples in the control loop. Combining this with the element and regional effects results in a statistical temperature increase of 348°R (3σ) above nominal for the core periphery after one power test.

(CRD) The nuclear power uncertainties which go to make up the radial power prediction (Item 4.1) are tabulated separately on Table A-1.4.1-2. Incorporation of measured power distributions from reactor tests into later predictions significantly reduces the uncertainties. This is demonstrated by Figure A-1.4.1-7 where the error in radial power distribution for NRX-A6 varies from +3% at the center to about -1% to -3% at radius 30 centimeters. The azimuthal variation contributes to the measured element-to-element temperature variation given by Table A-1.4.1-1, Item 3.0, and accounts partially for the difference between calculated element-to-element effects and measured data. If a second NRX-A6 reactor were to be built tomorrow, many of the factors which caused the discrepancies shown in Figure A-1.4.1-7 would be accounted for in predictions and the uncertainties would gradually be reduced for the succeeding reactors. Naturally, a point of diminishing returns is reached where measurement inaccuracies and corrections for fuel element degradation, fission product diffusion, etc., become controlling.

* Reference DRM 51422

A conservative estimate for the 3σ temperature increase for this trade study is taken to be 348°R which represents the core periphery, after one power test, neglecting control drum effects, but including a $\pm 63^{\circ}\text{R}$ uncertainty in mixed mean temperatures. The same uncertainty is assumed to apply to the exit gas, channel wall, tip braze joint, and adiabatic hot spot temperature in the fuel. The element-to-element uncertainty is 106°R , which omits regional effects.

(CRD) In summary, the nominal and 3σ temperatures for the reference R-1 fuel element are given below. (See Figure A-1.4.1-3 for additional details.)

<u>Location</u>	<u>Isolated Channel - MCAP- Calculation for R-1 Fuel Temperatures - $^{\circ}\text{R}$</u>	
	<u>Nominal</u>	<u>$3\sigma^*$</u>
Exit Gas	4284	4632
Tip Braze Joint	4361	4709
Maximum T_{wall} (Station 48)	4389	4737
Maximum T_{fuel} (Station 46)	4463	4811

* At core periphery, after one power test, excluding control drum effects.

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(U) REFERENCES (U)

- A-1.1 M. F. Taylor, "A Method of Correlating Local and Average Friction Coefficients for Both Laminar and Turbulent Flow of Gases through a Smooth Tube with Surface to Fluid Bulk Temperature Ratios from 0.35 to 7.35", Int. J. Heat Mass Transfer, Vol. 10, pp 1123-1128.
- A-1.2 WANL-TME-1642, "NRX-A6 Nuclear Subsystem Thermal and Nuclear Design Data Book".
- A-1.3 DRM 51479, "NBS Para-Hydrogen Property Values".
- A-1.4 DRM 51472, "Thermal Material Properties".
- A-1.5 WANL-TME-944 "A Multiple Channel Analysis Program (MCAP) for the Temperature Distribution in an Internally - Cooled Heat Generating Solid"

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FIGURE A-1.4.1-1 (U) RADIAL FISSION DISTRIBUTION FOR THE BERYLLIUM RING DESIGN AND HIGH CTE FUEL (U)

Figure A-1.4.1-2 (U) Nominal Volumetric Heating Rate in R-1 Fuel Element (U)

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Figure A-1. 4. 1-3. (CRD) Temperature Versus Axial Station for Graphite Elements (U)

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NODAL MODEL USED IN TRACK STUDY OF
FUEL ELEMENT

Figure A-1.4. 1-4. (CRD) Radial Temperature Plots Across a Single Fuel
Elements from Results of Track Study (U)

(Adiabatic External Boundary)

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Figure A-1.4.1-5 (CRD) Fuel Channel Static Pressure Versus Axial Station (U)

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Figure A-1.4.1-6 (CRD) Reynold's Number Based on Fluid Bulk Temperature Versus Axial Station (U)

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AFTER MULTIPLE PWR TESTS
 AFTER 1 PWR TEST
 CENTER
 3 σ
 3 σ
 EXPERIMENTAL POWER/PREDICTED POWER

ONE
 MULTIPLE PWR TESTS
 CENTER
 3 σ
 3 σ

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Figure A-1.4.1-7 (U) NRX-A6 Power Distribution Discrepancy for Each Element Location As A Function of Radial Position (U)

TABLE A-1.4.1-1

(CRD) SOURCE OF TEMPERATURE UNCERTAINTIES (3σ VALUES) (U)

	CENTRAL REGION (°R)	PERIPHERAL REGION (°R)
1.0 ELEMENT EFFECTS		
1.1 γ SCAN (U ASSAY) 1% CENTRAL 2% PERIPHERAL	52	88
1.2 AIR IMPEDANCE (NO X COAT) 0.4 MILS	54	31
1.3 ORIFICE STEP CHANGES ON ELEMENT BASIS 10°R	10	10
1.4 COMBINED CHANNEL EFFECTS	57.5	49.0
INDIVIDUAL CHANNEL EFFECTS LEADING TO 1.4		
1.4.1 FUEL HOMOGENITY	240	203
1.4.2 ORIFICE COEFFICIENT (ACCEPTANCE LIMIT 50°R)	50	50
1.4.3 HOT END SUPPORT 1% FLOW	52	44
2.0 COMBINED ELEMENT EFFECT	95	106
3.0 COMPARISON WITH NRX TEST DATA ⁽¹⁾ (3σ)		
3.1 NRX-A6 EXIT GAS T/C'S	+128°R (SOL)	+172°R (EOL)
3.2 NRX-A6 STATION 26 T/C'S	+156°R (SOL)	+179°R (EOL)
3.3 NRX-A6 THERMAL CAPSULES	+242°R (MELTED AT POWER PEAK SOL)	
3.4 NRX-A5 STATION 32 T/C'S	+156°R (SOL)	+172°R (EOL)
3.5 NRX-A5 STATION 26 T/C'S	+166°R (SOL)	+135°R (EOL)
3.6 NRX-A5 THERMAL CAPSULES	+141°R (MELTED AT DIFFERENT TIMES DURING RUN)	

(1) FROM LEAST-SQUARES CURVE FITTING, AFTER SUBTRACTING INSTRUMENT ACCURACY

TABLE A-1. 4. 1-1 (CONTINUED)

	<u>CENTRAL REGION ($^{\circ}\text{R}$)</u>	<u>PERIPHERAL REGION ($^{\circ}\text{R}$)</u>
4.0 REGIONAL EFFECTS		
4.1 RADIAL POWER PREDICTION - 4.2%(2.1%) CENTRAL 7.2%(3.6%) PERIPHERAL	225 (113) ⁽¹⁾	322 (161)
4.2 AVERAGE CORE TIN $\pm 10^{\circ}\text{R}$	5	15
4.3 LOCAL CORE TIN $\pm 10^{\circ}\text{R}$	27	40
4.4 AXIAL POWER DISTRIBUTION 3% AT STA. 2 INCHES	7	5
5.0 COMBINED REGIONAL EFFECTS	227 (116)	325 (167)
(2.0 COMBINED ELEMENT EFFECTS)	95	106
6.0 COMBINED ELEMENT AND REGIONAL EFFECTS	246 (150)	342 (198)
7.0 COMBINED ELEMENT AND REGIONAL EFFECTS WITH $\pm 63^{\circ}\text{R}$ (3σ) UNCERTAINTY IN GAS MEAN TEMPERATURE (REF. DRM 51422)	254 (164)	348 (208)

(1) AFTER ONE POWER TEST (AFTER MULTIPLE POWER TESTS)

TABLE A-1.4.1-2

(U) NUCLEAR POWER UNCERTAINTIES (3σ) FOR A CONSTANT DRUM ANGLE (%) (U)CONTINUING NCX+
MULTIPLE POWER TESTS
W/O DESIGN CHANGES
+ INPLACE SHIM

	PAX/R1 + NCX + INPLACE SHIM		PAX/R1 + NCX+ INPLACE SHIM + POWER TEST		CONTINUING NCX+ MULTIPLE POWER TESTS W/O DESIGN CHANGES + INPLACE SHIM	
	CORE CENTER	CORE EDGE	CORE CENTER	CORE EDGE	CORE CENTER	CORE EDGE
ERROR IN AMOUNT OF GAMMA AND NEUTRON HEATING DEPOSITED IN CORE	± 0.2	∓ 0.6	± 0.2	∓ 0.6	± 0.2	∓ 0.6
ERROR IN THE GAMMA AND NEUTRON HEATING DISTRIBUTION	± 3.3	∓ 2.1	± 3.3	∓ 2.1	± 1.5	∓ 1.0
ERROR IN EXPERIMENTAL FISSIONS/GRAM MEASUREMENTS	± 0.8	∓ 1.4	± 0.8	∓ 1.4	± 1.5	∓ 3.4
ERROR DUE TO EXPERIMENTAL MOCKUP	± 1.1	∓ 3.0	± 1.1	∓ 3.0		
ERROR IN COLD-TO-HOT SHIFT	± 3.0	∓ 9.0	± 2.3	∓ 6.0		
ERROR DUE TO SHIELD EFFECT	± 0.2	∓ 0.8	± 0.2	∓ 0.8		
STATISTICAL SUM	4.7	9.9	4.2	7.2	2.1	3.6

A-1.4.1.3 (CRD) Material Limits (U)

(CRD) This section discusses fuel element temperature limits for bead-loaded graphite and 30 v/o UC - ZrC - C composites.

(CRD) The melting temperature of pure NbC and ZrC is very high, as shown by the first set of temperatures on Table A-1.4.1-1. When mixed together and combined with small quantities of uranium, LASL has estimated a minimum melting temperature of 2800°C (5530°R). However, recent Y-12 data* on ZrC vapor deposited graphite elements indicate much lower temperatures. These data, given as Item 2.0 on Table A-1.4.1-1 indicate melting of the coating at approximately 2400°C , (4810°R). Based on the conservative maximum 3 σ channel wall temperature of 4737°R given in Section A-1.4.1.3.1 the use of ZrC coating on 630 mg/cc bead-loaded graphite elements appears feasible, especially if an NbC undercoating is used.

(CRD) Although bead migration is not considered a fuel operating temperature limitation, it is of interest because the eutectic formation with uranium affects the coating's melting point. For the standard bead heated in an oven for one hour, Item 3.0 shows 2% migrating through the shell at 2300°C (4630°R) and 100% migrating through the shell at 2460°C (4920°R). The 3 σ maximum fuel matrix temperature for isolated conditions is 4811°R , so that some localized bead migration in 10 hours is to be expected. Since the Y-12 tests on ZrC coated unfueled elements did not show melting until $2750\text{--}2900^{\circ}\text{C}$ ($5440\text{--}5710^{\circ}\text{R}$) was reached, it is to be expected that the lower loadings would show better temperature capability. If predominantly NbC coatings are used for the graphite elements, the coating melting temperature is much higher and based on previous experience, would not be limiting.

(CRD) Based on experience in previous reactors and developmental cyclic tests, the fueled graphite - composite tip joint appears capable of attaining 60 cycles, at the 3 σ maximum temperature of 4709°R .

* Reference Y - DA - 3194 "Melting Points of Coatings of NERVA Fuel Elements", December, 1969.

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(CRD) The maximum loading limit for bead loaded graphite fuel has not been determined, but based on the experience in developing the 630 mg/cc elements for PW-2 it appears that any substantial increases are very difficult to achieve without sacrificing fuel element permeability and strength. In a Fuel committee meeting on April 30, 1969, the representative from Y-12 presented data indicating that flexure strength remained constant between 300 and 650 mg/cc but compressive strength dropped by approximately 2000 psi for every 50 mg/cc increase in loading above 500 mg/cc.

(CRD) For the UC · ZrC - C composite fuel element, a recent LASL presentation gave the data shown on Figure A-1.4.1-8 for the temperature at which the liquid phase first appeared versus the uranium loading. A conservative maximum loading was chosen as 700 mg/cc which corresponds to $\sim 2600^{\circ}\text{C}$ (5170°R) for 30 v/o composite. An extreme limit, (which happens to correspond to the Y-12 estimate of the 2400°C melting point for ZrC on fueled graphite) was chosen as 900 mg/cc. Since the LASL experimental data is extrapolated beyond 500 mg/cc it is felt that the conservative 700 mg/cc limit is appropriate for this trade study.

(CRD) In summary, for bead loaded graphite elements, a maximum loading limit of 630 mg/cc has been established, and the desirability of coating with NbC or an NbC under-coating overlaid with ZrC has been indicated. For the 30 v/o composite, a preliminary maximum loading limit of 700 mg/cc was set but should be re-evaluated when more data for loadings above 500 mg/cc become available. None of the combinations of coatings and matrix appear to be temperature limited at the nominal conditions for the reference element. The lowest temperature at which a problem is indicated is based on Y-12's evaluation of ZrC coating on 630 mg/cc fueled graphite which melted at or below 2400°C (4810°R) in one hour.

Figure A-1.4.1-8 (CRD) Temperature at First Appearance of Liquid Phase Versus Uranium Loading (U)

TABLE A-1.4.1-1
(CRD) FUEL TEMPERATURE TECHNOLOGY (U)

MATERIALS

1.0 Melting Temperature:

NbC	3500°C (6800°R)
ZrC	3420°C (6650°R)
NbC - ZrC - U Minimum (LASL)	2800°C (5530°R)

2.0 Recent Y-12 Data* on Melting of Coated Specimens Heated for One Hour in Flowing Argon

ZrC on Unfueled Carbon Showed Melting At 2750 - 2900°C (5440 - 5710°R)
ZrC on 630 MG U/CC Showed Localized Melting of ZrC at 2400°C (4810°R)
and Complete Melting of ZrC at 2500°C (4990°R)
NbC-ZrC on 400 mg/cc graphite melted after heating to 2500°C (4990°R)

3.0 Bead Migration (Not Considered a Fuel Operating Temperature Limitation):

Std. Bead - Some beads migrate through shell (2%) in one hour	2300°C (4630°R)
All beads migrate through shell in one hour	2460°C (4920°R)

* Reference: Y-DA-3194 "Melting Points of Coatings of NERVA Fuel Elements," December 10, 1969

(CRD) A-1.4.1.4 Thermal Conductivity Data (U)

(CRD) The fuel element thermal conductivity affects only the ΔT between the fuel hot spot and channel wall. As shown by Figure A-1.4.1-3, this ΔT reaches a maximum of about 200°R at station 30 inches and drops to about 100°R at station 46 inches where peak fuel temperatures occur. The LASL data for ZrC-C composites recently presented at WANL is shown in Figure A-1.4.1-9. Above about 1500°C , it is higher than for ATJ graphite. In Figure A-1.4.1-10, this same data (which was tabulated in a LASL quarterly status report) is compared with WANL data for 630 mg/cc bead loaded graphite elements. Finally, on Figure A-1.4.1-11, the LASL data for 30 v/o ZrC-C composite is compared with the graphite fuel conductivity used for NRX-A6. LASL has stated that the conductivity for low uranium concentrations in $\text{ZrC} \cdot \text{UC} - \text{C}$ is comparable to that of $\text{ZrC} - \text{C}$. The conductivity depends very strongly on the percent of maximum theoretical density (or void fraction) within the sample. In comparing the LASL measurements, which were made by a flash diffusivity method, with analytical calculations for 100% dense material, a density effect was noted. This difference would be accounted for if the sample material were 70 - 74% of theoretical density. Because of the rather small effect of conductivity at the hot end of the fuel element (roughly 1°R per 1% change in conductivity at station 46) and the variations in conductivity caused by manufacturing for both composite and graphite, no adjustment in parametric curves has been made for the various materials in quoting maximum temperature levels (the AXM graphite data is used). Since the conductivities do differ more significantly at lower temperatures, however, the thermal stress calculations in Section 3.0 have incorporated the latest available data for each material.

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Figure A-1. 4. 1-9. (U) Thermal Conductivity Derived from Diffusivity Data for Extruded ZrC-C Composites (U)

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Reference: LA-4270-MS "Quarterly Status Report
of LASL Rover Program for Period
Ending August 31, 1969.

Curve fit equations to measured thermal conductivity
data by flash diffusivity method.

LASL DATA FOR ACROSS GRAIN:

- — ○ 20 V/o ZrC-C
- 30 V/o ZrC-C
- — □ 40 V/o ZrC-C
- x - - - x 100 V/o ZrC-C (Ref. LASL AN-519)

Note: Comparison of data for 30^V/o ZrC
with theoretical prediction indicates
sample Void fraction of 20 - 30^o/o.
100^o/o theoretical density would
cause ~60^o/o higher thermal
conductivity.

Figure A-1.4.1-10 (CRD) Thermal Conductivity Versus Temperature

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FIGURE A-1.4.1-11 COMPARISON OF 30^V/o COMPOSITE WITH NRX-A6 GRAPHITE THERMAL CONDUCTIVITY. (U)

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A-1.4.2 (CRD) EFFECT OF VARYING FUEL GEOMETRY AND CHANNEL COATING THICKNESS (U)

A-1.4.2.1 (CRD) Core Pressure Drop (U)

(CRD) If the fuel reamed diameter is reduced from the reference 0.102 inch dimension, or if the centerline spacing (pitch) between holes is increased, or if the channel coating thickness is increased, the cross sectional flow area for the core is reduced and consequently, the core pressure drop will increase. The reference size R-1 reactor, made up of 1878 fueled elements and 349 unfueled central elements, has a core pressure drop of 170 psi using the reference 19 hole fuel element with 0.102 reamed diameters and 2.18 mils average coating thickness. The pressure drop is set by the flow requirements of the individual flow channel with highest heating (assumed 1.1 radial power factor relative to average) and smallest impedance diameter (required minimum diameter for reference element is 0.0935 inches). Other channels with relatively lower heating and/or larger diameters have lower friction and momentum pressure losses. Proper flow balancing to achieve uniform gas exit temperatures is achieved by metering with orifices installed at the inlet end of each channel.

(U) The following subsections discuss the variation in core pressure drop with changes in fuel element and core geometry. For this parameter study, the fuel matrix material has no effect on core pressure drop, therefore, the discussion in this section will be applicable to all matrix materials.

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A-1.4.2.1.1 (U) Core Pressure Drop Relationship (U)

Table A-1.4.2-1 presents an equation for the pressure drop showing the influence of the various parameters. Note that in the expression for the friction pressure loss, which represents 77% of the total for a 1.1 power factor channel, an axial summation over finite increments has been included. This has been done because of the large axial variation in Reynold's number shown by Figure A-1.4.1-6.

For the reference R-1, the relative magnitude of inlet, momentum, friction, and exit losses are tabulated for 1.1 and 0.6 power factor channels. Assuming that the variation in fluid properties is negligible* the pressure drop for 1.1 power factor channel is approximately proportional to:

$$\Delta P \propto \frac{W^2}{D^4} f \left(D^{-.75} W^{-.25} \right)$$

Since the flow per channel is inversely proportional to the number of flow channels, substitutions for W in terms of geometry (hole pitch, reactor fueled diameter, number of central elements, etc.) can readily be made. Figure A-1.4.2-1 shows core pressure drop versus fuel reamed diameter for constant coating thickness and hole pitch in the reference core. Figure A-1.4.2-2 is a cross-plot showing ΔP versus hole pitch for constant reamed diameter. The relationship between reamed diameter, coating thickness and hydraulic diameter is discussed in the next paragraphs.

* For constant inlet and outlet temperatures, constant chamber pressure and small variations in inlet pressure this is a reasonable assumption.

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TABLE A-1.4.2-1
CORE PRESSURE DROP

$$\Delta P_{\text{Total}} = \Delta P_{\text{Inlet}} + \Delta P_{\text{Friction}} + \Delta P_{\text{Momentum}} + \Delta P_{\text{Exit}}$$

$$\Delta P_I = CL_I \frac{G^2 v_i}{2g}$$

$$\Delta P_F = \frac{G^2 L}{g D} \sum_{i=1}^n f_i (v_i + v_{i+1})$$

$$\Delta P_M = \frac{G^2}{g}$$

$$\Delta P_E = CL_E \frac{G^2}{2g}$$

where:

CL = Loss coefficient, which includes both frictional and momentum losses - dimensionless

CL_I = At inlet, for sudden contraction

CL_E = At exit, for sudden expansion

G = Mass flow rate per unit area - lb/sec-in² = $\frac{W \pi}{4D^2}$

v = Specific Volume - in³/lb.

v_I = At inlet

v_E = At exit

v_i = At intermediate station i

g = Gravitational constant - 386.088 in/sec²

L = Total Length of channel - inch

D = Channel diameter, constant along L - inch

TABLE A-1.4.2-1 (Continued)

n	=	Number of equal axial increments into which L is divided for summation of $f_i (v_i + v_{i+1})$ terms
f_i	=	Fanning friction factor - dimensionless NOTE: $f_i = f(R_{ei})$ where R_{ei} = Average Reynolds number in the i increment. A commonly used friction factor relationship is Blasius's law.* $f_i = \frac{1}{4} 0.316 (R_{ei})^{-.25}$
Re_i	=	$\frac{G D}{\mu_i} = \frac{W}{\pi D \mu_i} \quad (\text{average over increment})$
μ_i	=	Dynamic viscosity - lb/sec-inch (average over increment)
W	=	Mass flow rate per channel - lb/sec.

After substituting for μ_i and Re_i in the expression for ΔP_F :

$$\Delta P_F = \frac{W^2 L}{D^5} \cdot \frac{16}{\pi^2} \cdot \frac{.316}{4g} \sum_{i=1}^n \left[\left(\frac{4}{\pi} \frac{D \mu_i}{W} \right)^{.25} (v_i + v_{i+1}) \right]$$

Substituting, and collecting terms in the expression for ΔP_{Total} :

$$\Delta P_{Total} = \frac{W^2}{D^4} \cdot \frac{16}{\pi^2 g} \left\{ \underbrace{v_i \frac{C_{LI}}{2}}_{\text{Inlet}} + \underbrace{(v_o - v_i)}_{\text{Momentum}} + \underbrace{v_o \frac{C_{LE}}{2}}_{\text{Exit}} \right. \\ \left. + \underbrace{\frac{0.316}{4} \cdot \frac{L}{D} \cdot \frac{\pi D}{4W} \cdot .25}_{\text{Friction}} \sum_{i=1}^n (v_i)^{.25} (v_i + v_{i+1}) \right\}$$

* NOTE: This less complicated equation has been used for illustration rather than the Taylor friction factor correlation which is actually used in NERVA design.

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TABLE A-1.4.2-1 (Continued)

RELATIVE MAGNITUDE OF PRESSURE LOSSES

	$P_F = 1.1$	$P_F = 0.6$
Inlet	- 6.5%	- 67.9%
Momentum	- 20.9%	- 6.1%
Friction	- 76.9%	- 27.3%
Exit*	+ 4.3%	+ 1.3%

* Exit loss includes pressure recovery in plenum.

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Reference Core Size - 1878 FE, 52 Inches Long
Constant Coating Thickness = 2.18 mils

Figure A-1.4.2-1 Core Pressure Drop Versus Fuel Reamed Diameter, (D) at
(CRD)
Constant Hole Pitch, (S) (U)

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Figure A-1.4.2-2 Core Pressure Drop versus Hole Pitch for Constant Reamed Diameter (U)
(CRD)

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A-1.4.2.1.2 (CRD) Effective Impedance Diameter (U)

(CRD) The core pressure drop is determined by the channel having the highest local power generation and the smallest hydraulic impedance diameter. For the small coolant holes with vapor-deposited coating used in NERVA, the hydraulic diameters are measured with an air-impedance device which compares the channel with an equivalent smooth metal tube. The resulting equivalent impedance diameter includes the effect of the coating thickness on the reamed diameter and the roughness of the coated bore. Because of the manufacturing tolerances and variability in roughness effects only the minimum effective diameter of 0.0935 inches is specified for the reference element using 0.102 reamed diameters and an average coating thickness of 2.18 mils. If the bores were perfectly smooth and no manufacturing tolerances were involved, the minimum hydraulic diameter would be .0976 inches. The difference of 4.1 mils between this and the specified minimum of .0935 inches accounts for the tolerances in reamed diameter, coating thickness, and the effect of roughness. Experience with graphite bead loaded elements with both NbC and ZrC coatings indicates that the above relationship applies for reamed diameters in the range from .098 to .112 inches and for a variety of coating thickness profiles. Some difficulty has been experienced in the past in the case of extremely rough coatings caused by large nodules or for molybdenum "whiskers" in the flow channels. In the reference R-1 design, an allowance of one mil in air impedance was reserved for the possible use of moly bore overcoatings. This resulted in the core pressure drop of 170 psi for the 0.0935 inch minimum effective impedance diameter as measured after the final carbide bore coating, but the as-orificed effective diameter was 0.0925 inches because of the anticipated effect of the moly overcoat. This one mil allowance has been retained for the parametric study since it is quite possible that moly overcoatings will be needed for additional mid-band corrosion protection. In the event it is not required, then the pressure drop estimates would be reduced by approximately 8 psi.

(CRD) It was found by metallographic studies that a major contributor to bore roughness for NbC coated graphite elements was the vacant bead pockets left in the reamed channel wall by the leaching process. Since the composite elements have no fuel beads it may be expected that smoother bores and relatively larger effective impedance diameters would result. Tentatively, this appears to be the case based on the data from six sample composite elements recently shipped to WANL. The impedance diameters also appear to be remarkably uniform within a coating batch although there is some batch-to-batch variability apparently due to the differences in coating thickness. It may be that eventually a well controlled production process would nearly eliminate the variability.

(CRD) Based on the above discussion, there is some reason for optimism if the reamed diameters remain in the vicinity of 100 mils and if the old "roughness bugaboo" does not re-appear. There is one small dark cloud on the horizon if reamed diameters are reduced substantially as contemplated for the 37 hole element. The roughness effect is really determined by the "relative" roughness or E/D ratio where E is the average height of the roughness elements or ("absolute" roughness) and D is the physical channel diameter. Thus, for a given matrix material and coating process the absolute roughness, E, should remain constant. But reducing the physical diameter, D, increases the relative roughness and the corresponding friction losses. At the present time, we have no data to support this assertion and if the composite elements continue to exhibit smoother bores, as expected, the overall effect of increasing E/D may still be negligible.

(CRD) In summary, the minimum impedance diameter after final carbide coating for all matrices and coatings in this trade study was assumed as follows:

$$D = D_r - .0041 - 2(t)$$

where:

D = effective impedance diameter, inches

D_r = nominal reamed diameter, inches

t = nominal average coating thickness, inches

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(CRD) To allow for the possible use of moly overcoating for mid-band corrosion protection, an additional one mil was subtracted to obtain the minimum hydraulic diameter used in pressure drop calculations.

(U) The above procedure is applied universally in the parametric studies providing a fair comparison between fuel element alternates, but the final fuel element design and coating must be represented by actual data before a final core pressure drop determination can be made using detailed analyses.

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A-1.4.2.2 (CRD) Temperature (U)

Assuming that the relative axial power distribution does not shift with changes in fuel element geometry,* the gas temperature profile axially in the channel will not change from the reference distribution given in Figure A-1.4.1-3. The corresponding channel wall temperatures and the adiabatic fuel hot spot temperatures will be affected by the changes in hole diameter and pitch.

This is because these parameters affect the radial transfer of heat through the matrix and across the convection film in the channel. As can be seen from Figure A-1.4.1-3 the largest radial temperature difference in the element is the convection film drop which is 199°R at station 48 inches versus 67°R for the conduction temperature drop in the matrix. The convection film drop is:

$$T_w - T_B = T_{\text{wall}} - T_{\text{bulk}} = \frac{Q}{h A}$$

where

Q	=	total heat transferred to the channel over a given axial increment
h	=	local heat transfer film coefficient
A	=	Surface area of the channel at that increment

For the exit end of the fuel element, beyond station 26 inches, we are currently using the McCarthy - Wolf correlation which is:

$$h = .025 \frac{k}{D} (\text{Re})^{.8} (\text{Pr})^{.4} \left(\frac{T_w}{T_B} \right)^{.55}$$

where:

K	=	gas conductivity at bulk temperature
D	=	channel hydraulic diameter
Re	=	Reynolds number evaluated at bulk temperature
Pr	=	Prandtl number evaluated at bulk temperature

* Analysis in Appendix A-3 indicates this is a very good assumption.

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Figure A-1.4.2-3 Maximum Channel Wall Temperature versus Hole Pitch (CRD) for Constant Reamed Diameter (U)

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For the hot end of the fuel element, which is of most interest, changes in the value of

$(T_W/T_B)^{-0.55}$ are negligible. Therefore, simplifying:

$$h \propto \frac{W^{0.8}}{D^{1.8}}$$

and continuing,

$$T_W - T_B \propto \frac{Q D^{0.8}}{W^{0.8}}$$

For constant reactor power, the value of Q and W are both inversely proportional to N, the number of coolant flow channels in the core. Therefore,

$$T_W - T_B \propto \frac{(D)^{0.8}}{(N)^{0.2}}$$

The number of flow channels per core is a direct function of the hole pitch, core fueled radius, and fueled/unfueled ratio. This is the relationship used to obtain relative wall temperatures given in Figure A-1.4.2-3.

(CRD) The wall temperature is of interest for calculating relative corrosion rates as a function of fuel element geometry. The maximum wall temperature is reduced by 60°R from the reference case by changing to 0.077 inch reamed diameters on a 0.132 inch pitch for constant coating thickness and core size from Figure A-1.4.2-3.

A-1.4.2.3 (CRD) Axially Varying Void Fraction (U)

(CRD) Suggestions have been made in the past that a void fraction which increases axially in the fuel element could be advantageous for a number of reasons. Among these are:

- 1) A thicker coating can be applied at the hot end without restricting the hydraulic diameter excessively.
- 2) The pressure drop per inch is highest at the aft end of the core because of the higher gas temperatures. A small increase in diameter at the aft end could be compensated by a corresponding reduction in diameter at the forward end (keeping total fueled volume constant) and giving a net reduction in pressure drop for a fixed coating thickness profile. The end point of this diameter adjustment, if pressure drop were the only criterion, would be a linear axial pressure drop profile which would be the minimum overall ΔP possible for a fixed axial heating distribution, fueled volume, flow rate and exit pressure.
- 3) Shifting the fueled volume axially will cause a shift in the axial heating distribution such that the discussion under (2) is only approximately true. Since the heating distribution will shift toward the forward end, maximum fuel temperatures are reduced at the aft end.
- 4) Careful "tailoring" of the axial channel diameter could conceivably result in a net reduction in the maximum thermal stress, but this must be analyzed very carefully by all disciplines including nuclear.
- 5) If the fuel elements can be supported at the forward end, the tapered channel gives a larger cross sectional area to reduce tensile stress at the forward end, while reducing weight hanging at the aft end. The web thickness is increased at the forward end, providing more room for attachments.

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(CRD) An MCAP analysis to compare an arbitrary tapered channel with the R-1 reference channel was carried out in the following manner:

1) The reamed diameter was assumed to vary from 0.087 inches to 0.117 inches in a series of six steps of 0.005 inch each.

2) The reference axial power distribution was adjusted by the ratio of the fuel volume fractions. This assumes that the neutron flux density is not affected by the change in void fraction axially, which is not correct, but gives at least a first order effect.

3) A channel coating profile varying in thickness from 0 at core station 0 to 4.4 mils at station 52 inches was assumed.

4) The hydraulic impedance diameter was determined by the method discussed in Section A-1.4.2.1.2 for each axial increment. It was assumed that a smooth transition in diameter change would take place at the 5 mil steps. This is relatively easy to accomplish by using a series of stepped reamers. Limited experiments have successfully been carried out using this method on graphite elements at WNCO. With composite elements, the problem of reamer wear would occur and make reaming less desirable, but a variety of different extruded sizes could be brazed together.

(U) The results of this limited study are presented in Figure A-1.4.2-4. The assumed reamed diameter, coating thickness and approximate hydraulic diameter are shown. The MCAP program interpolates between the input hydraulic diameters which were a series of steps. The channel flow rate and inlet and exit temperatures and exit pressure were constant for the two cases. The tapered channel has a "linear" axial pressure profile and a 14 psi lower overall ΔP . Because of the increased heat addition near the inlet, the gas temperature is higher all along the channel and the wall temperature is higher up to station 44". The maximum wall temperature is slightly lower.

(U) This case is certainly not optimized since thermal stress, nuclear and corrosion effects have not been considered. An integrated approach to optimizing a

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Ref: Core Size and Element A. F.
 $S = 0.173$ inch

MCAP Calculation

Figure A-1.4.2-4 Effect of Taper in Reamed Channel Diameter (U)
 (CRD)

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tapered channel would probably require the development of a specialized computer program for that purpose. One would expect that the present trade study is best conducted with a straight reamed bore diameter and thus avoiding all complications introduced by tapered channels. Incremental improvements, if desired, in any selected alternate could then be made, based on firm property and performance information from the development program, at a later date.

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A-1.4.2.4 (CRD) Effect of Thicker Bore Coatings on Impedance Diameter
and Pressure Drop (U)

(U) The PW-2 coating thickness profile was developed to protect the fuel elements for one hour's operation at the temperatures in the 50 minute NERVA design with 4500°R minimum chamber temperature. For the new conditions applicable to the 10 hour NERVA at 4250°R nominal chamber temperature, it appears that a thicker carbide coating at the aft end may be desirable. In this section, a desirable coating thickness profile is developed from two independent sets of corrosion test information and is shown to be compatible with the selected alternate fuel element designs.

(CRD) For a given total mass of coating, the minimum total weight loss occurs when the mass loss per unit length is a constant. This has been derived using the calculus of variation in DRM 51336. Previous experience with reactor testing has indicated that the rate of fuel corrosion tends to accelerate after the incremental loss exceeds 1 gm/inch.

(CRD) Using the axial temperature profile for the reference element given in Figure A-1.4.1-3, the coating thickness profile to give 1 gm/inch carbon loss by diffusion* can be calculated. Two independent sources of diffusion rates were used; the A6-C electrical test on NRX-A6 vintage NbC coated elements (Figure A-4.0-2) and the LASL data for the ZrC coated composite elements (Figure A-4.0-6). For the A6-C test data, the inverse temperature used was the channel wall temperature in the curve abscissa. The abscissa for the LASL data in Figure A-4.0-6 is the optically measured inverse temperature at the bottom of a drilled hole at station 45 inches in the fuel element. This measurement was reported by LASL to represent the fuel matrix temperature at that station. Accordingly, the fuel matrix inverse temperature (T_f) given by Figure A-1.4.1-3 was used to obtain the coating thickness profile using the composite reference line (Figure A-4.0-6).

* The assumption that the element ΔW occurs mostly by diffusion corrosion is almost a requirement for a successful 10 hour element -

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(CRD) The ordinates of both sets of data is a normalized incremental weight loss rate multiplied by the coating thickness for the element at the axial station in question. For any given inverse temperature the required value of $\left(\frac{\Delta W}{\text{inch} \cdot \text{time}} \times \text{thickness}\right)$ is obtained from the curve. For 1 gm/inch in 10 hours, the required coating thickness is then obtained directly. Results are shown in Figure A-1.4.2-5. The dotted axial profile represents the WANL A6-C test data, and the solid profile represents the LASL test data on composite elements. The two profiles are remarkably similar up to station 44 inches and differ by ~ 1 mil at the exit.

(CRD) For comparison, the measured coating thickness profile for 72 PW-2 development GEM-NbC, HED-ZrC elements has been added on Figure A-1.4.2-6. The complete profile for these elements is given on Figure A-1.4.2-7. The average coating thickness is 2.17 mils, almost exactly the thickness used to establish air impedance values in the parameter study. (See Section A-1.4.2.1.2)

(CRD) To obtain the desired profile at the aft end of the element, an assumed specified nominal profile is drawn on Figure A-1.4.2-6, starting at 2.8 mils at station 37.5" and increasing linearly to 6.0 mils at station 47.5 inches, then constant at 6.0 mils from 47.5 inches to 52 inches. By integrating the area between this assumed profile and the actual measured PW-2 development profile, the average added coating thickness over 52 inches is found to be 0.55 mils.

(CRD) By the ground rules used in the trade study, the minimum impedance diameter would be reduced by 1.1 mils to accomodate this added coating. This corresponds almost exactly to the 1 mil allowance which was retained to account for the possible addition of moly overcoat if needed to reduce mid-band corrosion.

(CRD) Since the aft end coating actually results in a "tapered" bore, a detailed analysis would be needed to account for it, and the effect on pressure drop would be somewhat more than an equivalent one mil in effective impedance diameter. This has not been done.

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Figure A-1.4.2-5 Axial Coating Thickness Profiles for Uniform 1 Gm/Inch Diffusion Corrosion (U) (CRD)

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Figure A-1.4.2-6 Axial Coating Thickness Profiles for Uniform 1 GM / INCH
(CRD) Diffusion Corrosion (U)

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FIGURE A-1.4.2-7 : AVERAGE COATING PROFILE-72 PW-2 DEVELOPMENT ELEMENTS. (U) (CRD)

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(CRD) Based on previous experience, for the 0.102 inch and larger reamed diameters the effect of the taper on core ΔP would not be significantly greater than assuming a uniform 1 mil reduction in diameter. If it were found desirable, it would be possible to ream the aft 10 inches or so of the fuel elements a few mils larger to accomodate the thicker coatings. This has been done on graphite elements at WNCO with successful results. As will be seen in the thermal and hydraulic discussion of the specific alternates for the trade study (Appendix A-4), alternates 8 and 9 have sufficient margin to accomodate a thicker aft end coating and moly overcoat without exceeding 170 psi core pressure drop. The other alternates can accomodate either the moly overcoat or the thicker aft end coating without exceeding 170 psi, and all can accomodate both moly and thicker aft end coatings without exceeding 180 psi in core pressure drop. Of course, the final determination of core ΔP will have to await equivalent diameter measurements on development elements with the desired coating profile. Coating roughness as well as thickness, is a major variable, and roughness cannot be predicted. Whether or not such a coating profile can be applied and remain crack-free in operation without over-stressing the fuel matrix also remains to be seen. All fuel alternates do retain potential (in varying degrees) to accomodate the thicker coating without exceeding reasonable core pressure drop limits.

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A-1.4.3 (CRD) EFFECT OF INCREASING CORE SIZE (U)

(U) One obvious way of reducing thermal stress in the fuel element is to increase the core size to reduce the fuel element power density. This has the added advantage of reducing core pressure drop (with a larger fueled diameter) and uranium loading requirements, but the considerable disadvantage of increasing reactor weight. It may, however, turn out to be the most practical way to improve the fuel element thermal stress margin. When one varies both the core size and fuel element geometry simultaneously, the number of independent variables to be handled increases geometrically. The parametric study almost becomes "open ended". However, it is known in advance that an increase in core length is easier to accomplish from the design standpoint than an increase in diameter.

A-1.4.3.1 (CRD) Effect on Core Pressure Drop (U)

(CRD) Increasing the core diameter at the reference core length increases the number of flow channels and the parametric relationship derived in Section A-1.4.2.1.1 is still valid. In addition, lengthening the core causes the core pressure drop to increase in nearly linear fashion, with other conditions constant. Both of these relationships were verified by MCAP runs for different size cores using the reference fuel element. The results are shown parametrically in Figures A-1.4.3-1 and A-1.4.3-2. Two families of curves are presented, one assuming no moly coating is used and the other assuming that the moly overcoat reduces the minimum equivalent impedance diameter by one mil. For the latter case, the core pressure drop is reduced from 170 psi to 145 psi by increasing the core fueled diameter by 2 inches. Alternately, the 170 psi can be held constant if

the core is lengthened to 70 inches and increased in fueled diameter by 2.6 inches, a factor of 1.54 increase in volume.

A-1.4.3.2 (U) Effect on Temperature (U)

Increasing the core diameter and number of flow channels will result in reduced fuel element temperatures with other conditions constant. The relationship for the channel wall temperature-to-gas temperature difference given in Section A-1.4.2.2 is valid for a fixed core length:

$$T_w - T_B \propto \frac{(D)^{.8}}{(N)^{.2}}$$

The temperature difference between fuel material, T_F , and channel wall, T_w , is reduced linearly with the number of flow channels for fixed element geometry. As discussed in Section A-1.4.2.2, this ΔT is relatively less significant than the wall-to-bulk temperature difference. Because of this and the fact that maximum fuel material temperatures are not major variables for trade-off, the change in channel wall temperature used for corrosion analysis has been emphasized.

Assuming that the relative axial power distribution does not shift with changes in core length, Figure A-1.4.3-3 shows axial material temperature profiles for three core lengths. The maximum temperature is reduced for the longer core.

The maximum channel wall temperature is presented in Figure A-1.4.3-4 as a function of core length for fixed fueled radius. The relative change in temperature is the same regardless of fueled radius indicating that the two effects are separable. Increasing the core length from 52 inches to 70 inches causes a reduction in the maximum wall temperature from 4389°R to 4346°R. In Figure A-1.4.3-5, the channel wall temperature at the axial station where the peak volumetric heating rate occurs is presented as a function of core length. A significant change in temperature from 3105°R to 2860°R takes place for an increase in core length from 52 inches to 70 inches at the reference diameter.

To provide more convenient interpretation for the above data for different fuel element geometries, Figures A-1.4.3-6 and A-1.4.3-7 have been prepared which give the percentage change in the wall-to-bulk temperature difference as a function of core length and fueled radius. A reference ΔT can be obtained for any fuel element geometry in the reference core size from Section A-1.4.2.2. This ΔT is then used as the reference value in Figures A-1.4.3-6 or A-1.4.3-7.

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(CRD)

Figure A-1.4.3-1 Core Pressure Drop Versus Core Length for Constant Core Fueled Radius (U)

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(CRD)
Figure A-1.4.3-2 Core Pressure Drop Versus Radius
for Constant Core Length (U)

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(CRD)
 Figure A-1.4.3-3 Fuel Hot Spot Temperature, T_F , Versus Axial Station (U)

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(CRD)

Figure A-1.4.3-4 Maximum Channel Wall Temperature versus Core Length for Constant Fueled Radius (U)

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Assumes relative axial
power distribution does
not shift with change
in core length

Reference Fuel Geometry

P.F. = 1.1

Dr = .102

Dh = .0935

S = .173

(CRD)

Figure A-1.4.3-5 Channel Wall Temperature at Maximum Volumetric Heating versus Length for Constant Fueled Radius (U)

(CRD)

Figure A-1.4.3-6: Relative Difference Between Core and Channel Wall Temperature Versus Core Length (U)

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(CRD)

Figure A-1.4.3-7: Relative Difference Between Gas and Channel Wall Temperature at Location of Maximum Volumetric Heating (U)

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A-1.4.4 (CRD) FUEL ELEMENT CORROSION EFFECTS OF VARYING FUEL ELEMENT GEOMETRY AND CORE SIZE (U)

A-1.4.4.1 (CRD) Fuel Element Carbon Diffusion Corrosion (U)

(CRD) The principal mechanisms in which fuel elements lose carbon by hydrogen corrosion are: (1) corrosion through cracks (low temperature corrosion as in the mid-band region and also occurs at high temperatures when cracks exist in the hot end region), (2) corrosion through defects in the carbide coating other than cracks, and (3) diffusion of carbon through the carbide coating. The mechanism of carbon diffusion is highly significant because even with a defect-free coating a fuel element would still lose weight by this mechanism. Weight loss by carbon diffusion begins to be significant at about 3800°R and increases exponentially with temperature. The loss of carbon by diffusion through the carbon coating from the substrate occurs because surface decarburization of the carbide coating by hydrogen establishes a carbon concentration gradient across the thickness of the coating. The diffusion relation for the carbon removed per unit area of channel bore is given in terms of coating thickness t , operating time θ , and the diffusion coefficient of the coating material, D :

$$\frac{W}{A} = \frac{D\theta}{t} \quad (1)$$

Applying the reciprocal temperature exponential form of the diffusion coefficient gives the carbon loss relation as a function of the temperature of the coating, T , and a coating material constant, C :

$$\frac{W}{A} \propto \frac{\theta}{t} e^{-\frac{C}{T}} \quad (2)$$

It can be postulated that the protection offered by the coating would be drastically reduced when a large amount of carbon had been removed from the substrate beneath the coating.

(CRD) Under these circumstances, coating defects would occur and lead to accelerating fuel element deterioration, similar to that observed in cyclic tests of some elements. Thus, the extent of diffusion carbon removal from beneath the coating is considered to be a measure of element corrosion deterioration and is developed as a corrosion limit for the composite and graphite elements. Equation (2) shows that this corrosion measure is dependent only on the coating material and thickness and the operating temperature of the coating. The fuel element geometry and core size will only affect this corrosion parameter because they affect the axial temperature profile (See Figure A-1.4.3-3 shown in the preceding section).

(U) In section A-4.0.1 some reactor and electric furnace test data which substantiates the use of the diffusion equation are given. Based on this information, the relative diffusion corrosion depth behind coatings are developed as follows for graphite and composite fuel.

A-1.4.4.2 (CRD) Graphite Composite Fuel Carbon Diffusion Corrosion (U)

(CRD) Carbon diffusion losses which are correlated by equation (2) would be represented by a straight line on a reciprocal temperature semilogarithmic graph paper. Figure A-1.4.4-1 shows the corrosion correlation based on corrosion test data discussed in Appendix A-4, Section A-4.0-1, and is used to parameterize the effect of fuel element geometry on corrosion. In this study, only the diffusion corrosion mechanism is considered and the coating is assumed to be defect-free. As such, the corrosion parameters developed are descriptive of graphite, as well as composite fuels.

(CRD) On the basis of the above data, the slope of the diffusion corrosion line for NbC or ZrC was estimated to be equal to that of the reference line. This was used to establish the relative corrosion deterioration parameters given in Figures A-1.4.4-2, A-1.4.4-3, and A-1.4.4-4.

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A-1.4.4.3(U) Fuel Element Corrosion Reactivity Loss (U)

Reactivity loss due to fuel element corrosion is obtained by combining the carbon reactivity worths, C, given in the Nuclear Section 5, of this Appendix, and the corrosion carbon loss, W;

$$\Delta P = \int_0^L \int_0^R W(r, z) C(r, z) 2\pi r dr dz \quad (6)$$

However, the carbon worth can be given as the product of an average worth and normalized radial and axial distribution functions, while the corrosion can be considered to be directly proportional to the bore coating area and independent of radius

$$\Delta P \sim \bar{C} \frac{D}{S^2} D_c^2 L \int_0^L F(z) g(z) dz$$

where

ΔP	=	reactivity loss due to corrosion
D	=	reamed diameter
S	=	bore centerline pitch
D_c	=	core diameter
L	=	core length
$\bar{C}$	=	average carbon worth as a function of D, S, D_c , L and fuel
F(Z)	=	normalized axial carbon worth distribution
g(Z)	=	normalized axial corrosion distribution

Assuming F and g functions are independent of core length and fuel the reactivity loss normalized to the reference R-1 is given in Figure A-1.4.4-5.

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Figure A-1.4.4-1 Normalized Carbon Diffusion Rate versus Inverse Temperature (U) (CRD)

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Figure A-1.4.4-2
(CRD)

Normalized Corrosion Rate Versus Hole Pitch for
Constant Reamed Diameter (U)

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Reference Core Diameter and Length Maximum
Nominal Wall Temperature

Figure A-1.4.4-3 Normalized Carbon Removal Depth Versus Hole Pitch (U)

Figure A-1.4.4-4 Percent Carbon Removal From Minimum Nominal Web Dimension Relative To Reference Design (U)

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Figure A-1.4.4-5
(CRD)

Normalized Reactivity Loss Versus Hole Pitch (U)

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A-1.4.5 (CRD) SUPPORT STEM FLOW CONTROL EFFECT OF VARYING CORE DIAMETER AND LENGTH (U)

(U) A study was performed to determine the effect on support stem hydrogen inventory from core geometry variations (length and diameter). This data was used to evaluate the flow control reactivity adjustment obtained for geometry variations.

ASSUMPTIONS (U)

- (CRD) 1. Core diameters of 37.31 inches, 43.08 inches and 52.76 inches were investigated. The fuel elements were graphite, 0.753" across flats with nineteen .095 coolant channels. A single cluster TAP-B computer model with 6-1 fuel to unfuel ratio was used for the analyses.
- (CRD) 2. Hydrogen inventory was tallied by multiplying the hydrogen masses times the hydrogen worths to obtain reactivity feedback. The R-1 reference hydrogen worth was used for the 37.31" diameter core (see Table A-1.4.5-1. Hydrogen worth used with the 43.08" diameter core was .953 times the reference hydrogen worth and for the 52.76" diameter core, .871 times reference hydrogen worth was used (see Figure A-1.5-15 in the Nuclear Section).
- (U) 3. The minimum support stem flow was set so that the average support stem would have a nominal maximum temperature of 1000°R.
- (U) 4. The maximum stem flow was determined by assuming a necessary \$3.00 feedback reactivity variation between the minimum and maximum support stem flow settings.
- (U) 5. Bypass flow and support stem flow were assumed to total 21 lb/sec unless the stem flow exceeded 19 lb/sec. For these high stem flow cases, the bypass flow was assumed to maintain a minimum flow of 2 lb/sec, which would result in less coolant available for the nozzle and reflector. Analyses of these two components would have to be performed before we can determine if the 21 lb/sec stem and bypass flow can be exceeded.
- [REDACTED]

6. The hydraulic impedances of the shields, nozzle and reflector were assumed to remain the same as in the present reference design. The turbine and pump parameters were also assumed to retain their present values. The differences in support stem exit pressure for each of the core geometries were based only on differing core pressure drops. The support stem inlet pressure reported for maximum stem flow is the pressure needed at the stem inlet to drive the hydrogen through the stems at maximum stem flow.

A-1.4.5.1 Varying Core Diameter

The hydraulic performance indicated by this study for variable core diameter at constant core length and present reference fuel element geometry is shown on Figure A-1.4.5-1. Increasing core diameter causes increases in the maximum necessary stem flow, but causes the maximum necessary driving pressure differential through the stem to decrease. A possible limit exists for core diameters greater than 48" because the maximum necessary stem flow reaches 19 lb/sec for a 48" diameter core. Stem flows greater than 19 lb/sec require flow to be taken from the nozzle and reflector as mentioned in Assumption 5.

Since fuel element geometry remained constant for these analyses, core pressure drop decreased as the core diameter increased, which resulted in decreased stem exit pressure. Changing the fuel element configuration to change core pressure drop would have very little effect on the stem controllability as presented. If different ΔP fuel elements were used to maintain a constant core ΔP with core diameter variation, the support stem exit pressure curve would have a slope of zero and the inlet and exit pressure curves would retain about the same relationship to each other.

According to the above mentioned data, the present stem flow control can be maintained without major design configuration changes for core diameters up to 48 inches.

A-1.4.5.2 Varying Core Length

The hydraulic performance indicated by this study for variable core length at constant core diameter and present reference fuel geometry is shown on Figure A-1.4.5-2.

Core lengths between 40 and 70 inches were investigated. Increasing the core length decreased the total stem flow needed to achieve the \$3.00 reactivity swing, while the necessary driving ΔP increased only slightly. One restriction, however, comes if the core length is decreased to less than 45". The maximum necessary stem flow for a shorter core would exceed 19 lb/sec. See Assumption 5 for restrictions involved beyond this point.

Due to lack of available data, hydrogen worth was assumed constant with core length variations. Actually, the hydrogen worth could decrease by as much as 10% for a 10 inch increase in length. These reactivity changes will cause some change in the flow curves.

It can be concluded that the present flow control can be maintained with a longer core without major configuration changes.

TABLE A-1.4.5-1

HYDROGEN WORTH IN REFERENCE R-I REACTOR

<u>Core Station (X/L)*</u>	<u>Hydrogen Worth (\$/lb)</u>
0.0	.0839
0.0769	.889
0.1538	1.846
0.2308	2.585
0.3077	3.324
0.3847	3.814
0.4615	4.014
0.5385	3.90
0.6154	3.515
0.6923	2.744
0.7692	2.109
0.8462	1.147
0.9231	.585
1.00	.060

* X = Distance along the core from the core inlet end

L = Length of the core

Reference: DRM 51519

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FIGURE A-1.4.5-2 (C-RD) SUPPORT STEM FLOW AND PRESSURE REQUIREMENTS FOR CORE LENGTH VARIATIONS (U)

W59831-1

1. Core is 52 inches long with a 6-1 fuel - unfuel element ratio.

2. Hydrogen worth relative to present R-1 reference reactor

Core Diameter	Relative Hydrogen Worth
37.31 inches	1.00
43.08 inches	0.952
52.76 inches	0.871

SUPPORT STEM PRESSURE (PSIA)

Larger core diameters would require over 21 lb/sec stem and bypass flow (19 lb stem flow and 2 lb bypass)

SUPPORT STEM FLOW (LB/SEC)

FIGURE A-1.4.5-1 (C-RD) SUPPORT STEM FLOW AND PRESSURE REQUIREMENTS FOR CORE DIAMETER VARIATIONS (U)

(CRD) A-1.5.1 NUCLEAR EFFECTS OF VARYING COOLANT VOLUME FRACTION AND CHANNEL COATING THICKNESS (U)

(CRD) If the fuel coolant channel diameter is increased from the reference 102 mil dimension, or if the hole pitch (center-to-center spacing) is reduced while maintaining a channel diameter, the volume and mass of solid fuel material in the core is reduced. The reduction in average core density increases neutron leakage and, therefore, increases the uranium mass required for criticality. This increase, together with the reduced solid fuel volume may result in a very high volumetric uranium loading. The reference size of the R-1 reactor results in the use of a uranium loading of 630 mg/cc of fuel in elements of the highest loading grade with 75 gm Nb per element. As this is approximately the limit for graphite fuel, the R-1 reactor may be termed criticality-limited with respect to this fuel. The problem is minimized with composite and carbide fuels on the bases of higher allowable loadings. Therefore, a more extended discussion of the graphite fuel is given in this section.

(U) In the following subsections the variation in nuclear characteristics with void fraction are presented for a reactor of the reference dimensions, for each of the four fuel types of interest. Nuclearly, the first two candidates, high CTE and intermediate CTE graphite fuel are almost identical but will be discussed separately for consistency with other parts of this trade study.

(CRD) A-1.5.1.1 High CTE Graphite Fuel (U)

(CRD) Figure A-1.5-1 shows the volumetric loading required in a radially zoned graphite core of the reference size as a function of fuel void fraction. In a radially zoned core, the highest loading is typically 1.2 times the average loading. The lowest curve is for an uncoated element. The intersection of this curve, with the 630 mg/cc technology limit for the high CTE graphite element at about 37% void fraction, indicates the criticality limit for uncoated fuel. Any further increase in flow area (void fraction) would require changes to increase reactivity, such as thickening the reflector. (Larger cores would also relax the loading restriction, as discussed in Section A-1.5.2.)

(CRD) If the fuel is coated with NbC, the critical mass and, consequently, the maximum uranium loading increases. The upper family of curves in Figure A-1.5-1 describe fuel elements with a constant 2.2 mil NbC coating on each channel but with various channel diameters

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and void fraction. For a given channel diameter, increases in void fraction are achieved by decreasing hole pitch, i.e., increasing the number of holes. (This relation is given in Eq. (5) below.) The mass of niobium in the core increases approximately in proportion to the NbC coating thickness, the channel diameter and the number of fuel channels N in the core:

$$Nb \sim t_{NbC} \cdot D_C \cdot N \quad (1)$$

The number of fuel channels and the channel diameter determine the fuel void fraction in a core of fixed size:

$$VF \sim N D_C^2 \quad (2)$$

Combining these relations, the niobium mass in a core of fixed size is given by

$$Nb \sim \frac{t \cdot VF}{D_C} \quad (3)$$

Thus, Nb mass varies inversely with channel diameter if void fraction is kept constant.*

(A more familiar relation is that Nb mass is directly proportional to channel diameter if hole pitch is kept constant and void fraction is allowed to increase.) For the 2.2 mil NbC thickness assumed in Figure A-1.5-1, the associated Nb mass is 75 gm/element in the reference fuel geometry (102 mil reamed channels and 173 mil pitch, giving a 31.4% void fraction).

For other geometries, the mass is given by the relation

$$Nb \text{ mass, gm} = 75 \left(\frac{VF}{0.314} \right) \left(\frac{102}{D_C} \right) \quad (4)$$

where the channel diameter is in mils.

(CRD) Figures A-1.5-1 and A-1.5-2 (which shows the loading requirements for a 4.4 mil NbC coating thickness) provide the nuclear limits for each combination of void fraction and channel size. From the curve intersections with the 630 mg/cc loading limit, Table A-1.5-1 has been compiled to show the maximum feasible void fraction at each fuel void fraction. This, in turn, implies a minimum feasible pitch for that channel diameter, given by the equation

$$VF = \frac{(\pi/2\sqrt{3}) (D_C/P)^2}{\quad}$$

* This implies a pitch proportional to the channel diameter to shim these limits.

from which the pitch P is given by

$$P = (\pi/2 \sqrt{3})^{1/2} D_C (VF)^{-1/2} \quad (5)$$

$$P = 0.952 D_C (VF)^{-1/2}$$

The table also indicates that reductions in channel diameter do not yield a corresponding decrease in fuel web thickness for cores of equal reactivity.

(CRD) Figures A-1.5-1 and A-1.5-2 were based on an assumed excess reactivity of \$3.0. The consequence of a reduction in the excess shim reactivity would be that lower uranium loading densities or larger void fractions are possible. For the reference design, a \$1.5 reduction in excess reactivity requirement corresponds to a 29 mg/cc reduction in maximum uranium loading. From the slope of the curve for 102 mil channels in Figure A-1.5-1, this may be related to a possible increase in void fraction to 33%, or in other words a reduction in pitch from 173 mils to 169 mils.

(CRD) Figure A-1.5-3 presents the maximum loading curves for the reference High CTE design. Here hole pitch is maintained at 173 mils (implying the number of channels is 1878 x 19 in the reference core size), but channel diameter varies. Figure A-1.5-3 shows that if ZrC coating were applied to graphite element, the criticality limitation is relaxed; the maximum required loading at the reference 102 mil, 31.4% VF case decreases from 630 mg/cc (NbC coating) to 502 mg/cc (Zr coating).

(CRD) In the above discussion, potential reactivity losses due to increasing void fraction were assumed to be compensated by increasing uranium loading. To a good approximation, this maintains the neutron leakage from the core constant. Accordingly, the reflector thickness required to provide about \$5.5 control is nearly independent of void fraction, and does not impose an additional restriction on fuel geometry.

(CRD) Figure A-1.5-4 presents the reactivity worth coefficients for carbon and hydrogen as a function of fuel element void fraction. Values are presented for the High CTE graphite, composite and carbide fuel element designs. As shown, the reactivity worths of hydrogen and carbon increase in the composite and carbide fuels relative to the high CTE graphite. Figure A-1.5-5 shows the axial carbon reactivity worth profile for the High CTE graphite fuel and the 31.4 percent fuel void fraction.

[REDACTED]

(CRD) A-1.5.1.2 Intermediate CTE Fuel (U)

(CRD) Figures A-1.5-1 and A-1.5-2 can also be used to determine the uranium loading requirement as a function of void fraction for a core of GLFI graphite fuel elements. The maximum loading difference between high and intermediate is only 9 mg/cc. This material exhibits a coefficient of thermal expansion intermediate in the range of graphite properties. Table A-1.5-1 may also be utilized to identify the channel size and spacing limitations for NbC-coated GLFI fuel as well. Figure A-1.5-6 presents the maximum loading requirement as an explicit function of channel diameter, similar to Figure A-1.5-3 of the preceding discussion.

(U) Since thermal expansion of the core produces a small negative temperature coefficient of reactivity, the cold-to-hot reactivity swing would be slightly less than with high CTE fuel. This difference would not change the control drum span requirement significantly and is unimportant with respect to this trade study.

(U) The carbon and hydrogen worths presented in Section A-1.5.1.1, Figure A-1.5-4 are applicable to the intermediate CTE fuel as well.

(CRD) A-1.5.1.3 Composite Fuel (U)

(CRD) A 30 v/o ZrC-UC, 70 v/o graphite fuel element is discussed in this section. With the current core and fuel geometry, the core would contain about 1100 kg Zr and 160 kg less carbon than a high CTE graphite fueled core. The core weight at the same uranium loading would, therefore, be 940 kg, or 2070 pounds heavier than a high CTE graphite fueled core. The critical mass and maximum uranium loading are only slightly higher than an uncoated or ZrC-coated graphite element. This may be seen by comparing the volumetric loadings for composite fuel, Figure A-1.5-7, with those shown in Figure A-1.5-1. As in the previous sections, the maximum loading is also presented as a function of channel diameter (See Figure A-1.5-8.),

(CRD) One important difference arises from the change in material properties. With the composite fuel, the uranium content per unit volume of fuel may be increased to 700 mg/cc. The limiting void fraction for this fuel is, thus, 36.6% from Figure A-1.5-7.

(U) The control drum span is nearly the same for the composite-fueled as for a graphite-fueled core. Figure A-1.5-4 of Section A-1.5.1.1 shows the carbon and hydrogen reactivity worths as a function of fuel element void for the composite fuel. The corresponding axial carbon worth is shown in Figure A-1.5-9.

(CRD) A-1.5.1.4 Carbide Fuel (U)

(CRD) Figure A-1.5-10 indicates the maximum uranium loading in zoned cores using $ZrC \cdot UC$ fuel with various void fractions. With this fuel, the limiting uranium content is over 2000 mg/cc allowing use of a 64% void fraction. As in the previous two sections, the maximum loading is also presented as a function of channel diameter (See Figure A-1.5-11.).

(CRD) Because of its high zirconium content (2900 kg, as compared to 1100 kg for the composite fuel), the carbide fuel core weighs about 4800 pounds more than a graphite core. The reflector thickness for $\$5.6$ reactivity control is approximately the same as for the graphite fuel. Figure A-1.5-4 of Section A-1.4.11 shows the carbon and hydrogen reactivity worths as a function of fuel element void for the carbide fuel. The corresponding axial carbon worth is shown in Figure A-1.5-12.

[REDACTED]

(CRD) A-1.5.2 NUCLEAR EFFECTS OF VARYING CORE SIZE (U)

(U) This section presents the nuclear effects on fuel loading and material reactivity worths of increased core size. The core is expanded both radially and axially. The maximum critical uranium loadings are developed for each design size for a ± 3.0 total shim capability. In addition to the absolute reactivity calculations, the reactivity coefficients for carbon and hydrogen are developed as a function of core size. These nuclear effects are generated for the High CTE graphite, composite and carbide fuel element designs.

(CRD) A-1.5.2.1 Fuel Loading Effects (U)

(CRD) To support the optimization, the fuel element geometry for R-1, a parametric sizing of study was performed. This study was performed using a fuel element void fraction of 31.4 percent. For a 19-hole element with a 0.753-inch across the flats dimension, this corresponds to a reamed channel diameter of 0.102 inch. The variables considered were maximum fuel loadings, type of fuel, core radius and height. Preliminary results for the three core compositions (high CTE, composite and carbide) were obtained with ANISN (one-dimensional discrete ordinate transport code) calculations using both a 52- and 78-inch core height and four core diameters (37.31, 39.81, 42.31 and 52.0 inches). The reduction in maximum fuel loading as a function of core radius for 52-inch high core is shown in Figure A-1.5-13. Similar results for 78-inch core height are shown in Figure A-1.5-14.

(CRD) A-1.5.2.2 Coefficient Effects (U)

(CRD) Figure A-1.5-15 illustrates the effect of increasing core diameter on the core average and hydrogen reactivity coefficients. These values are for a 52-inch height core. Shown are values for all three fuel types and for the reference fuel element void fraction, 0.314. Figure A-1.5-16 shows the same results for a 0.40 void fraction fuel element. The effect of the higher void is to increase the carbon worth about 8.5% and the hydrogen worth about 4.1%.

(CRD) A-1.5.3 NUCLEAR EFFECTS OF VARYING THE RATIO OF FUEL TO UNFUELED ELEMENTS IN THE REFERENCE SIZE CORE (U)

(CRD) In this section, the effect of doubling the number of central support elements in a core of the reference size will be considered. The reference core contains 1878 fuel elements and 349 central elements, each of the latter containing a support stem. In the central part of the core, the ratio of fuel to central elements is 6:1. In the periphery, the number of central elements is increased, bringing the core average ratio to 5.37.

(CRD) Assuming that core size is constant and doubling the number of central elements to 698, the number of fuel elements must decrease to $(1878 - 349 = 1529)$, an 18.6% reduction in fuel volume.

(CRD) A-1.5.3.1 High CTE Graphite Fuel (U)

(CRD) Substitution of 349 graphite central elements for NbC-coated high CTE graphite fuel elements produces a reactivity loss of approximately $\$5.0$ for the dry reactor. No compensating loading increase is possible with this fuel, because the maximum fuel loading of 630 mg/cc is required in the reference reactor. Thus, the hypothetical reactor cannot be made critical in the reference size. A compensating change, such as a substantial reduction of void fraction, would be required to restore the reactor to critical. .

(CRD) The hydrogen worth in the stem system would be 518¢/kg if axially uniform, at 31.4% void fraction and would be slightly higher values at higher void fractions (see Figure A-1.5-4).

(CRD) A-1.5.3.2 Intermediate CTE Graphite Fuel (U)

(CRD) The discussion of the preceding section applies equally to the intermediate CTE graphite element; i. e., the void fraction must be rather drastically reduced to maintain loadings below the 630 mg/cc limit.

(CRD) A-1.5.3.3 Composite Fuel (U)

(CRD) The composite fuel element, because of its capacity to carry a greater uranium loading per unit volume, is less severely affected by the loss of fuel volume. The critical mass is not greatly affected: the net reactivity effect of the exchange of 349 fuel elements

[REDACTED]

for ZrC-graphite support elements is a \$1.93 reactivity loss due mainly to a reduction in graphite content, not including the uranium content of the deleted fuel elements. The critical mass increases from 225 kg as specified in Section A-1.5.1 to 242 kg for the reference 31.4% void fraction. The maximum uranium volumetric loading increases much more rapidly, from 579 mg/cc in the reference case to 765 mg/cc. As this is in excess of the current estimated limit of 700 mg/cc, the proposed exchange cannot be made unless the limitation is relaxed, the core dimensions increased, or the void fraction is reduced.

(CRD) If less than 349 elements are exchanged, the change becomes feasible. The new stem worth may be based on a uniform hydrogen worth of \$5.6/kg (at 31.4% void) weighted with the axial worth curve previously furnished for hydrogen. (See Figure A-1.5-4 for other void fraction.)

(CRD) A-1.5.3.4 Carbide Fuel (U)

(CRD) The exchange of 349 carbide fuel elements for central elements resulted in a \$0.56 reactivity loss. As a result, the critical mass increases slightly, from 388 kg to 394 kg. The maximum uranium loading increases from 900 mg/cc to 1129 mg/cc, which exceeds the recommended limit for this fuel.

(CRD) The worth of hydrogen in the carbide-fueled core may be taken as \$7.1¢/kg for a uniform distribution, at 31.4% void fraction. Figure A-1.5-4 indicates the worth of hydrogen for other void fractions.

TABLE A-1.5-1

(CRD) LIMITING FUEL VOID FRACTION AND COOLANT CHANNEL PITCH
IN NbC-COATED GRAPHITE ELEMENTS (CRD)

Channel Diameter mils	2.2 mils NbC				4.4 mils NbC			
	Nb Mass gm/ele.	Maximum V.F. %	Minimum Pitch mils	Minimum Web mils	Nb Mass gm/ele.	Maximum V.F. %	Minimum Pitch mils	Minimum Web mils
77	93.5	29.6	135	58	171	25.2	146	69
94	79.5	30.7	165	71	138	26.6	172	78
102	75*	31.4*	173*	71*	130	27.1	186	84
125	63.4	32.5	215	90	111	28.5	223	98

* Reference case

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Figure A-1.5-1. (CRD) Maximum Uranium Loading Required in Reference Size Graphite Fueled Cores (U)

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Figure A-1.5-2. (CRD) Maximum Uranium Loading Required in Reference Size Graphite Fueled Cores (U)

FIGURE 5-3

(CRD) MAXIMUM URANIUM LOADING VERSUS REAMED CHANNEL DIAMETER
FOR THE HIGH CTE FUEL DESIGN (U)

Figure A-1.5-4. (CRD) Reactivity Worths of Carbon and Hydrogen as a Function of Fuel Element Void Fraction (U)

Figure A-1.5-5. (CRD) Carbon Axial Worth Profile for High CTE Graphite Fuel (U)

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FIGURE 5-6

(CRD) MAXIMUM URANIUM LOADING VERSUS REAMED CHANNEL DIAMETER FOR THE INTERMEDIATE CTE FUEL DESIGN (U)

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Figure A-1.5-7. (CRD) Maximum Uranium Loading Versus Fuel Void Fraction for the Composite Fuel Design (U)

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FIGURE 5-8 (CRD) MAXIMUM URANIUM LOADING VERSUS REAMED CHANNEL DIAMETER FOR THE COMPOSITE FUEL DESIGN (U)

FIGURE 5-9

(CRD) CARBON AXIAL WORTH PROFILE FOR COMPOSITE FUEL (U)

Figure A-1.5-10. (CRD) Maximum Uranium Loading Versus Fuel Void Fraction for the Carbide Fuel Design (U)

FIGURE 5-11 (CRD) MAXIMUM URANIUM LOADING VERSUS REAMED CHANNEL DIAMETER FOR THE SOLID CARBIDE FUEL DESIGN (U)

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FIGURE 5-12

(CRD) CARBON AXIAL WORTH PROFILE FOR SOLID CARBIDE FUEL (U)

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Figure A-1.5-13. (CRD) Reduction Maximum Fuel Loadings Due to Change in Core Diameter with 52 Inch Core Height (CRD)

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Figure A-1.5-14. (CRD) Reduction Maximum Fuel Loadings Due to Change in Core Diameter with 78 Inch Core Height (CRD)

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Figure A-1.5-15. (CRD) Effect of Core Diameter on the Hydrogen and Carbon Reactivity Coefficient (U)

Figure A-1.5-16. (CRD) Effect of Core Diameter on the Hydrogen and Carbon Reactivity Coefficient (U)

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(CRD)A-1.6 FUEL ELEMENT AND REACTOR DESIGN CONSIDERATIONS (U)

The following evaluation defines the fuel element size limits and core sizes (diameter and/or length) limits as a function of manufacturing, axial core support and lateral core support requirements.

A-1.6.1 Fuel Element

(U) Fuel element nominal size limits due to manufacturing are a function of four specific dimensions; that of, flow channel size, web thickness (material thickness between a flow channel and periphery) across-flats size of the element and element length.

A-1.6.1.1 Flow Channel Size (As Reamed)

(U) Minimum - Estimated to be 0.060 inch
Maximum - None

Considerations

The limitations on flow channel size are related to element length, flow channel location tolerance requirements, web thickness, and overall element across-flats dimension. For fuel element lengths between 40 and 100 inches, the minimum flow channel size is dependent upon the location tolerance and the minimum web thickness. There is no maximum flow channel size limit, other than, that which would be set by the geometrical limitations of the element configuration, number of flow channels, and the minimum web thickness.

A-1.6.1.2 Web Thickness

(C-RD) Minimum - 0.015 inch
Maximum - 0.350 inch

Considerations

The minimum web thickness is controlled by machining difficulties, number of flow channels, overall element dimensions and element permeability (permeability of graphite matrix elements increases exponentially as the web thicknesses decreases). Based on previous

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experience, a limit of 0.015 inch is a practical lower limit for a multiple flow channel element 40 to 100 inches long. The major difficulty which sets the upper limit for web thicknesses is due primarily to element density variations and matrix cracking problems due to binder decomposition during element curing. Based on production experience of thick-walled central elements, web thicknesses greater than 0.350 inch appear to be feasible if long curing times are used. The practical upper limit for unfueled graphite may be about 0.750 inch as based on extrusions made for peripheral unfueled partial elements of previous NRX reactors.

A-1.6.1.3 Across Flats

- (C-RD) Minimum - 0.30 inch.
Maximum - 1.5 - 2.0 inches.

Considerations

The minimum across-flats dimension is based on the assumption of at least a seven flow channel element with minimum .060 inch flow channel size and minimum 0.015 inch web thickness. Although small across flats dimension will cause increases in tolerances (twist, bow, across flats), it is assumed the tolerances can be brought into permissible size by proper tooling and fixturing. The maximum limit is based on the anticipated sagging of the as-extruded elements made of heavier materials (i.e. composites).

A-1.6.1.4 Length

- (U) Minimum - 40 inches.
Maximum - 100 inches.

Considerations

No major problems are expected to be caused by length changes provided the recommended minimum across flats and web thickness are maintained. Coating difficulties are anticipated for very long and small-size flow channels, however, it appears that by proper zone controls in the coating furnaces the difficulties in coating can be overcome. Shorter fuel elements do not cause additional problems.

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A-1.6.2 (C-RD) AXIAL SUPPORT (U)

A-1.6.2.1 Fuel Element Size Limits

(U) Fuel element size limits due to axial support requirements are imposed by the minimum spacing of support stems in the core support plate and minimum aft-end bearing area of the support element.

(C-RD) The minimum spacing of the support stems in the core support plate, based on the requirement of 1.400 inch support stem plenum diameter needed for support stem flow, is 1.830 inch. The minimum across-flats dimension of an element, based on the support stem plenum diameter, is 0.692 inch.

A-1.6.2.2 Central Element Creep

(C-RD) One of the possible limitations on core pressure drop is imposed by creep of the central support element. If the cluster support load imposed on the central element is too high, creep will shorten it and cause excessive lateral growth. Excessive lateral growth will cause unbundling of the fuel and excessive interstitial gaps.

(C-RD) Unfueled graphite has demonstrated ample creep resistance to avoid stringent limitations on core pressure drop. Extruded ZrC-C composite, on the other hand, has not yet been well characterized with respect to creep. From experience with hot pressed NbC-C composite, consideration of the less refractory nature of ZrC, and the relatively poor creep characteristics of extruded material it appears prudent to limit the core pressure drop to approximately 170 psi when composite support elements are used.

A-1.6.3 (C-RD) LATERAL SUPPORT (U)

(C-RD) Core size limits due to lateral support are imposed on the core diameter and fuel element density. An increase in present reference (940J150) core diameter and increase in fuel element density greater than 1.45 times the density of fuel elements made of AXM graphite (30 v/o UC-ZrC-graphite composite) is limited by the capability of the reference lateral support system. Increased core length is not limited by the lateral support.

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A-1.7.0 (C-RD) EVALUATION OF AN INCREASED NUMBER OF SUPPORT STEMS (U)

(U) The following evaluation defines the effects of increased number of support stems. This concept was found to be not feasible because of the increased pressure drop through the axial flow paths in the core support plate.

A-1.7.1 Fuel Element Size

(C-RD) The increased number of support stems increases the minimum fuel element size, because of the need for adequate support stem plenum diameter. The minimum spacing of support stems in the core support plate, for three fueled elements per support stem, is 1.512 inch and the resulting across-flats dimension of an element is 0.754 inch.

A-1.7.2 Pressure Drop Limitations

(C-RD) The axial flow paths in the core support plate consist of 492 holes of 0.437 inch diameter giving a flow area of 74 square inches. As shown on Study Layout 914E542, the increased number of support stems requires a reduction in the hole diameter to 0.222 inches, with 560 flow holes or an area of 22 square inches. The rigidity of this new design is adequate provided the overall pressure drop (core plus support plate) stays at about the same level as the reference design. The pressure drop across the plate can be found from the reference value using the parameter relationship developed in Section A-1.4.2.1. The reference ΔP across the support plate is 11.3 psi.*

(C-RD) For the support plate design on layout 914E542, the estimated axial pressure drop across the plate is ~ 150 psi. This allows only ~ 31 psi for the core to maintain the same overall ΔP . Considering that the actual core ΔP would increase by about 30% from the value with the standard cluster, due to the reduction in the number of fueled elements, this concept is not feasible.

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[REDACTED]

A-1.8 (C-RD) WEIGHT EVALUATION (U)

(U) This section contains parametric information relating the weight of the nuclear shield (Section A-1.8.1) variations in the fuel element matrix material and void fraction and to changes in core size. Also given in Section A-1.8.2 is a parametric study of reactor weight (less shield) as a function of core diameter, length, and fuel element matrix material.

A-1.8.1 (U) Shield Weight (U)

(U) The effect of variations in (1) carbide (ZrC) concentration in the fuel, (2) core diameter, (3) core length, and (4) fuel element void fraction on shield weight was determined.⁽¹⁾ The shield weight was varied in a manner such that both the neutron and photon dose rates at 30 feet from the core center on the reactor centerline did not increase above their current levels. Figures A-1.8-1 through A-1.8-4 display the relative effects of these variations on the reference design BATH shield which weighs 3850 pounds.

(U) The shield weight for any arbitrary combination of variations in the previously mentioned parameters is determined in the following manner:

1) Determine the value of the relative shield weight factors for each of the four parameters from the appropriate curve, and

2) The shield weight (in pounds) is the product of each of the four relative shield weights factors and 3850.

A-1.8.2 (C-RD) Reactor Weight (Less Shield) as a Function of Core Diameter, Length and Fuel Element Material (U)

(U) The reactor weight dependency on core diameter, core length and fuel element material is described below and in Figures A-1.8-5 through A-1.8-7.

(C-RD) Items (A) through (G) list boundary conditions and assumptions used.

(1) All results were obtained by KAP point kernel calculations which assumed that the total source strength in the core remained unchanged. (See A-10-7 for details).

[REDACTED]

- A. Reference core diameter = 38.68 inches (including wrapper and filler strips)
- B. Reference core length = 52.00 inches
- C. 38.68 inches < core diameter < 54 inches
- D. 40 inches < core length < 100 inches
- E. Thermal and pressure conditions are the same as in reference design shown in 940J150
- F. Core periphery and unfueled elements are made of material of same average density as the fueled elements.
- G. All reactor components such as: vessel, forward closure, reflector, support plate, piping, axial and lateral supports are made from material of same density as reference design shown in 940J150.

(U) The reactor weights (less shield) are obtained as proportional ratios based on governing (stress or deflection) changes of components as a function of core length and/or diameter. The resultant formula for obtaining reactor weight (core made of a specific fuel element material) ratio is as follows:

$$\frac{W^1}{W_o} = \left\{ A n^4 + [B n^2 + C n] m \right\} \frac{1}{W_o}$$

W^1 = Weight of Reactor (Lbs.)

W_o = Weight of Reactor (Lbs.) of Reference (940J150) Size

A = Weights of Closure & Support Plate (n = m = 1)

B = Sum of Weights of Vessel, Reflector & Core (n = m = 1)

C = Sum of Weights of Piping, Axial & Lateral Supports (n = m = 1)

n = Ratio of Core Diameters

m = Ratio of Core Length.

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(C-RD) Specifically as a function of fuel element material, the coefficients of preceding formula are as follows:

for graphite fuel elements -

$$A = 570$$

$$B = 8,225$$

$$C = 1,413$$

for 30 v/o UC_2 -ZrC-graphite composite
fuel elements -

$$A = 570$$

$$B = 10,323$$

$$C = 1,413$$

for all-carbide fuel elements -

$$A = 570$$

$$B = 15,963$$

$$C = 1,413$$

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RELATIVE SHIELD WEIGHT

Figure A-1.8-1. (CRD) Relative Shield Weight as a Function of Carbide Concentration in the Fuel (U)

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Figure A-1.8-2. Relative Shield Weight as a Function of Core Diameter

Figure A-1.8-3. Relative Shield Weight as a Function of Core Length

Figure A-1.8-4. Relative Shield Weight as a Function of Fuel Element Void Fraction

Figure A-1.8-5(U) Reactor Weight (Less Shield) As a Function of Core Length and Diameter for Graphite Fuel Element (U)

Figure A-1.8-6 (CRD) Reactor Weight (Less Shield) As a Function of Core Diameter and Length For a 30 v/o Composite Fuel Element (CRD)

Figure A-1.8-7 (U) Reactor Weight (Less Shield) As a Function of Core Diameter and Length for an All Carbide Fuel Element (U)

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A-1.9 (C-RD) PARAMETRIC EVALUATION AND SELECTION OF DESIGN ALTERNATES (U)

(C-RD) In this section, the information developed in the preceeding sections is combined and used to develop a family of parametric curves. The boundaries of the regions from which the design alternates are selected are established on these parametric curves. Selected alternates are then chosen from within the region defined by these boundaries. Two alternates, which appear in the final trade study, were not selected in this manner, although the parametric curves were useful for their selection. The first is the single channel carbide element (Alternate 6) because of the absence of coating, potential to zone load axially, extreme sensitivity to thermal stress, and "non-corrodable" performance, these unique characteristics make the carbide element the proper subject of a special parameter study which is, in fact, still in progress but is completed to the point where the dimensions for the alternate can be selected. The second was the "zig-zag" composite element (Alternate 7) which is only a modification of the outer boundary of the reference 19 hole element in order to more closely approximate a homogeneous equilateral triangular array. This modification would eliminate the slightly higher temperatures at the edge of the hexagonal element as discussed in Section A-1.4.1.1. The "zig-zag" modification to the hexagonal element does not significantly affect any of the parameters in this study, therefore, this alternate was not selected based on the figures in this section. The summary section, A-1.2, gives a good description of the selection process for the alternates.

A-1.9.1 (U) Thermal Stress Function (U)

(U) One of the objectives of the fuel element trade study was to explore the effects of changes in the hole pitch and reamed diameter on the thermal stress due to heat generation.

The tensile thermal stress caused by internal heat generation in a fuel element (FE) is developed in Figure A-1.9-1. The maximum tensile stress occurring at the lowest temperature region at any axial position is given by equation 1. This is only the stress component caused by the temperature gradient in a fuel element cross section.

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- (U) The temperature distribution in a fuel element depends on the temperature of the adjacent fuel elements and central element and the power generation rate and coolant flow of each channel in the element. Idealizing the fuel elements to an equilateral triangular lattice of coolant channels in a heat generating medium, the temperature distribution depends on pitch, S , and diameter, D . Temperature effects of adjacent elements and uncertainties in power and flow are not included. For application to the stress equation, it is assumed* that the average to minimum temperature difference is proportional to the temperature difference of this equation 2.
- (U) A thermal stress parameter, $Q\psi$, which is based on geometry and a reference nominal heating rate is used as a figure of merit, independent of fuel material properties. It is assumed that different extrusions, caused by varying S and D would have the same properties. This could be a poor assumption.
- (U) The maximum volumetric heat generation rate, Q , is given in equation 3 as a function of core power, fuel volume, and axial and radial power factors, f_A and f_R . The Q value can also be given in terms of core size and fuel volume fraction which is a function of pitch and diameter.
- (U) Combining these results of equations 1, 2, and 3 gives the maximum thermal stress due to internal heat generation (equation 4). Equation 4 clearly separates the factors contributing to the peak thermal stress in an element, hence, the avenues that should be followed to improve the element performance:
1. Basic fuel matrix properties, $\frac{E\alpha}{K}$, are discussed in Section 3.
 2. Increased core size to reduce $Q_{\text{core total}} / \text{Core Size}$ is considered in Section A-1.9.3.
 3. The radial power factor, f_R , is 1.1 and the axial power factor, f_A , is 1.4.

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* This assumption is slightly conservative, yielding diameters a maximum of 2% larger than the more exact calculation.

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4. The coolant channel geometry factors are the last terms in the equations. From the parametric study in the next section, candidate fuel element designs were selected.

A-1.9.2 (C-RD) Effect of Fuel Element Geometry Changes (U)

- (U) The two parameters which can be varied to improve fuel thermal stresses are hole pitch, S , and fuel reamed diameter, D . For the equilateral triangular array of coolant channels considered in this study, the fuel element void fraction V , is given by:

$$V = \frac{\pi D^2}{3.4641 S^2}$$

- (U) The dependent variables of interest in this study are strong functions of void fraction and a convenient set of working curves turn out to be void fraction plotted versus reamed diameter for constant hole pitch.

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- (U) The stress function, $Q\psi$ discussed in Section A-1.9.1 can be superimposed on this plot and also appear as lines with a positive slope.

- (C-RD) Using the parametric curves in Section A-1.4.2.1, the core pressure drop for constant coating thickness can be added and appear as lines with a negative slope crossing the constant hole pitch lines at $\sim$ right angles. This parametric curve is given as Figure A-1.9.2. On this figure, an obvious way to reduce the thermal stress function without increasing core ΔP is to move along the constant $\Delta P = 170$ psi line from the reference element. This would reduce both hole pitch and reamed diameter at the same time while increasing the void fraction. For example, an element with 0.150 inch hole pitch and 0.0925 inch reamed diameter would have 0.346 void fraction with a thermal stress function of about 0.049 Btu/second-inch. However, the lines on Figure A-1.9.2 do not tell the whole story since there are penalties involved in fuel loading, corrosion, and probably in material properties. Additionally, for a 19 hole element the across flats dimension would be reduced below the 0.692 inch minimum given by Section A-1.6.1.3. Since the figure is valid for both 19 hole element and 37 hole element, the across flats dimension need not be considered as a limit. It is worth noting, that a change in across flats dimension requires considerable redesign of other components and may involve a schedule

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delay. As will be discussed later, any significant change (by ~ 3 mils or so) in the reference design 0.173 inch hole pitch will require a change in across flats dimension. Finally, a limit may be reached in the minimum web thickness for different combinations of hole spacing and reamed diameter. As given by Section A-1.6.1.2, the minimum web thickness is 0.015 inches for both composite and graphite matrix materials. This minimum web thickness limit line appears far off the top of Figure A-1.9-2 and, therefore, is not a limiting parameter for the range of variables shown there.

(C-RD) This figure has been discussed in some detail as an introduction to the working curves which have added the fuel loading limit lines and relative reactivity loss lines. The relative corrosion reactivity loss was developed in Section A-1.4.4 as a function of fuel geometry including the effects of temperature changes given by Section A-4.2.2. The base case, for 1.0 relative reactivity loss is the reference element. Lines of constant reactivity loss are nearly horizontal with a slight positive slope. The fuel loadings required in the reference core size are functions of both geometry and materials as given by Section 5.0. The ZrC coating has a small positive coefficient of reactivity which has been neglected on the parametric curves in this section. As shown by Figure A-1.5-7, the maximum loading for a composite element with no coating reaches 700 mg/cc (the chosen limit from Section A-1.4.1.3) at a void fraction of 0.366. For the NbC coated (2.2 mils average thickness) graphite element, the maximum uranium loading reaches the chosen limit (630 mg/cc) at the reference element geometry.

(C-RD) Figure A-1.9-3 is a "working curve" version of Figure A-1.9-2 with lines of constant relative reactivity loss and constant loading added. Two thicknesses of NbC coating (2.2 mils and 4.4 mils) are represented for the graphite element. The composite element loading (neglecting ZrC coating reactivity) is given as 650, 700, and 750 mg/cc.

- (U) Using this figure, the effect in fuel geometry can easily be determined. For a chosen combination of hole pitch and diameter the required across flats dimension is given in Figure A-1.4.9-4 for both 37 and 19 hole elements.
- (C-RD) Two of the selected alternates are marked on this figure along with the reference element. Alternate 4, the 37 hole element, was chosen at a hole pitch of 0.1309 inches because this results in a fuel element across flats of 0.799, the same as has been used in previous studies. This element, with zirconium coating violates both the 700 mg/cc loading line and the 170 psi pressure drop line slightly, but both limits are felt to be conservative. For this chosen across flats dimension and hole pitch, Figure A-1.9-5 shows in more detail the effect of changes in reamed diameter on void fraction, stress function, web thickness and pressure drop. Alternate 8, a high void element with 0.173 hole pitch and 0.753 across flats shows a relatively small improvement in $Q\psi$ but a 30 psi lower core pressure drop. Therefore, further improvements in $Q\psi$ can be gained by making the core longer. Figure A-1.9-6 shows the effects of increasing core length on stress, pressure drop, reactor weight, and shield weight.
- (U) The chosen alternate is 62 inches long, which allows some margin for thicker bore coatings and/or a reduction in the reamed diameter and void fraction.
- (U) Additional parametric working curves are attached as Figures A-1.9-7 through A-1.9-9 and are self-explanatory.

A-1.9.3 (C-RD) Effect of Increasing Core Size (U)

(C-RD) The lateral support system limits any significant increase in core diameter beyond the reference size according to Section A-1.6.3. In the preceeding section, it was shown that a high void (0.108 inch reamed diameter) element will permit the core length to be increased up to ~72 inches without violating the core ΔP limit of 170 psi. Since this could be accomplished without major redesign of the reflector and lateral support system, it would be the preferred approach to take. However, if a satisfactory element cannot be manufactured in that configuration (or if some other constraint limits the fuel element to basically the reference geometry) then an increase in core diameter and length is the only method to reduce stresses due to internal heating.

(U) The maximum volumetric heat generation is inversely proportional to the core volume, assuming that the radial and axial peaking factors, f_R and f_A , do not change. For the reference R-1 element, Figure A-1.9-10 shows the thermal stress function versus fueled radius at constant core length. Lines of constant core pressure drop are given based on the parametric curves in Section A-1.4.3. An alternate can be selected at any point along the constant $\Delta P = 170$ line depending on the value of $Q\psi$ that is required for the fuel.

(C-RD) Alternate 9 was selected at a fueled radius of 20.3 inches and length of 76 inches because the $Q\psi$ resulting is about the same as given for the 37 hole element (Alternate 4). Once the decision is made to make the core diameter bigger, it is, of course, an advantage to use the reference element. There is, however, a very substantial reduction in fuel loadings which would allow an increase in void fraction. An infinite combination of core sizes and element geometries can result, but based on the preceeding discussion, the only advantage of a larger core diameter over alternate 8 is that it permits us to retain the reference fuel geometry, while reducing the thermal stress function.

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$$(1) \text{ THERMAL STRESS} = E \alpha (\bar{T} - T_{\text{MINIMUM}})$$

$$(2) T_{\text{MAX}} - T_{\text{SURF}} = \frac{Q}{K} \Psi \quad \Psi = S^2 f\left(\frac{S}{D}\right)$$

$$(3) Q_{\text{MAX}} = \frac{Q_{\text{CORE TOTAL}} f_R f_A}{\text{CORE FUEL VOLUME}} = \frac{Q_{\text{CORE TOTAL}} f_R f_A}{\text{CORE SIZE} \times \text{FUEL FRACTION}}$$

$$(4) \text{ MAX. THERMAL STRESS, } \left(\frac{E\alpha}{K}\right) \left(\frac{Q_{\text{CORE TOTAL}}}{\text{CORE SIZE}}\right) f_R f_A S^2 f\left(\frac{S}{D}\right)$$

E = MATRIX MODULUS OF ELASTICITY*

α = MATRIX CTE*

K = MATRIX THERMAL CONDUCTIVITY*

Q = VOLUMETRIC HEAT GENERATION RATE

$Q_{\text{CORE TOTAL}}$ = TOTAL POWER GENERATED IN CORE FUEL

f_R = RADIAL PEAK TO AVERAGE HEAT GENERATION RATE

f_A = AXIAL PEAK TO AVERAGE HEAT GENERATION RATE

$f(S/D)$ = FUNCTION OF (S/D)

$\frac{*E\alpha}{K}$ VARIES WITH TEMPERATURE LEVEL, BUT IS TREATED AS A
CONSTANT FOR PURPOSES OF COMPARISON

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Figure A-1. 9-1. Thermal Stress Due to Steady State Heat Generation

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1.9-2 (C-RD) FUEL ELEMENT VOID FRACTION VERSUS REAMED DIAMETER (U)

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WORKING CURVE

FUEL ELEMENT VOID FRACTION VERSUS
REARED DIAMETER FOR CONSTANT
HOLE PITCH

EQUILATERAL TRIANGULAR ARRAY

REFERENCE CORE SIZE

CORE AP BASED ON 2.18 MILS AVE. CONTING THA'S.

UNLOADING FOR
30% COMPOSITE
NEGLECTING SMALL
+ REACTIVITY FOR 20C CONT

Figure A-1.9-3. (CRD) Fuel Element Void Fraction Versus Reamed Diameter for Constant Hole Pitch (U)

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Figure A-1.9-4. (CRD) Fuel Element Across Flats Versus Hole Spacing for Constant Reamed Diameter (U)

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Hole Pitch = 0.1309 Inch

Across Flats = 0.799 Inch

Average Coating Thickness = 2.18 Mils

FIGURE A-1.9-5 (C-RD) PARAMETRIC CURVES FOR 37 HOLE FUEL ELEMENT GEOMETRY (U)

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* Including Shield

FIGURE A-1.9-6 (C-RD) DESIGN PARAMETERS VERSUS CORE LENGTH, ALTERNATE 8 (U)

Figure A-1.9-7. (CRD) Stress Function Versus Element Across Flats - 19 Hole (U)

Figure A-1.9-8. (CRD) Stress Function Versus Across Flats - 37 Hole (U)

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Figure A-1.9-9. (CRD) Fuel Reamed Diameter - Inch (U)

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FIGURE A-1.9-10 (C-RD) STRESS FUNCTION VERSUS CORE FUELED RADIUS FOR CONSTANT CORE LENGTH (U)

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A-1.9.4 (C-RD) Parametric Study for UC -ZrC Carbide Elements (U)

(C-RD) A preliminary parametric study for the single channel carbide elements was carried out separately from the study for graphite and 30 v/o composite elements. Because of the absence of coating, the single channel geometry and significantly different material properties, the scale of geometry and some of the ground rules for this study were different. Many of the material properties for the carbide fuel are at best only tentative estimates at this time. However, the thermal study has shown the significance of variables such as geometry and conductivity for the properties selected. With the exception of the thermal conductivity, these property estimates are as shown in Figure A-1.3-7 of Section A-1.3. The thermal conductivity has been adjusted to account for the reduction associated with 30 percent porosity and 10 percent UC (versus the theoretical 100 percent density as shown in Figure A-1.3-7). This provides a probable lowest k value for extruded fuel elements and a limiting case for thermal stress.

(C-RD) In view of the above, the following must be remembered in interpreting the results of this study: 1) material property data are very preliminary, i.e., effect of improved modulus at high temperatures as well as lowering ultimate strength may be conservatively considered, 2) variation in material properties (fuel composition) and loading with axial station would help reduce stress levels. (This is being studied.) Therefore, constraints shown are initial scoping limits to provide some indication of the design range and limits which must be considered and are somewhat pessimistic. The following is a brief discussion of this carbide element study.

(CRD) A-1.9.4.1 Assumptions or Fixed Values (U)

1. Nominal FE exit gas temperature = 4300°R
2. 7% higher than NERVA nominal power
3. Chamber pressure = 450 psia
4. The carbide elements were assumed to consist of short, single-channel tubular sections. The flow through the interstices between tubes was neglected such that the hydraulic calculations are valid for both tubes and hexes. The

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effect of the interstitial flow was then imposed to evaluate the tube geometry. The maximum temperatures would be higher for a hex than for a tube, while the volumetric heating rates would be slightly lower, using the same pitch and hole diameter.

5. The maximum nominal radial power factor was assumed $= 1.0$ because of the capability for fine gradations in fuel loading and, therefore, excellent radial power flattening.

6. Because of item 5, no orificing is used. The inlet pressure loss consists of a sharp edged contraction from a plenum to the channel hydraulic diameter.

7. The axial power distribution is conservatively assumed to be the same as the reference with the axial maximum/average $q''' = 1.4$. This is because axial zone loading capability may be limited due to the finite length of these segments, maximum temperature limits (desire to keep the volumetric heating lower than average at the aft end), and the absence of end reflectors.

8. The hydraulic diameter is assumed equal to the reamed (or extruded) diameter because of the smooth bores and absence of coating.

9. Thermal conductivity for ZrC with 10 percent UC and 30 percent porosity.

(C-RD) Figure A-1.9-11 shows the constraints evaluated for thermal stress, fuel loading, pressure drop and minimum web thickness considerations for a hexagonal single channel carbide fuel element. The stress margin limits were established in Section A-1.3, based on the best available properties assuming a 30% porosity of fuel material. The carbide element shown for Alternate 6 is above the preliminary stress limit. However, consideration of the number of elements required for an R-1 core (indicated on the figure), the preliminary nature of the carbide properties, and the fact that LASL has investigated elements close to this configuration, justified selection of this alternate.

(C-RD) Thermal conductivity is a significant property in determining thermal stress. The effect of this property on fuel element temperature difference (a measure of thermal stress) is given in Figure A-1.9-12 for two carbide porosities. For comparison, the reference graphite fuel and 30 v/o composite values are given for a 0.170 inch fuel channel pitch.

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- (U) Figure A-1.9-13 gives comparative constant estimates for both hexagonal and tubular single channel elements.
- (U) The assumption of adiabatic conditions at the fuel element outer surface used for the tube geometry is not clearly acceptable. However, the study did show less than a 6 percent local effect on temperature profiles for acceptable values of element pitch (< 0.130 inches). Total interstitial flow heat pickup for this pitch was less than 2.5 percent of the total core heat generation. Core pressure drops associated with acceptable fuel geometries and void fractions ($\sim 40-50$ percent) were lower than present NERVA reference levels.

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FIGURE A-1.9-11 (C-RD) SINGLE CHANNEL HEXAGONAL CARBIDE ELEMENT SELECTION (U)

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~~GROUP 1, EXCLUDED FROM AUTOMATIC DOWNGRADING AND DECLASSIFICATION~~

FIGURE A-1.9-12 (C-RD) COMPARISON OF MAXIMUM FUEL WEB TEMPERATURE DIFFERENCE WITH VARIATION IN FUEL CONDUCTIVITY (U)

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FIGURE A-1.9-13 (C-RD) DESIGN CONSTRAINT ESTIMATES CARBIDE SINGLE CHANNEL
REFERENCE NERVA CORE SIZE WITH 10 AXIAL FUEL SEGMENTS
ZrC + 10% UC with 30% POROSITY (U)

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APPENDIX A-2

LAYOUTS OF CONCEPTS TRADED
TRADE STUDY FOR NERVA FUEL ELEMENT
DESIGN CONCEPTS

TRADE STUDY NO. 759

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A-2 LAYOUTS OF CONCEPTS TRADED (U)

(U) Following are the layouts of concepts traded.

FIGURE NUMBER	ALTERNATE NUMBER	DRAWING NUMBER AND TITLE
A-2.1.1-1	1	WANL Drawing Number 947C812 Fuel, Intermediate Expansion Graphite Element
A-2.1.1-2	2	WANL Drawing Number 947C813 Fuel, High Expansion Graphite Element
A-2.1.1-3	3	WANL Drawing Number 947C810 Fuel, Hybrid Element
A-2.1.1-4	4	WANL Drawing Number 947C816 Fuel, Modified Hybrid Element
A-2.1.1-5	5	WANL Drawing Number 947C809 Fuel, All Composite Element
A-2.1.1-6	6	WANL Drawing Number 947C817 Fuel, Carbide Element
A-2.1.1-7	7	WANL Drawing Number 947C807 Fuel, Zigzag Composite Element
A-2.1.1-8	8	WANL Drawing Number 947C814 Fuel, High Void, Composite, Long Element
A-2.1.1-9	9	WANL Drawing Number 947C815 Fuel, High Expansion Graphite, Long Element

(U) These layouts are enclosed following this list.

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A-2-3

947C813

Figure A-1.1-2

REVISIONS				
ZONE	LTR	DESCRIPTION	DATE	APPROVED

SCALE 4/1

LAYOUT

QTY	SYM	ITEM NO.	NOMENCLATURE OR DESCRIPTION	CODE IDENT NO	PART OR IDENTIFYING NO	MATERIAL	SPECIFICATION	ZONE
PARTS LIST								
UNLESS OTHERWISE SPECIFIED				CONTRACT NO. NP-1		Westinghouse Astronuclear Laboratory Pittsburgh, Pennsylvania		
INTERPRET DWG PER MIL-D-1000 DIM IN INCHES BASED ON 68°F				DFTN J. WOJCIK		DATE OF DWG 12-1-69		
TOLERANCES - DO NOT SCALE				CHECKER		REVIEW E. O. A.		
ZPL DEC SPL DEC ANGLES				APPROVED		RFS		
± ± ±				AA		STRENGTH		
- 1 -				MAYL		DFTN		
NEXT ASSY		USED ON		INFORMATION CATEGORY		DWG SIZE		
M		APPLICATION		SPECIFICATION		TYPE		C
DWG CODE NO.		4		10-763939-A-10 AAN		14683		947C810
SCALE 1/1		WT ACT		WT CALC		SHEET 1		OF 1

Figure A-2.1.1-3

A-2-5

LAYOUT

947C816

QTY REQD	SYM	ITEM NO.	NOMENCLATURE OR DESCRIPTION	CODE IDENT NO	PART OR IDENTIFYING NO	MATERIAL	SPECIFICATION	ZONE
PARTS LIST								
UNLESS OTHERWISE SPECIFIED				CONTRACT NO. NP-1		Westinghouse Astronuclear Laboratory Pittsburgh, Pennsylvania		
INTERPRET DWG PER MIL-D-1000 DWG IN INCHES BASED ON 88°F				DATE OF DWG 12-29-69		ELEMNT, FUEL, MODIFIED HYBRID, LAYOUT OF (U)		
TOLERANCES - DO NOT SCALE				CHECKER L. RAMSEY		RELIAB & C. A.		
ZPL DEC SPL DEC ANGLES				DESIGN 12-4-70		MFG		
± ± ±				ANALYSIS		STRESS		
- 1 0				MAYL		DRY SUPV 12-16-70		
INFORMATION CATEGORY				DWG SIZE C		CODE IDENT NO. 14683		DWG NO. 947C816
NEXT ASSY USED ON				SCALE 1/1		WT ACT.		WT CALC
W APPLICATION SPECIFICATION TYPE				SHEET 1		OF 1		
DWG CODE NO. 4				AUTHOR		DATE		

Figure 2.1.1-4

A-2-6

REVISIONS			
ZONE	LTR	DESCRIPTION	DATE

ϕ A .012 DIA
 SCALE 4/1

LAYOUT

947C809

QTY	SYN	ITEM NO.	NOMENCLATURE OR DESCRIPTION	CODE IDENT NO.	PART OR IDENTIFYING NO.	MATERIAL	SPECIFICATION	ZONE
PARTS LIST								
UNLESS OTHERWISE SPECIFIED				CONTRACT NO. NP-1		Westinghouse Astronuclear Laboratory Pittsburgh, Pennsylvania		
INTERPRET DWG PER MIL-D-1000 DIM IN INCHES BASED ON 80°F				DATE OF DWG 12-2-69				
TOLERANCES — DO NOT SCALE				CHECKER J. WOJCIK		RELIN & C. A.		
2PL DEC 3PL DEC ANGLES				DESIGN ANALYST		BFG STERN		
± — ± — ± —				BAYL		DPT. ENR		
NEXT ASSY USED ON				INFORMATION CATEGORY		DWG NO. 947C809		
M APPLICATION SPECIFICATION TYPE				14/69		CODE IDENT NO. 14683		
DWG CODE NO. 4 D-763957-A-10 AM				AUTHORISED SIGNATURE DATE		SCALE 1/1 WT ACT. — WT CALC — SHEET 1 OF 1		

Figure A-2.1.1-5

A-2-7

REVISIONS				
ZONE	LTR	DESCRIPTION	DATE	APPROVED

10 @ 5.20 - 52 00

.120 ± .001 DIA THRU
A .012 DIA

.170 BASIC HEX
A

5.20

MATERIAL ZrC UC
COATING NONE

LAYOUT

QTY	SYN	ITEM	NOMENCLATURE OR DESCRIPTION	CODE IDENT NO	PART OR IDENTIFYING NO.	MATERIAL	SPECIFICATION	ZONE
REQD								

UNLESS OTHERWISE SPECIFIED				CONTRACT NO. NP-1	
INTERPRET DWG PER MIL-D-1000 DIM IN INCHES BASED ON NET				DATE OF DWG 12-29-69	DATE OF REV 12-29-69
TOLERANCES - DO NOT SCALE				CHECKER WELLS & O A	DESIGN WPG
2PL DEC 3PL DEC ANGLES				ANALYSIS STRENGTH	MATL WPG
* * *				WPG	
- 1 1				WPG	
NEXT ASSY USED ON				INFORMATION CAT	
W APPLICATION SPECIFICATION TYPE				1/1/70	
DWG CODE NO 4				AUTHOR	

Westinghouse Astronuclear Laboratory Pittsburgh, Pennsylvania		
ELEMENT FUEL, CARBIDE, LAYOUT OF (U)		
DWG SIZE C	CODE IDENT NO. 14683	DWG NO. 947C817
SCALE 4/1	WT ACT	WT CALC
SHEET 1	OF 1	

Figure A-2.1.1-6

A-2-9

REVISIONS			
ZONE	LTR	DESCRIPTION	DATE
		APPROVED

LAYOUT

947C814

QTY	SYN	ITEM NO.	NOMENCLATURE OR DESCRIPTION	CODE IDENT NO.	PART OR IDENTIFYING NO.	MATERIAL	SPECIFICATION	ZONE
REQD								
PARTS LIST								
UNLESS OTHERWISE SPECIFIED				CONTRACT NO. NP-1		Westinghouse Astronuclear Laboratory Pittsburgh, Pennsylvania		
INTERPRET DWGS PER MIL-D-1000 DIM IN INCHES BASED ON 68°F				DATE OF DWG 12-29-69		HIGH VOID COMPOSITE ELEMENT, LONG, FUEL LAYOUT OF, (U)		
TOLERANCES - DO NOT SCALE				CHECKER J. WOJCIK				
ZPL DEC 3PL DEC ANGLES				DESIGNER [Signature] 1/79				
± — ± — ± —				ANALYSIS [Signature]				
— — — — —				MATERIAL [Signature]		DWG SIZE C		
NEXT ASSY USED ON				INFORMATION [Signature]		CODE IDENT NO. 14683		
APPLICATION SPECIFICATION TYPE				DATE		DWG NO. 947C814		
DWG CODE NO. 4				0-73955-A-10A		SCALE 1/1		
						WT ACT. — WT CALC — SHEET 1 OF 1		

Figure A-2.1.1-8

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APPENDIX A-3

NUCLEAR DESIGN SUPPORTING DATA
TRADE STUDY FOR NERVA FUEL ELEMENT
DESIGN CONCEPTS
TRADE STUDY NO. 759

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(CRD) A-3.0 INTRODUCTION (U)

(CRD) This Appendix describes the nuclear design data in support of the Fuel Trade Study. These data are employed in an evaluation of the nine alternate fuel designs described in Section 2.0 insofar as they satisfy the following design requirements:

(U) 1. Requirement Nuclear Criticality--"The fuel element as a part of the NSS shall be capable of sustaining a controlled nuclear heat generation through 60 operating cycles during 10 hours operation with fuel loading limitations of a) 630 mg/cc graphite, b) 700 mg/cc composite and c) about 2000 mg/cc carbide."

(U) The nominal loadings and critical uranium mass are estimated for each alternate fuel design based on both a ± 1.50 and a ± 3.00 shim capability. A ± 3.00 shim capability requirement is indicated from development with the NRX reactor series (comparison of predicted and observed cold critical drum bank angles), provision for design changes and from expected manufacturing variations in fuel element material inventories. Statistical analysis has indicated a small probability (0.35 percent) of reactivity variation exceeding the ± 3.00 capability. The development of this requirement is documented in the Cluster, Stem and Hot End Hardware Trade Study.

(U) The best estimates of reactivity worth coefficients for major core materials are given for each alternate fuel design. Pertinent experimental worth measurements are discussed where applicable and areas of weakness in analytical technique requiring further consideration are noted.

(U) Critical uranium loadings are estimated parametrically in shim content, reamed channel diameter, clad material and/or clad mass. These design variations among others are treated in more detail in Appendix A-1.

(U) 2. Requirement Reactivity Shimming--"The core reactivity change available through shimming shall be ± 3 ."

(U) The uranium loadings specified for each alternate design under Requirement (Nuclear Criticality) are based on this ± 3.00 shim capability requirement. The effect of a lower design shim content is discussed under Requirement (Nuclear Criticality). The Cluster, Stem and Hot

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End Hardware Trade Study discusses shim design and shim material for this requirement.

(U) 3. Requirement Reactivity Loss--"The total reactivity loss of the core, after the performance requirements have been met, shall not exceed \$3 (Design Goal)."

(U) The axial carbon reactivity worth profile is given for each alternate fuel design. Thermal Design Supporting Data (Appendix A-4) estimates the operating reactivity losses via corrosion using these profiles in conjunction with expected carbon weight losses.

(CRD) 4. Requirement Weight--"The weight of the fuel shall be minimized within the limits imposed by the functional requirements. The Weight of the Reference Design (939J685, Rev. E with Graphite Fuel) fuel envelope is 3650 pounds."

(U) The total fueled envelope weight is estimated for each alternate fuel design.

(U) 5. Requirement Nuclear Loading--"The fueled elements shall have varied fuel loadings as required to achieve a peak-to-average reactor radial power factor from 0.6 to 1.1. The loading range for the Reference Design (939J685, Rev. E High Expansion Graphite Fuel) is from 268 to 630 mg/cc."

(U) The radial fuel loading density profile was calculated directly for the Alternate 2 (Standard Geometry-Hith CTE) Design under this requirement and ranges from 268 mg/cc to 630 mg/cc. The remaining alternate design loadings are based on this same loading profile and may require adjustment with further analysis in satisfaction of this fission density requirement. A qualitative evaluation in this regard is made for each alternate design.

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(CRD) A-3.1 Design Concept 1: Standard Geometry--Intermediate Expansion Graphite
(Drawing 947C812) (U)

(CRD) A-3.1.1 Requirement Nuclear Criticality (U)

(U) The peak uranium loading density of 621 mg / cc with a ± 3.00 shim content required for this fuel design does not exceed the 630 mg./cc. limitation for graphite fuel as discussed in Appendix A-1.

(CRD) The nominal design parameters pertinent to the nuclear characteristics of this design are,

1. 1878 fueled elements
2. 349 unfueled graphite centrals
3. 0.753-inch across flats element dimension
4. 19 flow channels per fueled element
5. 0.102-inch reamed channel diameter
6. NbC clad at 75 gms Nb per fueled element plus ZrC clad at 15 gm ZrC per fueled element
7. The fueled graphite density (GLFI flour) versus uranium loading is shown in Figure 1.
8. 52-inch fuel element length

(CRD) This design has an additional core graphite content of 12.6 kg relative to the Alternate 2 (Standard Geometry-High CTE) Design due to an increased fueled graphite density. Critical loadings were estimated by a reactivity adjustment of the calculated Alternate 2 uranium inventory to accommodate this carbon mass change. This adjustment assumes that the Alternate 2 radial loading density profile results in a satisfactory fission power density distribution when applied to this design. On this basis, the nominal critical loadings corresponding to ± 3.00 shim content are:

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maximum	621 mg/cc
minimum	264 mg/cc
average	507 mg/cc
Total U mass	272k g

(U) The Alternate 2 (Standard Geometry - High CTE) Design core material worths coefficients as measured in the PAX/R-1 assembly are applicable:

Uranium worth	+12.1¢/kg
Carbon worth	+ 3.9¢/kg
Niobium worth	- 4.8¢/kg

The same high degree of confidence, with a $\pm$ \$3.00 shim capability, in predicting the nuclear requirements of the Alternate 2 (Standard Geometry-High CTE) Design should apply to this design.

(U) Figure 2 shows the dependence of peak uranium loading density on shim content, reamed channel diameter, clad material and clad mass. The results indicate that the loading decreases by approximately 4.7 percent with a reduction in shim content from $\pm$ \$3.00 to $\pm$ \$1.50. In particular, the nominal critical loadings at $\pm$ \$1.50 shim content are,

maximum	592 mg/cc
minimum	252 mg/cc
average	483 mg/cc
Total U mass	260 kg.

(U) A-3.1.2 Requirement Reactivity Shimming

The $\pm$ \$3.00 shim capability requirement is satisfied in this design with a peak uranium loading density of 621 mgm/cc. Various shim materials have been investigated for the Alternate 2 (Standard Geometry-High CTE) Design and the results are discussed in the Cluster, Stem and Hot End Hardware Trade Study. These results are applicable to this design due to the similarity in neutron spectrum.

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(U) A-3.1.3 Requirement Reactivity Loss

The carbon axial worth profile of the Alternate 2 (Standard Geometry-High CTE) Design is applicable (see Figure 5). Thermal Design Supporting Data (Appendix A-4) estimates the operating reactivity loss via corrosion for this design by employing this profile in conjunction with an expected carbon mass loss.

(CRD) A-3.1.4 Requirement Weight (U)

(CRD) The total core weight (fueled envelope) consistent with the required loadings of this design is approximately 3660 pounds.

(U) A-3.1.5 Requirement Nuclear Loading

The radial uranium loading profile of the Alternate 2 (Standard Geometry-High CTE) Design is applicable due to the close similarity of these two designs. This profile should satisfy the fission density constraint imposed by this requirement.

(CRD) A-3.2 Design Concept 2: Standard Geometry-High Expansion Graphite
(Drawing 947C813) (U)

(U) A-3.2.1 Requirement Nuclear Criticality

The peak uranium loading density of 630 mg/cc with a ± 3.00 shim content required for this design does not exceed the 630 mg/cc limitation for graphite fuel as discussed in Appendix A-1.

The nominal design parameters of the Alternate 2 (Standard Geometry-Intermediate Expansion) Design are applicable with the exception of the fueled graphite density (AXM flour) versus loading shown in Figure 3. The following nominal critical loadings (based on a ± 3.00 shim capability) were determined for this design using a two-dimensional, 16-group neutron diffusion model (EXTERMINATOR).

maximum	630 mg/cc
minimum	268 mg/cc
average	515 mg/cc
Total U mass	277 kg

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(U) Core material worths for this design were experimentally determined by measurements in the PAX/R-1 critical assembly:

Uranium worth	+12.1¢/kg
Carbon worth	+ 3.9¢/kg
Niobium worth	- 4.8¢/kg

This experimental data together with the high degree of success (with $\pm \$3.00$ shim capability) in predicting the previous NRX series reactors indicate a high degree of confidence in the predicted nuclear requirements of this design.

Figure 4 shows the dependence of peak uranium loading on shim content, reamed channel diameter, clad material and clad mass. In general the variations are the same as those exhibited in the Alternate 1 (Standard Geometry-Intermediate Expansion) Design. The results indicate that the loading decreases by approximately 4.6 percent with a reduction in shim content from $\pm \$3.00$ to $\pm \$1.50$. In particular, the nominal critical loadings at $\pm \$1.50$ shim content are,

maximum	601 mg / cc
minimum	256 mg / cc
average	492 mg / cc
Total U mass	264 mg / cc

(U) A-3.2.2 Requirement Reactivity Shimming

The $\pm \$3.00$ shim capability requirement is satisfied in this design with a peak loading density of 630 mg / cc.

(U) A-3.2.3 Requirement Reactivity Loss

The carbon axial reactivity worth profile is shown in Figure 5. This profile was calculated using one-dimensional transport perturbation theory (PERT) normalized to the experimentally determined uniform worth of +3.9¢/kg. Thermal Design Supporting Data (Appendix A-4) estimates the reactivity loss via corrosion for this fuel design using this worth profile in conjunction with an expected carbon mass loss.

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(CRD) A-3.2.4 Requirement Weight (U)

(CRD) The total core weight (fueled envelope) consistent with the required loadings of this design is approximately 3640 pounds.

(U) A-3.2.5 Requirement Nuclear Loading

The radial loading profile for this design was calculated directly using a two-dimensional, 16-group neutron diffusion model (EXTERMINATOR) with the requirement that the fission density in any fuel channel does not exceed 110 percent or is less than 60 percent of the average fission density.

(CRD) A-3.3 Design Concept 3: Standard Geometry-Hybrid (Drawing 947C810) (U)

(CRD) A-3.3.1 Requirement Nuclear Criticality (U)

(U) The peak loading density of 520 mg / cc with a +\$3.00 shim content required for this design does not exceed the 630 mg / cc and 700 mg / cc constraints for graphite and composite fuels, respectively, as discussed in Appendix A-1.

(CRD) The nominal design parameters of the Alternate 1 (Standard Geometry-Intermediate CTE) Design are applicable with the exception of:

1. All ZrC clad at 75 gm ZrC/element
2. Element averaged fueled graphite density shown versus uranium loading in Figure 6.

(CRD) The critical uranium loadings at +\$3.00 shim content of this design were calculated as follows:

1. The average loading for the high CTE fuel with 75 gm ZrC/element clad is 411 mg / cc.
2. The average loading for the composite fuel with 75 gm ZrC/element clad is 474 mg / cc (see Alternate 5 (Standard Geometry-Composite) Design results).
3. These loadings would apply to the hybrid element with an axial power density discontinuity at the high CTE/composite interface.

4. A single hybrid element loading (which eliminates this axial discontinuity) was determined by a reactivity adjustment of these loadings.

On this basis, the nominal hybrid loadings (with ± 3.00 shim content) are,

maximum	520 mg /cc
minimum	221 mg /cc
average	425 mg /cc
Total U mass	228 kg

(U) The effective reactivity worth coefficients were determined by sine weighting the Alternate 2 (Standard Geometry-High CTE) and Alternate 5 (Standard Geometry-Composite) Design worths. The sine weighted coefficients are:

1. In the high CTE portion of the element

Uranium worth	+13.6¢/kg
Carbon worth	+ 4.4¢/kg
Niobium worth	- 5.4¢/kg
2. In the composite tip

Uranium worth	+ 8.6¢/kg
Carbon worth	+ 3.2¢/kg
Zirconium worth	+ 0.004¢/kg
3. The total element averaged coefficients are

Uranium worth	+11.9¢/kg
Carbon worth	+ 4.0¢/kg
Zirconium worth	+ 0.004¢/kg

The same comments regarding the uncertainty in zirconium worth and the recommended experimental program discussed for the Alternate 5 (Standard Geometry-Composite) Design is applicable.

(U) Figure 7 shows the dependence of maximum uranium loading on shim content, reamed channel diameter, clad material and clad mass. The results indicate that the

loading decreases by approximately 5.6 percent in reducing the shim capability from ± 3.00 to ± 1.50 . In particular, the nominal loadings at ± 1.50 shim content are,

maximum	491 mg / cc
minimum	209 mg / cc
average	401 mg / cc
Total U mass	215 kg

Figure 8 shows the nominal peak loading variation with length of the high CTE portion for this design.

(U) A-3.3.2 Requirement Reactivity Shimming

The ± 3.00 shim capability requirement is satisfied in this design with a peak loading density of 520 mg / cc.

(U) A-3.3.3 Requirement Reactivity Loss

The carbon axial reactivity worth profile is a combination of the Alternate 2 (Standard Geometry-High CTE) and 5 (Standard Geometry-Composite) Design worth profiles as shown in Figure 9. Thermal Design Supporting Data (Appendix A-4) estimates the reactivity loss via corrosion for this fuel design using this worth profile in conjunction with an expected carbon mass loss.

(CRD) A-3.3.4 Requirement Weight (U)

(CRD) The total core weight (fueled envelope) consistent with the required loadings of this design is approximately 4125 pounds.

(U) A-3.3.5 Requirement Nuclear Loading

The critical loadings determined for this design under Requirement (Nuclear Criticality) are calculated using the radial loading profile as calculated for the Alternate 2 (Standard Geometry-High CTE) Design. This profile may not satisfy the fission density requirement in the composite portion of this fuel design due to a slightly faster neutron spectrum in this region and may need to be revised with a subsequent increase in peak loading density. This increase should be small and should not compromise the peak loading density requirement above.

(CRD) A-3.4 Design Concept 4: 37-Hole Hybrid (Drawing 947C816) (U)

(CRD) A-3.4.1 Requirement Nuclear Criticality (U)

(CRD) The peak uranium loading density of 618 mg./cc with +\$3.00 shim content required for this design does not exceed the 630 mg./cc and 700 mg./cc constraints for graphite and composite fuels, respectively, as discussed in Appendix A-1. The nominal design parameters pertinent to the nuclear characteristics of this design are,

1. 1650 fueled elements
2. 319 unfueled centrals
3. 0.799-inch across flats element dimension
4. 37 flow channels per fueled element
5. 0.084-inch reamed channel diameter
6. ZrC clad at 120 gm ZrC./fueled element
7. The fueled graphite density versus loading of the Alternate 3 (Standard Geometry-Hybrid) Design is applicable (Figure 6).
8. 52-inch fuel element length

(CRD) In addition to the fuel element void fraction and clad mass, this design differs from the Alternate 3 (Standard Geometry-Hybrid) Design by,

1. A 17 kg gain in graphite mass in the central unfueled elements due to an increase in hex size with the same stem volume.
2. A 6 kg gain in ZrC mass in the composite portion of the central unfueled elements.

(U) The following critical loadings were estimated (for a +\$3.00 shim content) by a reactivity balance with the same assumptions outlined for the Alternate 3 (Standard Geometry-Hybrid) Design:

maximum	618 mg / cc
minimum	262 mg / cc
average	505 mg / cc
Total U mass	247 kg

(CRD) The material worths for this design differ from those of the Alternate 3 (Standard Geometry-Hybrid) Design due to an increased core void fraction. The fuel element void is about 37 percent as compared to the standard hybrid void of 31.4 percent. The reactivity coefficients for this case are:

Uranium worth	11.8¢/kg
Carbon worth	4.2¢/kg
Zirconium worth	+ 0.040¢/kg

The same comments regarding the uncertainty in zirconium worth and the recommended experimental program discussed for the Alternate 5 (Standard Geometry-Composite) Design is applicable.

(CRD) Figure 10 shows the dependence of maximum uranium loading on shim content, reamed channel diameter, clad material and clad mass. The loading is approximately 28 percent greater in this design if the high CTE portion of the element is NbC clad at 80 gm/element. The nominal loadings at ±\$1.50 shim content are,

maximum	582 mg / cc
minimum	248 mg / cc
average	475 mg / cc
Total U mass	232 kg

(U) A-3.4.2 Requirement Reactivity Shimming

The ±\$3.00 shim capability requirement is satisfied in this design with a peak loading density of 618 mg / cc.

(U) A-3.4.3 Requirement Reactivity Loss

The carbon axial reactivity worth profile of this design is shown in Figure 11. This profile was obtained by renormalizing the Alternate 3 (Standard Geometry-Hybrid) Design profile to a uniform carbon worth of 4.2¢/kg. Thermal Design Supporting Data (Appendix A-4) estimates the reactivity loss via corrosion using this worth profile in conjunction with an expected carbon mass loss.

(CRD) A-3.4.4 Requirement Weight (U)

(CRD) The total core weight (fueled envelope) consistent with the required loadings of this design is approximately 4060 pounds.

(U) A-3.4.5 Requirement Nuclear Loading

The critical loadings determined for this design under Requirement (Nuclear Criticality) are calculated using the radial loading profile as calculated for the Alternate 2 (Standard Geometry-High CTE) Design. The higher void fraction of this design indicates a faster neutron spectrum which may violate the fission density constraints of this requirement. Further analysis would be required to ascertain whether a modified profile which accommodates this requirement is consistent with the loading density requirement above.

(CRD) A-3.5 Design Concept 5: Standard Geometry-Composite (Drawing 947C809) (U)

(CRD) A-3.5.1 Requirement Nuclear Criticality (U)

(U) The peak uranium loading density of 579 mg / cc with a ±3.00 shim content required for this design does not exceed the 700 mg / cc constraint for composite fuel as discussed in Appendix A-1.

(CRD) The nominal design parameters of the Alternate 1 (Standard Geometry-Intermediate CTE) Design are applicable with the exception of,

1. ZrC clad at 75 gm ZrC/element rather than NbC.
2. Fueled graphite density versus uranium loading shown in Figure 12.

(CRD) The significant difference between this design and the Alternate 2 (Standard Geometry-High CTE) Design is in the addition of approximately 1100 kg of zirconium and a 160 kg loss in graphite mass. The following critical loadings (with ±3.00 shim content) were calculated with 16 group neutron transport theory (ANISN):

maximum	579 mg / cc
minimum	247 mg / cc
average	474 mg / cc
Total U mass	255 kg

(U) The carbon and uranium reactivity worth coefficients were calculated using transport perturbation theory results normalized to the PAX/R-1 experimental results obtained for the Alternate 2 (Standard Geometry-High CTE) Design. An adjustment was necessary in zirconium absorption cross sections in order to duplicate the experimental results. A revised WANEF experimental program would be required to determine the core neutronic effect of the large zirconium mass evident in this design. This program would be required prior to release of the R-1 fuel loading specification. The material reactivity worths estimated for this design are:

Uranium worth	+11.5¢/kg
Carbon worth	+ 4.3¢/kg
Zirconium worth	+ 0.006¢/kg.

(CRD) Figure 13 shows the dependence of maximum uranium loading on shim content, reamed channel diameter and ZrC clad mass. The results show that the loading decreases by approximately 5.2 percent in reducing the shim capability from $\pm \$3.00$ to $\pm \$1.50$. In particular, the nominal loadings at $\pm \$1.50$ shim content are,

maximum	549 mg / cc
minimum	234 mg / cc
average	449 mg / cc
Total U mass	241 kg.

(U) A-3.5.2 Requirement Reactivity Shimming

The $\pm \$3.00$ shim capability requirement is satisfied in this design with a peak loading density of 579 mg / cc.

(U) A-3.5.3 Requirement Reactivity Loss

The carbon axial reactivity worth profile as determined by one-dimensional perturbation theory is shown in Figure 14. Thermal Design Supporting Data (Appendix A-4) estimates the reactivity loss via corrosion for this fuel design using this worth profile in conjunction with an expected carbon mass loss.

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(CRD) A-3.5.4 Requirement / Weight (U)

(CRD) The total core weight (fueled envelope) consistent with the required loadings of this design is approximately 5515 pounds.

(U) A-3.5.5 Requirement Nuclear Loading (U)

(CRD) The critical loadings determined for this design under Requirement (Nuclear Criticality) are calculated using the Alternate 2 (Standard Geometry-High CTE) Design radial loading profile. The faster neutron spectrum of this design may require an adjustment of this loading profile to accommodate this fission density requirement. An appreciable (20%) increase in peak loading can be tolerated in this regard without compromising the peak loading requirement of 700 mg/cc.

(CRD) A-3.6 Design Concept 6: Single Channel-Carbide (Drawing 947C817) (U)

(CRD) A-3.6.1 Requirement Nuclear Criticality (U)

(CRD) It is felt that the peak loading density constraint of carbide fuel is considerably higher than the required 1200 mg/cc loading estimated for this design with a ± 3.00 shim capability.

(CRD) The nominal design parameters pertinent to the nuclear characteristics of this design are,

1. 102 fueled elements per unfueled central
2. 0.170-inch across flats element dimension
3. 0.120-inch reamed channel diameter
4. One flow channel per fueled element
5. No cladding
6. 52-inch fuel element length
7. The fueled graphite density versus loading is shown in Figure 15.

(U) The distinguishing feature of this design is a fast neutron spectrum due to less carbon neutron moderation. The zirconium is a considerably less effective moderator than is carbon so that the critical loadings are significantly increases. The following critical loadings (for a ± 3.00 shim content) were calculated using 16-group neutron diffusion theory (EXTERMINATOR);

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maximum	1200 mg / cc
minimum	511 mg / cc
average	981 mg / cc
Total U mass	433 kg

(U) The material reactivity worth coefficients as determined from transport perturbation theory are:

Uranium worth	+8.8¢/kg
Carbon worth	+5.2¢/kg
Zirconium worth	+0.028¢/kg.

Available technology significantly limits the confidence level in the predicted nuclear requirements for this design since,

1. The considerable zirconium core mass magnifies the effect of any uncertainty in zirconium cross section data as was discussed for the Alternate 5 (Standard Geometry-Composite) Design.
2. The lack of experimental data evaluating the carbon and uranium worths in this neutron spectrum.

(U) Figure 16 shows the dependence of maximum uranium loading on shim content and reamed channel diameter. The results show

1. The loading decreases by approximately 4.2 percent in reducing the shim capability from ± 3.00 to ± 1.50 . In particular, the nominal loadings at ± 1.50 shim content are,

maximum	1150 mg / cc
minimum	490 mg / cc
average	940 mg / cc
Total U mass	415 kg

(U) A-3.6.2 Requirement Reactivity Shimming

The ± 3.00 shim capability requirement is satisfied in this design with a peak loading density of 1200 mg./cc.

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(U) A-3.6.3 Requirement Reactivity Loss

The carbon axial reactivity worth profile as determined using transport perturbation theory is shown in Figure 17. Thermal Design Supporting Data (Appendix A-4) estimates the reactivity loss via corrosion for this fuel design using this worth profile in conjunction with an expected carbon mass loss.

(CRD) A-3.6.4 Requirement Weight (U)

(CRD) The total core weight (fueled envelope) consistent with the required loadings of this design is approximately 8445 pounds.

(U) A-3.6.5 Requirement Nuclear Loading

The critical loadings determined for this design under Requirement (Nuclear Criticality) are calculated using the Alternate 2 (Standard Geometry-High CTE) Design radial loading profile. The hard neutron spectrum encountered in this design indicates a more severe radial fission power spike due to thermal neutron return from the reflector. The High CTE loading profile will probably not satisfy the fission power density constraint requirement in this design. A loading density profile adjustment to accommodate this requirement will likely result in a significant increase in peak loading density. Further analysis will be required in this regard if this design deserves further consideration.

(U) A-3.7 Design Concept 7: Zigzag-Composite (Drawing 947C807)

This design is nuclearly equivalent to the Alternate 5 (Standard Geometry-Composite) Design. The nuclear data outlined in Section A-3.5 is applicable to this design.

(CRD) A-3.8 Design Concept 8: High Void-Composite-Long Core (Drawing 947C814) (U)

(CRD) A-3.8.1 Requirement Nuclear Criticality (U)

(CRD) The peak uranium loading density of 575 mg / cc with a ± 3.00 shim capability required for this design does not exceed the 700 mg / cc constraint for composite fuel as discussed in Appendix A-1. The nominal design parameters pertinent to the nuclear characteristics of this design are,

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1. 1878 fueled elements
 2. 349 unfueled centrals
 3. 0.753-inch across flats element dimension
 4. 0.108-inch reamed channel diameter
 5. 19 flow channels per fueled element
 6. ZrC clad at 95 gm ZrC/element
 7. 62-inch fuel element length
 8. Fueled graphite density versus loading is shown in Figure 11.
- (U) The nominal critical loadings of this design with a +\$3.00 shim content as determined by 16-group transport theory (ANISN) are,

maximum	575 mg / cc
minimum	245 mg / cc
average	471 mg / cc
Total U Mass	285 kg

The core material worth coefficients for this design as determined by transport perturbation theory are,

Uranium worth	+12.5¢/k.g
Carbon worth	+ 4.3¢/k.g.
Zirconium worth	- 0.068¢/k.g

(U) Figure 18 shows the dependence of peak uranium loading on shim content, reamed channel diameter and clad mass. The results show that the loading decreases by approximately 4.9 percent in reducing the shim capability from +\$3.00 to +\$1.50. In particular, the nominal loadings at +\$1.50 shim content are,

maximum	547 mg / cc
minimum	233 mg / cc
average	448 mg / cc
Total U Mass	271 kg

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(U) A-3.8.2 Requirement Reactivity Shimming

The ± 3.00 shim capability requirement is satisfied in this design with a peak loading density of 575 mg./cc.

(U) A-3.8.3 Requirement Reactivity Loss

The carbon axial worth profile as determined using transport perturbation theory is shown in Figure 19. Thermal Design Supporting Data (Appendix A-4) estimates the reactivity loss via corrosion for this design using this worth profile in conjunction with an expected carbon mass loss.

(CRD) A-3.8.4 Requirement Weight (U)

(CRD) The total core weight (fueled envelope) consistent with the required loadings of this design is approximately 6300 pounds.

(U) A-3.8.5 Requirement Nuclear Loading

The combination of increased void fraction and increased core height results in a core neutron spectrum similar to the Alternate 5 (Standard geometry-Composite) Design. The Alternate 2 (Standard Geometry-High CTE) Design loading profile should satisfy the loading density requirement of this design.

(CRD) A-3.9 Design Concept 9: Standard Geometry-Graphite-Large Core (Drawing 947C815) (U)

(CRD) A-3.9.1 Requirement Nuclear Criticality (U)

(U) The peak uranium loading density of 425 mg/cc with a ± 3.00 shim capability required for this design does not exceed the 630 mg/cc constraint for graphite fuel as discussed in Appendix A-1.

(CRD) The nominal design parameters of this design are identical to those of the Alternate 2 (Standard Geometry-High CTE) Design with the exception of an increased core diameter with 2249 fueled elements and 418 centrals and fuel element length (76 inches) with the associated increase in clad mass.

(U) The nominal critical loadings of this design as determined by 16-group transport theory (ANISN) based on a ± 3.00 shim content are,

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maximum	425 mg / cc
minimum	181 mg / cc
average	347 mg / cc
Total U mass	326 kg

(U) The following core material worth coefficients of this design were estimated from those of the Alternate 2 (Standard Geometry-High CTE) Design.

Carbon worth	+3.3¢/kg
Uranium worth	+10.3¢/kg
Niobium worth	-4.1¢/kg

(U) Figure 20 shows the dependence of peak loading density on shim content, reamed channel diameter, clad material and clad mass. The results show a reduction in loading of 4.7 percent in going from a $\pm$ \$3.00 to a $\pm$ \$1.50 shim content. In particular, at the $\pm$ \$1.50 shim content, the loadings are,

maximum	405 mg / cc
minimum	172 mg / cc
average	331 mg / cc
Total U mass	311 mg / cc

(U) A-3.9.2 Requirement Reactivity Shimming

The $\pm$ \$3.00 shim capability requirement is satisfied in this design with a peak loading density of 425 mg / cc.

(U) A-3.9.3 Requirement Reactivity Loss

The carbon axial worth profile is shown in Figure 21. Thermal Design Supporting Data (Appendix A-4) estimates the reactivity loss via corrosion for this design using this profile in conjunction with an expected carbon mass loss.

(CRD) A-3.9.4 Requirement Weight (U)

(CRD) The total core weight (fueled envelope) consistent with the required loadings of this design is approximately 6085 pounds.

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(U) A-3.9.5 Requirement Nuclear Loading

The radial fission density profile is closely similar to that of the Alternate 2 (Standard Geometry-High CTE) Design and should, therefore, satisfy the fission density constraint requirement.

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TABLE 1

(CRD) LIST OF ALTERNATES FOR FUEL ELEMENT CONCEPTUAL DESIGN TRADE STUDY (U)

Alternate	No. Holes	Across Flats	Ream. Dia.	Matrix Material	Coating*		Core Dimensions		
					TK. Mils	Material	Dia. in.	Length in.	Misc.
1. Std. Geom.-Int.Exp. Graph.	19	.753	.102	GLFI	2.2	75gNb, ZrC	37.1	52	
2. Std. Geom.-Hi Exp. Graph.	19	.753	.102	AXM	2.2	75gNb, ZrC	37.1	52	
3. Std. Geom.-Hybrid	19	.753	.102	35"AXM, 17"Comp.	2.2	ZrC	37.1	52	
4. 37 Hole - Hybrid	37	.799	.084	35"AXM, 17"Comp.	2.2	ZrC	37.1	52	
5. Std. Geom.-Composite	19	.753	.102	Composite	2.2	ZrC	37.1	52	
6. Single Channel-Carbide	1	.170	.120	ZrC-UC		None	37.1	52	
7. Zig Zag-Composite	19	.753**	.102	Composite	2.2	ZrC	37.1	52	.173 Pitch
8. Hi Void-Comp.-Long Core	19	.753	.108	Composite	2.2	ZrC	37.1	62	
9. Std. Geom.-Graph-Large Core	19	.753	.102	AXM	2.2	84%NbC	40.6	76	

* The thickness given is the mean thickness over the entire length.

** This is the distance between the "Mean Line" of the two opposite sides.

A-3-21

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Figure A-1.3-1. (CRD) Intermediate CTE Graphite Density Versus Uranium Loading (U)

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Figure A-1.3-2. (CRD) Maximum Uranium Loading Versus Reamed Channel Diameter for the Intermediate CTE Fuel Design (U)

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Figure A-1.3-3. (CRD) High CTE Graphite (AXM Flour) Density Versus Uranium Loading (U)

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Figure A-1.3-4. (CRD) Maximum Uranium Loading Versus Reamed Channel Diameter for the High CTE Fuel Design (U)

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Figure A-1.3-5. (U) Carbon Axial Worth Profile for High CTE Graphite Fuel

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Figure A-1.3-6. (CRD) Hybrid Element Graphite Density Versus Uranium Loading (U)

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Figure A-1.3-7. (CRD) Maximum Uranium Versus Reamed Channel Diameter
for the 19-Hole Hybrid Fuel Design (U)

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Figure A-1.3-8. (CRD) Effect on Maximum Fuel Loading of Changing the Interface Dimension in the Hybrid Fuel Element Design (U)

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Figure A-1.3-9. (U) Carbon Axial Worth Profile for the Standard Hybrid Design

Figure A-1.3-10. (CRD) Maximum Uranium Loading Versus Reamed Channel Diameter for the 37-Hole Hybrid Fuel Design (U)

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[REDACTED] A
[REDACTED] 54

Figure A-1.3-11. (U) Carbon Axial Worth Profile for the 37-Hole Hybrid Design

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Figure A-1.3-12. (CRD) Composite Graphite Density Versus Uranium Loading (U)

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Figure A-1.3-13. (CRD) Maximum Uranium Loading Versus Reamed Channel Diameter for the Composite Fuel Design (U)

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Figure A-1.3-14. (U) Carbon Axial Worth Profile for Composite Fuel

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Figure A-1.3-15. (CRD) Carbide Graphite Density Versus Uranium Loading (U)

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Figure A-1.3-16. (CRD) Maximum Uranium Loading Versus Reamed Channel Diameter for the Solid Carbide Fuel Design (U)

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Figure A-1.3-17. (CRD) Carbon Axial Worth Profile for Solid Carbide Design (U)

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Figure A-1.3-18. (CRD) Maximum Uranium Loading Versus Reamed Channel Diameter for the High Void-Composite Long Core Design (U)

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Figure A-1.3-19. (CRD) Carbon Axial Worth Profile for High Void-Composite - Long Core Design

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Figure A-1.3-20. (CRD) Maximum Loading Versus reamed Channel Diameter for Standard Geometry Graphite-Large Core (U)

Figure A-1.3-21. (CRD) Carbon Axial Worth Profile for Standard Geometry-Graphite-Large Core Design

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APPENDIX A-4

THERMAL DESIGN SUPPORTING DATA
TRADE STUDY FOR NERVA FUEL ELEMENT
DESIGN CONCEPT
TRADE STUDY NO. 759

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A-4(CRD) THERMAL DESIGN SUPPORTING DATA (U)

A-4.0 (CRD) INTRODUCTION AND ANALYTICAL BACKGROUND (U)

(U) This appendix presents the thermal, hydraulic and corrosion aspects of the nine fuel element design concepts evaluated in this study and an assessment is made of how well each fuel element concept meets the requirements in these areas. The individual requirements are:

Flow and Thermal Characteristics

(U) The fuel elements must have axial flow passages for the propellant, exhausting it at the nominal fuel exit gas conditions of 450psia, 4284°R. The pressure drop to maintain the required flow must be within limits imposed by the capability of the axial support system. The core pressure drop has been calculated for each design alternate using a consistent set of assumptions, as given by Table A-4.0-1. The table includes the average flow rate per channel, the minimum equivalent impedance diameters for two assumed coating profiles, and the resulting core pressure drops for the maximum radial power factor channel.

(U) The propellant flow must be distributed throughout the fuel elements so as to equalize temperature levels as closely as possible. Differences in average material temperature from element to element and transverse gradients across an element are to be minimized in order to reduce stress levels within the fuel. To accomplish these objectives, the hydraulic impedance of the flow passages should be as low as possible and the local heat generation and flow impedance during operation should be accurately predicted. Also, the elements must be compatible with the Cluster Hardware Specification No. EC-677558, and the element must present the minimum thermal resistance to transverse transfer of heat. The thermal performance of each fuel alternate is summarized in Table A-4.0-2. Nominal and statistical maximum (3σ) temperatures are given for the channel wall and the fuel matrix. Also given is the maximum temperature difference between fuel matrix and channel wall, maximum volumetric heating rate, maximum thermal stress function, $Q\Psi$, (see Appendix A-1, Section A-1.9.1 for definition) and a normalized measure of the temperature gradient that results

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TABLE A-4.0-1

(CRD) SUMMARY - HYDRAULIC PERFORMANCE OF FUEL ELEMENT ALTERNATES (U)

Alternate	Average Flow Per Channel lb/sec.	Equivalent Impedance Diameter inches		Pressure Drop ⁽¹⁾ Psi	
		(2)	(3)	(2)	(3)
1, 2, 3, 5, and 7	.002520	.0935	.0914	162	176
4	.001473	.0755	.0734	161	177
6	.002684	.120*	NA	~50	NA
8	.002520	.0995	.0974	141	150
9	.002097	.0935	.0914	158	174

1. Based on 1.1 power factor except Alternate 6 (carbide) which is based on 1.0 power factor.
2. Minimum equivalent diameter for PW-2 development coating profile - 2.18 mils average thickness.
3. Minimum equivalent diameter for PW-2 experimental thick hot end coating (Dwg. 939J709-G01).

* No Coating

TABLE A-4.0-2

(CRD) SUMMARY - THERMAL PERFORMANCE OF FUEL ELEMENT ALTERNATES (U)

Alternate	(1, 3)				(1, 2, 3)	
	Maximum Nominal Temp. -°R				Max. 3~ Temp. -°R	
	T _{wall}	Sta. Inch	T _F	Sta. Inch	T _{wall}	T _F
1 and 2	4389	48	4463	46	4737	4811
3, 5, and 7	4389	48	4473 ⁽⁵⁾	46	4737	4822 ⁽⁵⁾
4	4353	50	4378	50	4701	4726
6	4417	46-48	4449 ^(6,7)	46	4765	4797 ^(6,7)
8	4371	59.5	4434	57	4719	4782
9	4340	73	4373	73	4688	4721

1. Isolated channel calculation by MCAP, for equilateral triangular array of coolant holes, and 1.1 power factor.
2. Represents value for Core Periphery after one power test, neglecting control drum effects, but including $\pm 63^{\circ}\text{R}$ uncertainty in core mixed mean temperature.
3. Assumes radial and axial max/average peaking factors are not affected by changes in fuel and core geometry.
4. Relative transverse thermal stress function defined by Appendix A-1, Section A-1.9.1.
5. Assuming thermal conductivity of 30 v/o composite is 88% that of fueled graphite at $\sim 4500^{\circ}\text{R}$, See Appendix A-1, Section A-1.4.1.4, Figure A-1.4-19.
6. Thermal conductivity for 90% theoretical density material.
7. Assuming axial distribution of power for carbide fuel is the same as reference. For perfectly flat axial profile, divide Q max. and Q ψ by 1.4. Assumes radial power factor of 1.0.

Table continued on next page.

TABLE A-4.0-2 (Continued)

(CRD)SUMMARY-THERMAL PERFORMANCE OF FUEL ELEMENT ALTERNATES (U)

Alternate	Max. $\Delta T_{F - T_{wall}}^{(1,3)} \text{ } ^\circ\text{R}$	Max. $Q^{(3)} \text{ Btu/sec-in}^3$	Max. $Q\psi^{(3,4)} \text{ Btu/sec-inch}^3$	Normalized Temp. ⁽⁸⁾ Gradient Per Unit Heat Flux
1 and 2	173	66.4	0.0714	1.0
3, 5, and 7	177	66.4	0.0714	1.33
4	88	72.5	0.037	1.42
6	94 ⁽⁷⁾	75.3 ⁽⁷⁾	0.0447 ⁽⁷⁾	0.3
8	136	58.9	0.054	1.33
9	94	45.4	0.040	1.0

1. Isolated channel calculation by MCAP, for equilateral triangular array of coolant holes, and 1.1 power factor.
2. Represents value for Core Periphery after one power test, neglecting control drum effects, but including $\pm 63^\circ\text{R}$ uncertainty in core mixed mean temperature.
3. Assumes radial and axial max/average peaking factors are not affected by changes in fuel and core geometry.
4. Relative transverse thermal stress function defined by Appendix A-1, Section A-1.9.1.
5. Assuming thermal conductivity of 30 v/o composite is 88% that of fueled graphite at $\sim 4500^\circ\text{R}$, See Appendix A-1, Section A-1.4.1.4, Figure A-1.4-19.
6. Thermal conductivity for 90% theoretical density material.
7. Assuming axial distribution of power for carbide fuel is the same as reference. For perfectly flat axial profile, divide Q max. and $Q\psi$ by 1.4. Assumes radial power factor of 1.0.
8. Normalized temperature gradient per unit heat flux across the element equals across flats dimension, F , over thermal conductivity, K , (at 4000°R), ratioed to reference.
9. Element-to-element temperature difference, neglecting control drum effects is estimated to be $\pm 106^\circ\text{R}$ (3σ) for all alternates.

from conducting a fixed amount of heat across the element. The assumptions and background information for the material in this table are given in the footnotes. The statistical maximum (3σ) element-to-element temperature difference, neglecting radial conduction between elements, is estimated to be the same for all alternates, as given by the table.

Reactivity Loss

(U) The total reactivity loss of the core, after the performance requirements have been met, shall not exceed the capability of the control system to provide compensating positive reactivity insertion. The reactivity loss consists primarily of the effect of fuel element carbon loss through corrosion. The radially and axially integrated carbon reactivity loss must, therefore, be less than the specified \$2 requirement to \$3 goal.

(CRD) There is little to be gained in increased amounts of reactivity compensation for corrosion beyond the point where the element has begun to deteriorate rapidly. For example, the NRX-A5 operated 30 minutes at high power with a total reactivity loss of \$2.00, however, the last \$1.00 in reactivity loss only increased the operating time by 6 minutes. The relation between corrosion reactivity loss and fuel element condition is considered below for the various fuel element designs.

(CRD) First, the incremental weight loss which a fuel element can sustain without rapid deterioration must be considered. Experience with NRX fuel is indicated in Table A-4.0-3. From these results, a loss of about 1.5 gm/in per standard graphite element is the most that would be desirable, and even this amount may be questionable. There is no similar experience for composite elements, however, their capability before rapid corrosion deterioration starts should equal or exceed that of graphite elements.

(CRD) The reactivity loss depends on the carbon loss distribution, since the reactivity worth of carbon varies with position. Therefore, depending on the way in which the element's corrosion is distributed axially, a point of rapid deterioration may be reached at various reactivity levels. Assuming that the maximum incremental weight loss at the hot end of the element is equal to the maximum incremental weight loss in the mid-band region, with the relative axial distribution as described in Section A-4.0.2, the operating life could be

TABLE A-4.0-3

(CRD) NRX FUEL ELEMENT MAXIMUM INCREMENTAL WEIGHT LOSS EXPERIENCE (U)

Elements	Axial Peak Incremental Weight Loss		Comment
NRX-A6 Reactor	Core Average	0.44 gm/in	No rapid deterioration. Reactivity loss was nearly linear with time.
	Central Region Average	0.55 gm/in	
NRX-A6 QC Test	Average	0.95 gm/in	No rapid deterioration evident from these and NRX-A6 element electric tests for other time intervals.
NRX-EST	Best radial region	1.0 gm/in	Extremely rapid deterioration at end of test.
	Worst radial region	1.6 gm/in	
NRX-A5	Best radial region	1.6 gm/in	Extremely rapid deterioration at end of test.
	Worst Radial region	4.3 gm/in.	

NOTE: With 1.5 gm/inch removed equally from the surface of all 19 bores, a total of 22% of the internal web between bores is removed.

extended with provision for more corrosion reactivity compensation rather than \$2.00. However, the possibility of the peak mid-band and hot end incremental weight losses being equal appears to be remote. Any other axial weight loss distribution would result in a lower reactivity loss when the point of rapid deterioration is reached. For example, in the all graphite element with standard nineteen hole geometry (Alternates 1 and 2) \$2.00 reactivity loss is reached when both mid-band and hot end weight loss peaks are 1.17 gm/inch. But when the entire weight loss is in the hot end of the element (assuming mid-band corrosion is completely suppressed) the \$2.00 reactivity loss is reached at 3.07 gm/inch, maximum in the hot end, well past the expected point of rapid deterioration.

Corrosion

(CRD) The weight loss characteristics of the core, after meeting the performance requirements must satisfy the reactivity loss requirements given above. In addition, the peak incremental weight loss should not exceed the amount required to assure that the element, in conjunction with the hot end support components, remains sufficiently intact to support the imposed axial load and prevent ejection of parts. For all elements except the UC-ZrC carbide, corrosion protection is provided by coating the fuel element bores with NbC, ZrC, or a combination. Coatings on the fuel element external surfaces may or may not be required to limit the amount of interstitial surface corrosion.

(U) In addition to discussions evaluating each fuel element concept, this appendix first reviews pertinent background information that is relevant to the cited requirements. These preceeding sections contain detailed discussions on fuel element bore corrosion effects, fuel element corrosion/reactivity loss and fuel element external corrosion. For a detailed discussion on flow and thermal characteristics, refer to Section A-4.1.

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(CRD) The fuel element bore corrosion effects in Section A-4.0.1 include comparisons of test data on diffusion corrosion of fuel elements. Comparisons of diffusion corrosion for graphite and composite fuel are made in terms of incremental weight loss rate multiplied by local coating thickness plotted against inverse temperature. Diffusion weight loss compared in this manner is not dependent on matrix, i.e., graphite or composite fuel, based on tests thermal cycled for one hour. However, with extended testing beyond one hour, all elements exhibit a tendency (in varying degrees) to depart from this diffusion weight loss relationship, which places a limitation on its use.

(CRD) In Section A-4.0.2 some relative measures of fuel element corrosion and reactivity losses are defined. These are calculated for the fuel alternates using the reactivity coefficients given in Appendix A-3. One of these corrosion measures is hot end weight loss consisting only of diffusion carbon loss (defect-free coating) and its partial contribution to the total reactivity loss. This is calculated as a function of axial position using channel wall temperatures from MCAP and a coating thickness profile based on PW-2 development elements. Another measure is the relative mid-band weight loss and its reactivity loss contribution. This assumes 10 grams per element of mid-band corrosion assigned to the reference element, with a geometric adjustment proportional to bore surface and element length for other alternates.

(U) The assessment of fuel element external surface corrosion in Section A-4.0.3 pertains to cores utilizing close-fitting fuel and central elements which form a full length distributed seal against this axial leakage. NRX-A6 reactor test results are reviewed to form the basis for evaluating external surface corrosion. Advantages of an external carbide coating are given and finally, the relative external surface corrosion is compared for the nine fuel element alternates. Specifically, Appendix A-4 contains:

- A-4.0.1 Development of the carbon diffusion relationship for coated coolant channels
- A-4.0.2 Corrosion and reactivity loss based on fuel element geometry and core size
- A-4.0.3 Fuel element external surface corrosion
- A-4.1 through A-4.9 Thermal, hydraulic and corrosion evaluations for the nine fuel element design alternates.

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A-4.0.1 (CRD) Development of Carbon Diffusion Relationship for Coated
Coolant Channels (U)

(CRD) In this section a generalized corrosion rate as a function of temperature is developed for carbon diffusion through NbC or ZrC coating. The relationship is based on experimental data from electrical corrosion tests on three different groups of fuel elements, (1) LASL tests on ZrC coated 30 % composite elements, (2) WANL tests on PW-2 development graphite elements with GEM NbC-HED ZrC coatings, and (3) WANL tests on the NRX-A6 vintage GED NbC coated elements with Moly overcoating. The limitations of such an approach are discussed in connection with the tendency for corrosion rates in cyclic tests to depart from this diffusion relationship after a period of testing.

A-4.0.1.1 Fuel Element Carbon Diffusion Corrosion

(U) In a fuel element in which the carbide coating in the bore maintains complete separation of the matrix material from hydrogen, corrosion caused by diffusion of carbon through the carbide coating will still occur. The diffusion relation for the carbon removed per unit area of channel bore is given in terms of coating thickness t , operating time Θ , and the diffusion coefficient of the coating material, D :

$$\frac{W}{A} = \frac{D\Theta}{t} \quad (1)$$

Applying the reciprocal temperature exponential form of the diffusion coefficient gives the carbon loss relation as a function of the temperature of the coating, T , and a coating material constant, C :

$$\frac{W}{A} \propto \frac{\Theta}{t} e^{-\frac{C}{T}} \quad (2)$$

Equation (2) shows that this corrosion measure is dependent only on the coating material

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and thickness and the operating temperature of the coating. The fuel element geometry and core size will affect the amount of carbon loss per unit bore area because they affect the coating temperature.

(U) In the following subsection some reactor and electric furnace test data which substantiates the use of the diffusion equation are given.

A-4.0.1.2 (CRD) Graphite Fuel Carbon Diffusion Corrosion (U)

(U) Carbon diffusion losses which are correlated by equation (2) would be represented by a straight line versus reciprocal temperature on semilogarithmic graph paper. NRX-A6 fuel element data from the reactor and electric test have been correlated in this manner. Figure A-4.0-1 shows the average axial corrosion loss distribution for both the reactor and electric QC test. These results and results of electrical tests of NRX-A6 fuel elements at other conditions are plotted in the manner described above on Figure A-4.0-2. The 35-40" and 45-50" increments were selected because there were furnace temperature measurements available on these increments. The vertical bands are a measure of the scatter in the results for individual elements, while the estimated accuracy of reactor temperatures are indicated by the horizontal bands. The electric test results correlate well with a straight line. The reactor results are higher than the electric test results, however, the difference is within the uncertainty of the channel wall temperature determinations. The same reference line will be included in all figures of this type.

(CRD) The effect of coating thickness could not be shown by these NRX-A6 results since coating profiles were similar for all elements. Figure A-4.0-3 gives the coating thickness profile of the NRX-A6 elements and PW-2 development GEM-HED fuel elements. The axial distribution of corrosion weight losses of these elements for a one hour test at the same conditions are compared in Figure A-4.0-4. Although the comparison is complicated by the fact that the development element tests were 6 cycles and the

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(CRD) Figure A-4.0-1 Average Bore Weight Loss Axial Distribution NRX-A6 Reactor Test & NRX-A6 Q.C. Furnace Testing (U)

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(CRD) Figure A-4.0-2 Comparison of NRX-A6 Reactor and Furnace Test Results 5-Inch Increments and Bore Wall Temperature (U)

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A-4-13

(CRD) Figure A-4.0-3 Comparison of NRX-A6 and PEWEE-2 Development Element Coating Thickness Profiles (U)

(CRD) Figure A-4.0-4 Comparison of NRX-A6 and PW-2 Development Element Electric Tests (U)

TABLE A-4.0-4

(U) WANL-LASL ELECTRICAL TEST COMPARISON LASL NO. 3 FURNACE (U)

ITEM	WANL	LASL	COMMENTS
Heated Length (in)	46.5	45.2	Shorter heated length gives rise to greater local heat generation and higher values of maximum fuel temperature.
Thermal Radiation Shields	Moly shields - 3 concentric partial length at hot end and one full length outer shield.	Full length uncoated graphite shields.	Graphite shields produce higher radiation losses from element attendant with greater radial temperature gradients - also full length graphite shields conduct heat axially to the cold end and may transfer heat back to the element.
Helium Exhaust	Hot End	Near cold end for "ΔP" tests.	Cold end exhaust produces convective heating at cold end - helium to element.
Forward point of contact - hot end chuck (52 in. element)	49.1 in. 6-finger chuck	48.6 in. 3-jaw x 3/4 in. forward of 2 in. broached block	
Measured hot end temperature	Scraped surface at Station 49	Drilled hole to outer web at station 45 also exit gas.	Temperature difference web to channel wall is greater than max. surface to channel wall.
Helium Pressure	$\Delta P_{20} = 30$ psig	Mostly overpressure occasionally under pressure.	Potential influence on pinholing
Power KW	950, 880, 840	Depends on test.	

A-4-15

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NRX-A6 QC test was only 2 cycles, the effect of the thicker coating is obvious. Corrosion rates times coating thickness data from these and other PW-2 development elements is shown on a reciprocal temperature log plot in Figure A-4.0-5. These results are in reasonable agreement with the NRX-A6 elements for tests up to 60 minutes (all data points within cross-hatched envelope). The elements tested for more than 60 minutes, outside the cross-hatched envelope show an accelerated corrosion rate and do not conform to the diffusion corrosion reference line. Although relatively few data points are available, the tendency for corrosion rates to increase after the first hour is apparent for both A6.07 tests and D1.01 tests on AXM and GLF matrices. In the tests up to 60 minutes no difference between high expansion (AXM) and intermediate expansion (GLFI) matrices can be detected from this plot. On the basis of the above data, the slope of the diffusion corrosion line for NbC coating on graphite matrices was estimated to be the same as that of the reference line. This is used in all "defect-free" diffusion corrosion evaluations.

A-4.0.1.3 (CRD) Carbon Diffusion Corrosion for ZrC Coated Composite Fuel (U)

(CRD) Because of the differences in the test facility, test conditions and instrumentation, a really good comparison of the LASL composite test results and WANL graphite test results cannot be made. Some of these differences are given on Table A-4.0-4. In addition, to prevent cracking of composite elements during start-up, LASL found it necessary to extend the start up period on the first cycle from the standard 3 minute ramp to an 8 or 10 minute "soaking period". During this period, the power and flows are brought up gradually while staying below 600 mw for a reasonable time. After the first

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(CRD) Figure A-4.0-5 PW-2 Development Element Station 45 Corrosion Rates versus Bore Wall Temperature (U)

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cycle the standard 3 minute cycle can be used. To prevent the composite elements from sagging when at temperature, three (3) supports are located along the length in the horizontal furnace. There has been little evidence of elements actually touching the supports. The high loaded elements have more tendency to sag than lower loadings.

(CRD) The LASL corrosion test data on the basis of the station 45 inch incremental losses are given in Figure A-4.0-6. A good fit to the 78 k-cal/mole reference line is obtained for cyclic tests up to twelve -10 minute cycles. The station 45 inch temperature is a material temperature measured optically at the base of a hole drilled into the side of the element, as illustrated in Figure A-4.0-7. The diffusion corrosion line for ZrC coating on composite matrices is also selected to be the reference line, as for graphite fuel.

(CRD) Caution should be used in making direct comparisons of this LASL data to similar WANL data for graphite elements in Figures A-4.0-2 and A-4.0-5, because of the differences in furnaces, operating conditions, temperature measurement, and the fact that different axial increments are considered. It is noteworthy that the composite elements do not appear to be experiencing accelerated corrosion rates after one hour of cyclic testing as was indicated in Figure A-4.0-5 for graphite elements. However, very limited data, at the time of writing, does indicate that they may begin to deteriorate after two hours of cyclic testing. These data are called out at the lower right of Figure A-4.0-6. The two 40 % composite elements were tested for 1 x 180 and 11 x 10, respectively, and the latter shows approximately a factor of two higher corrosion rate. More significantly, the two 30 percent composite elements tested for 10 hours indicate a factor of two higher corrosion rate with 60 temperature cycles instead of one. Both of these indications are at relatively low temperatures (4000°R measured at the 45 inch drilled hole).* Additional cyclic testing of 20 to 50 cycles at different temperature levels is required to learn more about the influence of cycling on both mid-band and hot-end loss rates. A more complete discussion of these LASL 10-hour tests is contained in Section A-4.5.2.

* Graphite elements have not been tested at comparably low temperatures.

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(CRD) Figure A-4.0-6 LASL Corrosion Rate Comparisons Composite
Fuel versus Station 45 Outer Web Temperature(U)

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Figure A-4. 0-7. (CRD) Diagram of LASL Drilled Hole Location at Station 4b (U)

(CRD) A-4.0.1.4 Limitations of Carbon Diffusion Relationship (U)

(CRD) At several points during the preceding discussion, mention was made of the tendency for corrosion rates to increase or "accelerate" after a period of cyclic testing. This departure from the diffusion corrosion relationship is a fundamental and serious limitation in its use and one which is only partly understood. The phenomenon has sometimes been given the name "defect corrosion" since it is an indication that the coating diffusion barrier has been penetrated at localized places in the fuel element bores. There may be several different causes for these defects, acting concurrently, and in fact there is reason to believe that different defect propagation mechanisms may exist in different axial sections of the element. Note that the data supporting the carbon diffusion relationship was all based on the incremental weight losses taken from the "hot-end" of the element, beyond axial station 35 inches. In this region, the defects apparently are absent during the initial cycles and may not appear at all when the element is not cycled (LASL 1x600 test). This suggests a "fatigue" type of behavior which builds up damage to coating due to some gradual worsening of initially minor coating defects or some change in the coating and/or adjacent matrix properties. The differential thermal expansion between coating and matrix in the first cycle causes the matrix to be in tension while the coating is in compression at the hot end. In this state, any coating cracks which may have existed at ambient conditions in this region would tend to be closed.

(CRD) The tendency for defects to form at the hot end of the fuel element is a very complex situation to analyze. Some of the variables which are thought to influence defect formation are the following:

1. The coating deposition temperature.
2. The average temperature of the coating and matrix in operation at the axial station(s) of interest.
3. The adherence of the coating to the matrix as a function of time at temperature, corrosion and thermal cycles.

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4. The geometry of the coating and matrix, including relative matrix area to coating cross section areas.

5. Properties of both coating and matrix, including the manner in which the properties change with temperature, time at temperature, thermal cycles, and corrosion.

5.1 Thermal coefficients of expansion.

5.2 Ductility, (or brittleness).

5.3 Residual stresses within coating or matrix as a result of manufacturing process.

5.4 Tendency to take "permanent sets" of dimension changes caused by graphitization, creep at high temperature, "thermal ratcheting" or other temperature induced causes.

(CRD) The preceding is an oversimplified discussion of some of the variables which may affect coating defects at the hot end of the fuel element. It is intended to illustrate the very complex nature of the phenomenon, how it may interact with diffusion corrosion, and why a theoretical approach to explain differences and predict performance is not justified at the present time. Also, it is possible that other defect-causing mechanisms may be present or may appear after two or more hours of cyclic testing.

(CRD) The preceding has said nothing about mid-band corrosion in the cooler portion of the element which also is related to coating defects. However, in this instance the mechanism is different, and when serious mid-band corrosion occurs, it is present in the first cycles of testing. Based on the coating-matrix axial stresses in Appendix A-5, the mid-band defect corrosion is probably caused by the tensile stresses in operation, opening coating cracks which are normally closed at ambient temperature. Differences in mid-band corrosion resistance exist for graphite and composite elements and are discussed in the appropriate sections for the fuel alternates.

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A-4. 0.2 (CRD) Corrosion and Reactivity Loss Based on Fuel Element Geometry and Core Size (U)

(U) The evaluation of each fuel element design alternate with respect to corrosion and reactivity loss consists of two parts. One is an assesment of each coating-matrix combination's potential to meet the long duration cyclic requirements, discussed in the respective section devoted to corrosion for each alternate. It relies primarily on interpretation of electrical corrosion test results for the standard geometry graphite and composite elements, alternates 1, 2, and 5. The other part, described in this section, calculates corrosion and reactivity loss on the basis of fuel geometry and core size. It is assumed that no fundamental difference in basic material properties or residual stresses due to manufacturing porcesses would be caused by a change from the standard geometry.

(U) A geometric description of each fuel element, the peak channel wall temperature of a nominal element and the carbon reactivity coefficients are given in Table A-4. 0-5.

Hot End Diffusion Corrosion and Reactivity

(CRD) One measure of the relative capability of the different element geometries and core sizes is their relative diffusion corrosion loss and associated reactivity loss. These values were calculated for ten hours of operation based on the following assumptions:

1. Coating thickness distribution is an average of the PW-2 development element distribution.
2. Defect-free hot end corrosion based on carbon diffusion only.
3. Carbon diffusion coefficient developed in the previous section (Reference line on Figures A-4. 0-2 and A-4. 0-6). This is the same for ZrC coated composite elements and NbC/ZrC coated graphite elements.
4. Normalized carbon reactivity coefficient axial profile.

(U) The hot end weight loss was calculated using the channel wall temperature distribution obtained from the MCAP computer program for each fuel element alternate and the PW-2 coating profile shown in Figure A-4. 0-8 . The worth of carbon loss for each

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(CRD) Figure A-4.0-8 PEWEE-2 Fuel Development Average Coating Profile Normalized Axially (U)

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increment was determined from the carbon axial reactivity profile shown in Figure A-4. 0-9 and the reactivity data from Appendix A-3, which are also listed in Table A-4. 0-5.

(CRD) The results of this evaluation are given for each alternate in Table A-4. 0-6. Columns 2 and 3 give the diffusion carbon loss per fuel element and the maximum axial incremental weight loss per fuel element. The amount of carbon which diffuses through a unit area of carbide coating could be a measure of the loss of coating adherence to the matrix. This number which is only a function of fuel channel wall temperature, is listed in Column 4 on a normalized basis. In this case the 37 hole element is better than all others excepting the large core element.

(U) A final measure of fuel condition is the axial peak percent of carbon removed from the interior web between fuel element coolant channels, Column 5.*

(CRD) Alternate 9, the graphite element in the large core is best in this respect followed by the two standard geometry graphite elements, alternates 1 and 2. The reactivity loss caused by the hot end diffusion corrosion is listed in Column 6.

Relative Midband Corrosion

(CRD) An assumed midband corrosion weight loss is arbitrarily assigned to the reference element for comparative purposes. This midband corrosion was adjusted to the other elements in direct proportion to their bore surface area. This relative midband corrosion and the associated reactivity loss is given in the final two columns of the table for the nine alternates. The axial distribution of the assigned midband corrosion is based on the NRX-A6 reactor distribution shown in Figure A-4. 0-1.

* Assuming carbon is removed equally from the circumference of the bore, then incremental grams lost can be related to an annular region from which carbon has been removed and having a certain depth dimension from the bore wall. The percent carbon removed from an interior web is defined as the ratio of this distance to the minimum web thickness.

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Figure A-4.0-9(U) Normalized Fuel Carbon Reactivity Coefficient (U)

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A-4-26

TABLE A-4. 0-5

(CRD) ALTERNATE DESIGN FUEL ELEMENT PARAMETERS(U)

Design Alternate	Length - in	Reamed Chan-Diam.	Chan-Center Spacing-in	Fuel Void Fraction	No. of Channels	Core Fueled Diameter	Max. Wall Temp. -°R	Avg. Carbon Reac. Coeff. ℓ/Kgm	Channels Per Element
1.(Std.Int-Exp.Graph)	52	.102	.173	.314	35682	37.1	4389	3.90	19
2.(Std.H-Exp.Graph)	52	.102	.173	.314	35682	↓	4389	3.90	19
3.(Std.Hybrid)	52	.102	.173	.314	35682		4389	3.99	19
4.(37 Hole Hybrid)	52	.084	.1309	.373	61050		4353	4.28	37
5.(Std. Composite)	52	.102	.173	.314	35682		4389	4.30	19
6.(Single-ch.Carbide)	52	.120	.170	.453	36843		4417	5.24	1
7.(Zig-Zag Comp)	52	.102	.173	.314	35682		4389	4.30	19
8.(HV-Comp.Long)	62	.108	.173	.354	35682	↓	4371	4.30	19
9.(Std.H-Exp.Graph.Large)	76	.102	.173	.314	42864	40.6	4340	3.21	19

TABLE A-4. 0-6
(CRD) FUEL GEOMETRY AND CORE SIZE EFFECT ON HOT END DIFFUSION
CORROSION AND RELATIVE MIDBAND CORROSION (U)

Design Alternate	Hot End Diffusion Corrosion (a, b) For 10 Hour Operation					Relative Mid-Band Corrosion (c)	
	GMS-HE Per Elem.	Maximum Gms/In.	Relative Gms Carbon Removed Per Unit Bore Area	Percent Carbon Removed From Min. Web Dim.	Reactivity Loss - \$	GMS-MB Per Elem.	MB - \$
(1)	(2)	(3)	(4)	(5)(d)	(6)	(7)	(8)
1.(Std.Int-Exp.Graph)	14.7	1.17	1.0	17.2	0.70	10.0	0.78
2.(Std.H-Exp.Graph)	14.6	1.17	1.0	17.5	0.70	10.0	0.78
3.(Std. Hybrid)	14.6	1.17	1.0	19.7	0.76	10.0	0.78
4.(37 Hole Hybrid)	20.6	1.66	0.87	26.0	1.02	16.0	1.17
5.(Std. Composite)	14.6	1.17	1.0	19.7	0.78	10.0	0.85
6.(Single-ch.Carbide)	-----	-----	-----	-----	-----		
7.(Zig-Zag Comp.)	14.6	1.17	1.0	19.7	0.78	10.0	0.85
8.(HV-Comp.Long)	17.0	1.13	0.92	19.7	0.86	12.6	1.08
9.(Std.H-Exp.Graph.Large)	15.8	0.96	0.84	14.4	0.68	14.6	1.12

- a) PEWEE-2 development coating profile, i.e., ~3.1 mils thickness at hot end.
b) Defect free hot end corrosion based on carbon diffusion loss only.
c) Mid-band corrosion proportional to bore surface area and fuel length with a ten gram loss assumed for the reference element.
d) Assuming carbon is removed equally from the circumference of the bore, then incremental grams lost can be related to an annular region from which carbon has been removed and having a certain depth dimension from the bore wall. The percent carbon removed from an interior web is defined as the ratio of this distance to the minimum web thickness.

(CRD) The corrosion and reactivity comparisons on the basis of fuel element geometry as outlined above are valid insofar as the listed assumptions remain valid. The most important caution which the reader must bear in mind is that no coating or coating-matrix* combination available at the present time remains defect-free through 10 hours of cyclic operation. On the basis of limited, and so far inconclusive electrical corrosion tests, design alternates 1, 2 and 5, which have been tested, appear to differ in ability to resist the formation of defects in operation. It is not known whether or not changes in fuel element geometry would affect the formation of coating defects, except that circumferential cracks in the coating theoretically are minimized by keeping the coating in axial compression (and the matrix in tension). This condition is more strongly a function of coating temperature than geometry, however.

(U) A rational choice for the best performing fuel element must, therefore rely on incomplete information for possibly the most important difference between the alternates; namely, the ability to resist coating defect formation and the resulting corrosion. In the discussions given for each alternate design, all available information is given but the final selection, in this respect, must still remain a matter of judgement.

* Excluding UC-ZrC carbide, which has no coating.

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A-4.0.3 External Surface Corrosion

(U) In this section the corrosion caused by hydrogen leakage flow in the inter-element gaps is discussed. This assessment is made for cores utilizing close-fitting fuel and central elements which form a full length distributed seal against this axial leakage. The cold-end and hot-end seal concepts are analyzed in detail in Appendix 4 of the Periphery Trade Study, Data Item S-54, Trade Study-752.

(CRD) Both of these concepts are theoretically capable of reducing the interelement leakage flow if the fuel elements were impermeable. For relatively small whole body leakages, however, the interelement mass flow is very uncertain and could be an appreciable fraction of that which would occur with no cold-end or hot-end seal. For this and other reasons, only the reference design using a distributed axial seal concept is considered in this discussion.

(CRD) In Section A-4.0.3.1 the NRX-A6 reactor test results form the basis for an evaluation of external surface corrosion on non-OD coated graphite elements. The capability to operate 10 hours in R-1 is extremely marginal for this element. In Section A-4.0.3.2, the advantages of an external carbide coating are shown to result in substantially reduced surface corrosion which is potentially acceptable. Finally, in Section A-4.0.3.3 the relative external surface corrosion is compared for the fuel element alternates on the basis that the amount of corrosion will be directly proportional to the hydrogen leakage. A summary giving relative external surface corrosion for the nine alternates is contained in Table A-4.0-7.

(CRD) A-4.0.3.1 External Surface Corrosion of a Non-OD Coated Graphite Fuel Element (U)

(CRD) External corrosion of the graphite element which does not have an external carbide coating is minimized by limiting hydrogen leakage flow between the fuel elements. The capability of the element to withstand this corrosion for long duration operation is

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estimated. NRX-A6 reactor test results form the basis for this evaluation. It is doubtful that a decrease in interelement leakage and external corrosion rates over the NRX-A6 core values can be made in the R-1 core.

Interelement Leakage Gap Geometry

(CRD) There are small random spaces between the fuel elements and central elements in the core. These spaces or Interelement gaps form a 51 inch long seal against axial hydrogen leakage from the core inlet to the core outlet. The leakage path width defined by the perimeter of the fuel and central elements is about 240 feet, while the average gap thickness is less than one mil. The average gap has been determined by analysis to be a function of the value and variation in fuel element and central element across flats dimension; the value and variation in the fuel element and central element coefficient of thermal expansion; the value and variation in the fuel element and central element temperature, including axial and radial variations as well as element-to-element variations; the fuel and central element bundling force, which depends on the interelement pressure and LSS seal pressure; and the ambient average gap size, which is measured. For the NRX-A6 reactor, the average interelement gap was calculated to be a minimum of 0.3 mil at axial station 10", where the central and fuel element were calculated to be the same width at their operating temperature in that location.

NRX-A6 Interelement Corrosion

(U) The NRX-A6 reactor, which operated one hour at fuel element temperatures very close to R-1 nominal temperatures, gave a measure of the element external corrosion which can provide a base point for R-1 evaluations.

(CRD) External corrosion of the NRX-A6 fuel was obtained with a mechanical device which measured surface regression by means of a stylus. WANL-TNR-223, "NRX-A6 Test Analysis Report", contains a description of this surface regression measurement and the results, as well as a comparison with the results of a visual method of estimating surface corrosion. Surface regression measurements were remarkably similar

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for all elements. An average regression profile is given in Figure A-4.0-10.

As discussed in the test report and shown on the figure, the indicated hot end regression is attributed to fuel element shrinkage rather than corrosion. In addition, although readings were interpreted as for one side only, the stylus probably measured regression on both sides of the element. On this basis, the average interelement corrosion is less than 1.0 grams per element (one gram is equivalent to one mil of regression over 12 inches of fuel element length). This compares to an estimate of 0.78 grams per element obtained by the visual method, and a total average measured weight loss of 1.05 grams per element between stations 10 and 25 inches. From the figure, it is apparent that the leakage hydrogen was saturated with methane and corrosion ceased at about station 23 inches. Further downstream carbon precipitated out of the gas as soot or plated out as pyrocarbon.

(CRD) From these surface corrosion measurements and weight loss measurements on the NRX-A6, it is concluded that:

Average Total External Surface Corrosion - 0.5 - 1.0 gm/element
from Station 10 - 25"

Element Average Axial Peak Regression - 0.5-1.2 mils

Element-to-element variation in regression - $\pm 1/2$ mil

NRX A-6 Fuel Element Bonding

(CRD) In the NRX-A6 core, bonding occurred to the extent that fuel matrix material was transferred from about 600 elements to adjacent fuel elements when they were pried apart. Typically, half or more of the element faces were bonded, however, only bonding of axial streaks of pyrographite deposit occurred and fuel elements did not bond to the NbC coated central elements. A calculation of interelement leakage was performed with the ICAP computer program to estimate the maximum effect of bonding on the NRX-A6. The flow gap between fuel elements was set at 0.335 mils from station 22 to 26 inches and 0.67 mils at all other axial locations. The total leakage flow rate was calculated to be 13.5 lb/hour compared to a total of 17.6 lb/hour with a uniform 0.67 mil gap. Reducing

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A-4-33

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Figure A-4.0-10. (CRD) NRX-A6 Regression Profile in Outer Region With Surface Appearance Indications (U)

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the flow gap by 50% is equivalent to increasing the flow impedance by a factor of 8. Therefore, this is equivalent to complete bonding of the fuel elements with flow only in the fuel-to-central element gap between 22 and 26 inches.

(CRD) Considering the small flow reduction for this extreme case and the limited amount of bonding which actually occurred in the NRX-A6 core, it is concluded that bonding did not significantly affect the interelement leakage flow rate or interelement corrosion. For 10-hour R-1 operation, it is conceivable that even the gaps surrounding the central elements would be reduced by carbon deposition. However, for this evaluation it is assumed that a significant reduction in interelement flow due to bonding and deposition will not occur.

Analysis of Interelement Corrosion

(CRD) Using the interelement gaps as described previously, the interelement leakage and corrosion have been calculated for NRX-A6 (NRX-A6 Reactor Analysis Data Book, WANL-TME-1642). The average surface corrosion prediction exclusive of pinhole effects was 0.61 grams per element. This agrees reasonably well with the 0.5 to 1.0 gram observed. However, the regression depth predictions of 3 to 5 mils are in poor agreement with the approximately one mil observed. Possible causes of the sharper peaking corrosion distribution obtained from the calculation have been analyzed. These include fuel element bonding, methane boundary layer effects, different interelement gaps, different fuel hydrogen reaction rate and a different temperature profile. Results of some of these analyses are shown in Figure A-4.0-11. Boundary layer effects, bonding and gap size variation do not significantly change the peak-to-average value of the calculated regression. It was found that a reduced chemical reaction rate of the fuel and an operating temperature which fluctuates with time will both cause the calculated profile to be less sharply peaked. A variable temperature across the face of the fuel element and methane mixing across this face will

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(CRD) Figure A-4.0-11

Effect of Interelement Gap, Temperature Profile and Corrosion Constant on Interelement Corrosion (U)

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also tend to reduce the peak, although these effects are not included in the analytical model.

R-1 Surface Corrosion Estimate

(CRD) The results of the preceeding analysis show items which must be considered in predicting R-1 external element corrosion.

- (1) Close chamber temperature level control combined with the flow control reactivity adjustment will lead toward constant element temperature with time and thereby, a higher peak regression rate than observed in the NRX-A6.
- (2) The calculations of interelement corrosion between stations 10 and 25 mentioned in the previous section indicate that the regression peak associated with the observed NRX-A6 surface corrosion weight loss should have been higher than stylus measurements or visual estimates shown. A complete explanation of this difference has not been obtained. Therefore, the possibility of surface corrosion effects extending more than one mil into the NRX-A6 elements must be considered. It is certainly possible that carbon removal by corrosion (such as binder corrosion) extended several mils deeper into the matrix than could be measured. On the basis of these items and the information summarized below, the capability of an uncoated graphite element to operate 10 hours in R-1 is extremely marginal.

(1)	External corrosion weight loss (10 x NRX-A6)	5 to 10 gm/FE
(2)	Mils regression based on NRX-A6 (10 x NRX-A6)	6 to 12 mils
(3)	Mils regression based on item 1 and analysis	30 to 50 mils
(4)	Reactivity loss of alternates 1 and 2, 0.754" graphite element, 52" core	0.47 to 0.94 \$
(5)	Web thickness - nominal	0.0275 inch
	minimum	0.0205 inch

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(CRD) A-4.0.3.2 Surface Corrosion of an Externally Coated Fuel Element (U)

(CRD) Surface corrosion of an element with an external carbide coating will occur by carbon diffusion through the coating or by corrosion of the matrix material at coating defects for either a composite or a graphite matrix. Hydrogen leakage flow, although limited by the close fitting fuel elements, will be greater because element tolerances will be larger than with uncoated elements.

Comparison of Diffusion Corrosion through Coating to Uncoated Graphite
Element Corrosion

(CRD) Carbon diffusion corrosion was observed under the thin external element coatings in the PW-1 reactor. The diffusion corrosion rate for a two mil external coating is compared to the corrosion rate for uncoated fueled graphite in Figure A-4.0-12. For the uncoated element, as described in the preceeding section, the reaction is inhibited by progressively increasing methane content in the hydrogen until corrosion ceases at about 3000°R.

(CRD) The coated element loses carbon under the coating at the higher temperatures. At station 46 inches (4463°R) the carbon loss shown on the figure is equivalent to a regression of 15 mils in ten hours (1.2 gm/in) for the 0.754 inch graphite element in the 52 inch core. The actual corrosion at the hot end of the core would be much less than this because it would be inhibited by methane in the hydrogen. The effect of methane on diffusion through carbide coatings has not been established, therefore external corrosion cannot be calculated for a coated element. There is some evidence from LASL experiments that several hundred parts per million of methane in hydrogen will significantly reduce this corrosion.

(CRD) If 500 ppm would completely inhibit diffusion corrosion and if R-1 inter-element leakages on the order of NRX-A6 are attained, total external corrosion would be only 0.2 grams per element in 10 hours. In summary, external coatings can potentially result in acceptable element surface corrosion.

(CRD) Figure A-4.0-12 External Corrosion of Uncoated Graphite Fuel Element and ZrC Coated Fuel Element (U)

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(CRD) A-4.0.3.3 Effect of Fuel Element and Core Geometry on Fuel Element External Corrosion (U)

(CRD) Regardless of the element type used in the core, with the exception of carbide, minimizing axial interelement leakage will minimize surface corrosion of the fuel elements. A comparative measure of the total axial interelement leakage was made on the basis of laminar flow between parallel plates assuming the same interelement gap thickness between fuel elements for all alternates. For Alternate 7, the Zig-Zag element, the assumption of a constant gap thickness is probably too optimistic. If the gap is larger for this design, the mass flow rate would increase as the cube of the gap thickness. For laminar hydrogen flow between parallel walls with a specified axial temperature profile the flow rate is related to the core pressure drop, ΔP , average pressure in the gaps, $\bar{P}$, and the passage width, B , and length, L ;

$$\dot{W} \sim \frac{\bar{P} \Delta P B}{L} \quad (3)$$

the width can be related to the element across flats dimension, F , and core diameter, D :

$$B \propto (D_c)^2 \frac{1}{F} \quad (4)$$

For the R-1 reference design, B is equal to 227 feet. Combining the above two equations gives the hydrogen flow (as external corrosion measure) as,

$$\dot{W} \propto \frac{\bar{P} \Delta P D_c^2}{FL} \quad (5)$$

For most of the designs considered, the core pressure drop is near the reference; therefore, for minimum interelement corrosion, small elements and large diameter/short length cores are undesirable. This relation is tabulated in Table A-4.0-7 for each fuel alternate

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TABLE A-4.0-7
(CRD) RELATIVE INTERELEMENT HYDROGEN LEAKAGE (U)
GEOMETRIC FACTOR $\propto \frac{\bar{P} \Delta P D_c^2}{FL}$

Alternate	ΔP Psi	$\bar{P}$ Psi	D_c Inches	F Inches	L Inches	Factor/Ratio
1, 2, 3, and 5	170	535	37.1	.753	52	$3.2 \times 10^6 / 1.0$
4	169	535	37.1	.799	52	$3.0 \times 10^6 / 0.94$
6	50	475	37.1	.170	52	$3.7 \times 10^6 / 1.16$
7	170	535	37.1	(.753)*	52	$3.7 \times 10^6 / 1.153$
8	145	523	37.1	.753	62	$2.2 \times 10^6 / 0.70$
9	163	532	40.6	.753	76	$2.5 \times 10^6 / 0.78$

- $\bar{P}$ = Average pressure in interelement gaps
- ΔP = Core pressure drop
- D_c = Core fueled diameter
- F = Element across flats dimension
- L = Core length

Interelement gap thickness, t , assumed constant for all alternates.

* Surface area ratio to regular .753 hex = 1.153.

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A-4.1 (CRD) INTERMEDIATE EXPANSION GRAPHITE ELEMENT - DWG. 947C812 (U)

A-4.1.0 (CRD) Introduction and Summary (U)

(U) The flow and thermal performance characteristics for this fuel element design are satisfactory for the R-1 reference conditions.

(CRD) Flow, thermal, and corrosion performance are the same as for the reference element, Alternate 2. Based on a limited number of electrical furnace corrosion tests, which are discussed in Section A-4.1.2.2, this matrix appears to be more susceptible to hot end defect corrosion and pinhole formation than the high coefficient of expansion elements in cyclic testing for more than one hour. However, it is also apparent that the coating process is more influential in this respect than the matrix material. Since a very large coating defect occurs when an element breaks, its resistance to thermal stresses must also be taken into account. For satisfactory corrosion performance with a graphite matrix, the prevention of element breakage is extremely important. Performance is compared with the reference, Alternate 2, in Table A-4.1-1.

(CRD) The following points are worth noting regarding projected improvements in corrosion performance:

1. This design has the longest history of use and development in the NERVA program. Consequently, the understanding of how to achieve the best corrosion performance through coating on this matrix is relatively higher than for any other alternate. The present stage of development represents the culmination of many years of trial-and-error experimentation, analysis, and materials development. By comparison, the understanding of how to minimize corrosion for a composite element could be considered in a less advanced stage of development.
2. There is nothing in the foreseeable future that appears capable of eliminating coating defects, and the consequent increase in corrosion rate with cycles.

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TABLE A-4. 1-1

(CRD) FUEL TEMPERATURE TECHNOLOGY (U)

MATERIALS

1.0 Melting Temperature:

NbC	3500°C (6800°R)
ZrC	3420°C (6650°R)
NbC - ZrC - U Minimum (LASL)	2800°C (5530°R)

2.0 Recent Y-12 Data* on Melting of Coated Specimens Heated for One Hour in Flowing Argon

ZrC on Unfueled Carbon Showed Melting At 2750 - 2900°C (5440 - 5710°R)
ZrC on 630 MG U/CC Showed Localized Melting of ZrC at 2400°C (4810°R)
and Complete Melting of ZrC at 2500°C (4990°R)
NbC-ZrC on 400 mg/cc graphite melted after heating to 2500°C (4990°R)

3.0 Bead Migration (Not Considered a Fuel Operating Temperature Limitation):

Std. Bead - Some beads migrate through shell (2%) in one hour	2300°C (4630°R)
All beads migrate through shell in one hour	2460°C (4920°R)

* Reference: Y-DA-3194 "Melting Points of Coatings of NERVA Fuel Elements," December 10, 1969

3. Presuming that we would have to "live with" the presence of coating defects, the accelerating corrosion rates noted in electrical corrosion tests on graphite elements appears to limit reactor lifetime to approximately 2 to 5 hours with this alternate.

A-4.1.1 (CRD) Flow and Thermal Characteristics (U)

A-4.1.1.2 (U) Thermal (U)

(CRD) The core pressure drop is 162 psi if the PW-2 reference coating profile is used. For an additional thickness of coating at the aft end, as defined by Drawing 939J709-G01, for PW-2 experimental thick hot end coatings, the core pressure drop would increase to 176 psi. Flow performance is identical to that of the Reference element, Alternate 2 as described by Table A-4.1-2. For comparison with the other alternates, the average flow per channel, minimum effective impedance diameter, and core pressure drop is summarized in Table A-4.0-1.

A-4.1.1.2 (U) Thermal

(U) All fuel temperatures given in this Appendix are based on the MCAP* calculation for an equilateral triangular array of coolant holes without radial conduction. As noted in Appendix A-1, Section A-1.4.1.2, the outer perimeter of the hexagonal fuel element is slightly hotter than this because of the change in hole pattern.

The axial fuel element temperature profile is shown in Figure A-4.1-1. Maximum temperature levels are given on Table A-1.4-2 for the channel wall and the fuel matrix temperature at the center of a triangular array of coolant holes, for comparison with the other alternates. The maximum statistical temperature levels given are based on the uncertainties in nuclear power prediction for the core periphery after incorporating post-test results of one R-1 power test, neglecting control drum effects. $\pm 63^\circ\text{R}$ uncertainty in fuel exit mean temperature is included.

*Reference WANL-TME-944 "A multiple Channel Analysis Program for the Temperature Distribution in an Internally Cooled Solid", October 5, 1964.

TABLE A-4. 1-2

Page 1 of 4

(CRD) COMPARISON OF FLOW, THERMAL, AND CORROSION PERFORMANCE FOR
STANDARD GEOMETRY INTERMEDIATE EXPANSION ELEMENT, ALTERNATE 1 (U)

	Alternate Dwg. 947C812	Reference Alternate 2 (Dwg. 947C813)
0. DESCRIPTION		
0.1 Matrix Material	GLFI	AXM
0.2 Coating Material	75 gm Nb, ZrC	75 gm Nb, ZrC
0.3 Reamed Diameter - Inch	0.102	0.102
0.4 Hole Pitch - Inch	0.173	0.173
0.5 Across Flats - Inch	0.753	0.753
0.6 Void Fraction	0.314	0.314
0.7 Core Length - Inch	52	52
0.8 Core Fueled Diameter - Inch	37.1	37.1
0.9 Number of Channels Per Element	19	19
0.10 Number of Channels Per Core	35682	35682
1. FLOW		
1.1 Equivalent Impedance Diameter-in.		
1.1.1 Min. Based on PW-2 Avg. Coating (2.18 mils)	0.0935	0.0935
1.1.2 Min. Based on Experimental PW-2 Thick Hot End Coating (Dwg. 939J709-G01)	0.0914	0.0914
1.2 Core Pressure Drop ⁽¹⁾ - Psi		
1.2.1 Based on diameter in 1.1.1	162	162
1.2.2 Based on diameter in 1.1.2	176	176

(1) Based on 1.1 power factor, except Alternate 6 (single channel - carbide), which is based on 1.0 power factor. Neglects taper in bore due to varying coating thickness.

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TABLE A-4. 1-2 (CONTINUED)

Page 2 of 4

	Alternate Dwg. 947C812	Reference Alternate 2 Dwg. 947C813
2. THERMAL		
2.1 (T_W) Max. Channel Wall Temp ⁽²⁾ °R		
2.1.1 Nominal	4389	4389
2.1.2 Hot Channel ⁽³⁾ (3 σ)	4737	4737
2.1.3 Difference from reference	0	—
2.1.4 Axial Station - Inch	48	48
2.2 (T_F) Max. Fuel Hot Spot Temp ⁽²⁾ °R		
2.2.1 Nominal	4463	4463
2.2.2 Hot Channel ⁽³⁾ (3 σ)	4811	4811
2.2.3 Difference from reference	0	—
2.2.4 Axial Station - Inch	46	46
2.3 Temperature Differences		
2.3.1 Max. ΔT ⁽²⁾ within nominal Element $(T_F - T_W)$ - °R	173	173
2.3.2 Statistical uncertainty in element-to-element average exit temp. ⁽³⁾ (3 σ) - °R	106	106
2.3.3 Normalized temp. gradient per unit heat flux ⁽⁴⁾	1.0	1.0
2.4 Max. Volumetric Heating Rate ⁽⁵⁾ Btu/sec-inch ³	66.4	66.4
2.5 Max. Thermal Stress Function ^(5, 6) ° - Btu/sec-inch	0.0714	0.0714

(2) Isolated channel calculation by MCAP for equilateral triangular array of coolant holes and 1.1 power factor (except alternate 6, single channel - carbide).

(3) Represents value for core periphery after one power test, neglecting control drum effects, but including $\pm 63^\circ\text{R}$ uncertainty in mean fuel exit gas temperature.

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TABLE A-4. 1-2 (CONTINUED)

Page 3 of 4

	Alternate Dwg. 947C812	Reference Alternate 2 Dwg. 947C813
3.0 CORROSION BASED ON CORE SIZE AND FUEL ELEMENT GEOMETRY		
3.1 Avg. Carbon Reactivity Coefficient $\frac{g}{KG}$	3.90	3.90
3.2 Hot end diffusion loss for defect-free coating(7, 8) using corrosion rate rela- tionship in Fig. A-4.0-2		
3.2.1 Grams per element	14.6	14.6
3.2.2 Max. Incremental gm/inch	1.17	1.17
3.2.3 Reactivity Loss - \$	0.70	0.70
3.2.4 Relative bore deterioration at max. incremental gm/in. (item 3.2.2)		
3.2.4.1 Normalized grams carbon removed per unit bore area	1.0	1.0
3.2.4.2 Percent of carbon mass lost from nominal internal web	17.5	17.2
3.3 Assumed Mid-Band Loss ⁽⁹⁾		
3.3.1 Grams per element	10.0	10.0
3.3.2 Reactivity loss - \$	0.78	0.78
3.4 Relative interelement hydrogen leak- age based on geometry (11)	1.0	1.0

(4) Represents element across flats dimension divided by matrix thermal conductivity (at 4000°R) normalized to reference. This is a measure of the temperature gradient to conduct a fixed amount of heat per unit area across one element.

(See Next Page for Additional Footnotes.)

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TABLE A-4. 1-2 (CONTINUED)

Page 4 of 4

- (5) Assumes radial and axial maximum/average nuclear power peaking factors are not affected by changes in fuel and core geometry.
- (6) Relative thermal stress function defined by Appendix A-1, Section A-1.9.1.
- (7) For PEWEE-2 development coating profile on WNCO GEM-HED elements, as measured on 72 elements. See Figure A-4.0-3.
- (8) Defect-free hot end corrosion based on carbon diffusion only. Corrosion rate based on Figure A-4.0-2, independent of coating and matrix materials.
- (9) Mid-band corrosion proportional to bore surface area and fuel length with a ten(10) gram loss assumed for the reference element. Possible differences in mid-band performance for different materials are not considered.
- (10) Assuming interelement gap thickness constant for all alternates.

W59932

STANDARD GEOMETRY GRAPHITE ELEMENTS

Alternate 1, Drawing 947C812

Alternate 2, Drawing 947C813

PF = 1.1

Channel Diameter = 0.0950 inch

Figure A-4. 1-1. (CRD) Temperature Versus Axial Station for Graphite Elements (U)

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(CRD) The brazed joint temperature is not calculated by MCAP, but can be interpolated from the fuel element temperatures calculated adjacent to the brazed joint location.

Figure A-4.1-2 shows the temperatures at the aft end of the fuel element on an expanded scale. Analytical calculations for the axial conduction from fueled to unfueled material across the joint show that temperatures beyond 1/8 inch on either side of the joint are not affected. The nominal brazed joint temperature for a 1.1 power factor channel is 4361°R, and the 3 σ value is 4709°R based on uncertainties after incorporating results of one power test into predictions.

(U) Neglecting effects of control drum rotation and conduction from element-to-element, the estimated 3 σ element-to-element temperature difference is 106°R at the core periphery. The maximum temperature difference between the fuel hot spot at the center of three adjacent channels and the channel wall is 173°R. As a measure of the temperature gradient to conduct a given heat flux radially across the element, the element across flats dimension was divided by the matrix conductivity at 4000°R. This value in units of °R/BTU/sec-inch², was then normalized with the values for the reference element, Alternate 2. The thermal conductivity of GLFI fueled graphite has not been reported, but is expected to be nearly the same as for AXM having the same loading. Therefore, the normalized relative temperature gradient per unit heat flux is 1.0 for Alternate 1.

Temperature Limits

(CRD) A summary of current graphite fuel temperature technology is given on Table A-4.1-1. For the 75 gm NbC, ZrC coating on this element, the coating melting temperature is estimated to be higher than the 2400°C (4810°R) temperature reported* by Y-12, since the standard A6.07 electrical corrosion test routinely reaches channel wall temperatures near 4800°R.

(CRD) Although not considered a temperature limitation, migration of uranium through

*Y-DA-3194, "Melting Points of Coatings of NERVA Fuel Elements", Dec. 10, 1969.

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Standard Geometry Graphite Elements,
Alternate 1, Dwg. 947C812,
Alternate 2, Dwg. 947C813

PF = 1.1

Channel Dia. = 0.0950 Inch

Figure A-4. 1-2. (CRD) Axial Temperature Profile at Exit End of Fuel Element (U)

the pyrocarbon shell is estimated to occur in one hour at temperatures between 4630°R and 4920°R. At the lower temperature, only 2 percent of the beads were observed to migrate and at the higher level, 100 percent migrated.

(CRD) Experience with PW-2 development brazed joints indicates that cycle integrity is attainable at the 3rd temperature of 4709°R.

Maximum Volumetric Heating Rate

(U) The maximum volumetric heating rate for the element is 66.4 BTU/sec-inch³, occurring at fuel temperatures between 3000°R and 3300°R at core station 24 inches. The corresponding relative stress function (defined in Appendix A-1, Section A-1.9.1) for comparison of transverse stresses due to internal heating is 0.0714 BTU/sec-inch.

A-4.1.2 Corrosion and Reactivity Loss

A-4.1.2.1 Geometric Factor for Defect-Free Hot End Coating

(CRD) Assuming a defect-free coating at the hot end with the same profile used in PW-2, for 10 hours' operation, the maximum incremental hot end weight loss is 1.17 grams per inch. The total hot end diffusion weight loss is 14.6 grams per element. The normalized maximum weight of carbon removed per unit bore area is 1.0, identical to the reference. This represents 17.2 percent of the carbon in a nominal web. When 10 grams per element of mid-band and surface corrosion is assumed, the resulting total reactivity loss is 1.48\$. The comparison of corrosion and reactivity losses on a geometric basis is given in Table A-4.0-6.

A-4.1.2.2 Cyclic Performance - Influence of Coating Defects

(CRD) The bore corrosion resistance of the element is almost entirely dependent on the ability of the coating to maintain integrity during life, combined with the rate of carbon

[REDACTED]

diffusion through the intact coating. The latter process is exponentially dependent on temperature level and inversely proportional to coating thickness.⁽¹⁾ The coating integrity is lost as a result of two mechanisms;⁽²⁾ 1) differential thermal expansion between coating and matrix and 2) bore corrosion causing loss of coating adherence. Both mechanisms act concurrently and are interdependent. A satisfactory fuel element to last for ten hours with 60 thermal cycles must have bore coatings thick enough to limit carbon diffusion to acceptable levels, yet have high resistance to the formation and propagation of coating defects. These objectives must be achieved over the full length of the heated fuel element with a wide range of temperatures. The GEM NbC - HED ZrC coating on PW-2 development elements is an improvement for cyclic operation over the NRX-A6 vintage element but does not eliminate the propagation of defects. Evidence discussed in the following paragraphs based on electrically heated corrosion tests with ten-minute cycles at WANL is that corrosion rates for these elements increase after the first hour. This can be considered as evidence that the corrosion rate depends on the previous degree of corrosion (or loss of coating integrity). Expressed mathematically, this concept leads to an exponentially increasing corrosion rate with time. Differences between the electrically heated corrosion test and the reactor can alter the time constants in the exponential, but it is believed that the general trend will also apply to the reactor. Extrapolation of the electrical corrosion data leads to the conclusion that the graphite elements of present vintage may last two to five hours at R-1 conditions with an average weight loss between ten and thirty grams, taking the accelerating rate of defect corrosion into account.

(1) Other factors such as matrix permeability, stoichiometry of coating and gas, etc. are thought to play a secondary role.

(2) Other mechanisms such as hydrolysis are eliminated by environmental control.

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(CRD) Figure A-4. 1-3 illustrates recent electrical corrosion test data for individual elements tested for more than six ten-minute cycles. Weight losses in the first 60 minutes are to be compared with the weight loss during the second 60 minutes. These data have been averaged for several types of graphite elements as plotted in Figure A-4. 1-4. The intermediate expansion (GLFI) element exhibits poorer performance (based on the four samples) than the high expansion element. Moreover, these intermediate expansion elements when cycled in the electrical furnace beyond 60 minutes were prone to extensive pinholing. Also compared in Figure A-4. 1-4 is the performance of different types of coating, i. e., GEM-HED and GEM-GEM. The GEM-GEM coatings appear better but such a conclusion may be premature based on so few samples. The data from one low expansion (GLF) element tested for 17 x 10 cycles in the D1.01 test is also given on Figure A-4. 1-4. The element had a GEM-NbC and GEM-ZrC coating and shows remarkably good performance, both in regard to the initial corrosion rate and the acceleration in corrosion with cycles. Since the D1.01 test is at a lower temperature, the comparison with the other elements tested at A6.07 conditions is unfair on Figure A-4. 1-4. This same element was included in the normalized corrosion rate versus inverse temperature plot, Figure A-4. 0-5. In this figure, the initial corrosion rate for the GLF element is significantly higher than the reference line, but the indicated acceleration in corrosion rate is much less than might be expected on the basis of the data for the other elements. On the basis of this one element test, one may conclude that the formation of defects and consequent acceleration in corrosion rates is less rapid at lower temperatures. The axial weight loss distribution for the GLF element with GEM-GEM coating is given in Figure A-4. 1-5.

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(CRD) Figure A-4.1-3 Comparison of Graphite Fuel Hot End Weight Loss One Hour versus Extended Thermal Cycles (U)

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(CRD) Figure A-4.1-4 Comparison of Graphite Fuel Element Average Corrosion A6.07 and D1.01 Electrical Test Conditions (U)

(CRD) Figure A-4.1-5 Incremental Weight Loss Low Expansion Graphite Fuel with Extended Test Cycles (U)

[REDACTED]

A-4.2 (CRD) HIGH EXPANSION GRAPHITE ELEMENT, DWG. 947C813 (U)

A-4.2.0 (CRD) Introduction and Summary (U)

(CRD) The flow and thermal performance characteristics for this fuel element design are satisfactory for the R-1 reference conditions, and are identical to the values reported for Alternate 1, the intermediate expansion graphite element. The available evidence from electrical corrosion tests indicate that the graphite elements (both intermediate and high expansion) are suffering accelerated corrosion when cycled beyond 60 minutes. The probable cause of this increase in corrosion rate with cycles is the formation of coating defects. The overall corrosion performance is more strongly dependent on coating process than on the matrix. The prevention of element breakage for satisfactory corrosion performance is necessary. The projected corrosion performance for this alternate is basically the same as given for the intermediate expansion element in Section A-4.1. There is no foreseeable coating development to prevent the formation of some coating defects with cycles. This fuel element design is called the "reference element" and used as the basis for comparison with other alternates throughout this Appendix.

A-4.2.1 (U) Flow and Thermal Characteristics (U)

(U) All flow and thermal characteristics for this fuel element alternate are identical to those of the intermediate expansion element, Alternate 1. Refer to Section A-4.1.1 for a discussion.

(CRD) A-4.2.2 Corrosion and Reactivity Loss (U)

A-4.2.2.1 Geometry Effect for Defect-Free Hot End Coating

(CRD) Assuming a defect-free hot end coating and the other assumptions used in Section A-4.0.2, the corrosion and reactivity loss are almost identical to the intermediate expansion fuel element, Alternate 1. The only difference, caused by a small difference in carbon density, is that the normalized maximum weight of carbon removed per unit bore area, given in Table A-4.0-5, represents 17.5 percent of the nominal web rather than 17.2 percent.

[REDACTED]

A-4.2.2.2 Cyclic Performance - Influence of Coating Defects

(CRD) The discussion of cyclic performance for the intermediate expansion graphite element in Section A-4.1.2.2 is generally applicable to the high expansion graphite elements. The recent tests of development fuel elements reveal measurable improvements in both midband and hot-end corrosion performance when compared with NRX-A6 reactor and QC test elements. Figure A-4.0-4 shows average incremental bore corrosion weight losses for the NRX-A6 QC test elements compared with PW-2 development, GEM-HED coated high expansion graphite elements. This comparison is complicated by the fact that the development elements were tested for six 10-minute cycles while the NRX-A6 elements were tested for two 30-minute cycles. As explained more fully in Section A-4.0.1, the improvement can be attributed to the thicker coatings used on the PW-2 development elements (see Figure A-4.0-3) which are capable of remaining defect-free for the six cycles.

(CRD) When the same PW-2 development elements are tested for more than six cycles, present indications are that accelerated corrosion caused by defects is taking place. Figures A-4.1-3 and A-4.1-4 both illustrate the increase in weight loss for the second 60 minutes of testing versus the first 60 minutes. The single high expansion element with GEM NbC-GEM ZrC coating which was tested beyond 60 minutes (see Figure A-4.1-4) gives some hope that further improvements in coating are possible, but the break in the curve at 60 minutes shows an accelerated corrosion rate. Since the goal is ten hour's operation, it is obvious that any significant corrosion acceleration after one hour at the rates shown is intolerable. Further, although no test data is available, there is every reason to expect that the corrosion rate would continue to increase after the second hour.

A.4-3(CRD) HYBRID FUEL ELEMENT, DWG. 947C810 (U)

(CRD) Thermal and flow performance as well as the geometric factor for defect-free

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hot end corrosion are nearly identical to the other alternates with standard geometry in the reference core. No corrosion test experience is available although LASL has planned to use some hybrid elements as experiments in PW-2. A comparison with the reference high expansion graphite element (Alternate 2) is given in Table A-4.3-1.

(CRD) A-4.3.1 Flow and Thermal Characteristics (U)

A-4.3.1.1 Flow

(U) Identical to Alternate 1, the Intermediate Expansion Graphite Element; the High Expansion Graphite Element, Alternate 2; and the All Composite Element, Alternate 5.

A-4.3.1.2 Thermal

(U) Very similar to Alternates 1, 2, and 5. The difference in thermal conductivity causes about 20°R step up in fuel matrix (hot spot) temperature at Station 35 inches. This does not present a problem. The axial temperature profile is shown in Figure A-4.3-1.

(CRD) The channel wall temperatures are not affected by a change in matrix. The brazed joint at Station 35 inches is at a nominal temperature of 4100°R, considered to be within capabilities of brazing technology. A discussion of maximum temperatures and temperature limits given in Section A-4.5.1 is pertinent.

(CRD) A-4.3.2 Corrosion and Reactivity Loss (U)

A-4.3.2.1 Geometric Factor for Defect-Free Hot End Corrosion

(CRD) The relative corrosion and reactivity loss for defect-free hot end coating is nearly identical to that of the all-composite element (Alternate 5).

A-4.3.2.2 Cyclic Performance Capability - Influence of Coating Defects

(CRD) Mid-band corrosion performance for the graphite section offers potential improvement over the current all-composite element. However, no cyclic

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TABLE A-4.3-1

Page 1 of 4

(CRD) COMPARISON OF FLOW, THERMAL, AND CORROSION PERFORMANCE FOR
19 HOLE HYBRID FUEL ELEMENT, ALTERNATE 3 (U)

	Alternate Dwg. 947C810	Reference Alternate 2 (Dwg. 947C813)
0. DESCRIPTION	35 inch AXM	
0.1 Matrix Material	17 inch composite	AXM
0.2 Coating Material	ZrC	75 gm Nb, ZrC
0.3 Reamed Diameter - Inch	0.102	0.102
0.4 Hole Pitch - Inch	0.173	0.173
0.5 Across Flats - Inch	0.753	0.753
0.6 Void Fraction	0.314	0.314
0.7 Core Length - Inch	52	52
0.8 Core Fueled Diameter - Inch	37.1	37.1
0.9 Number of Channels Per Element	19	19
0.10 Number of Channels Per Core	35682	35682
1. FLOW		
1.1 Equivalent Impedance Diameter-in.		
1.1.1 Min. Based on PW-2 Avg. Coating (2.18 mils)	0.0935	0.0935
1.1.2 Min. Based on Experimental PW-2 Thick Hot End Coating (Dwg. 939J709-G01)	0.0914	0.0914
1.2 Core Pressure Drop ⁽¹⁾ - Psi		
1.2.1 Based on diameter in 1.1.1	162	162
1.2.2 Based on diameter in 1.1.2	176	176

(1) Based on 1.1 power factor, except Alternate 6 (single channel - carbide), which is based on 1.0 power factor. Neglects taper in bore due to varying coating thickness.

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TABLE A-4.3-1 (CONTINUED)

Page 2 of 4

	Alternate Dwg. 947C810	Reference Alternate 2 Dwg. 947C813
2. THERMAL		
2.1 (T_W) Max. Channel Wall Temp ⁽²⁾ °R		
2.1.1 Nominal	4389	4389
2.1.2 Hot Channel ⁽³⁾ (3 σ)	4737	4737
2.1.3 Difference from reference	0	—
2.1.4 Axial Station - Inch	48	48
2.2 (T_F) Max. Fuel Hot Spot Temp ⁽²⁾ °R		
2.1.1 Nominal	4473	4463
2.1.2 Hot Channel ⁽³⁾ (3 σ)	4822	4811
2.1.3 Difference from reference	+11	—
2.1.4 Axial Station - Inch	46	46
2.3 Temperature Differences		
2.3.1 Max. ΔT ⁽²⁾ within nominal Element $(T_F - T_W)$ - °R	177	173
2.3.2 Statistical uncertainty in element-to-element average exit temp. ⁽³⁾ (3 σ) - °R	106	106
2.3.3 Normalized temp. gradient per unit heat flux ⁽⁴⁾	1.33	1.0
2.4 Max. Volumetric Heating Rate ⁽⁵⁾ Btu/sec-inch ³	66.4	66.4
2.5 Max. Thermal Stress Function ^(5,6) Q - Btu/sec-inch	0.0714	0.0714

(2) Isolated channel calculation by MCAP for equilateral triangular array of coolant holes and 1.1 power factor (except alternate 6, single channel - carbide).

(3) Represents value for core periphery after one power test, neglecting control drum effects, but including $\pm 63^\circ\text{R}$ uncertainty in mean fuel exit gas temperature.

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TABLE A-4.3-1 (CONTINUED)

Page 3 of 4

	Alternate Dwg. 947C 810	Reference Alternate 2 Dwg. 947C813
3.0 CORROSION BASED ON CORE SIZE AND FUEL ELEMENT GEOMETRY		
3.1 Avg. Carbon Reactivity Coefficient $\frac{\text{g}}{\text{KG}}$	3.99	3.90
3.2 Hot end diffusion loss for defect-free coating(7, 8) using corrosion rate relationship in Fig. A-4.0-2		
3.2.1 Grams per element	14.6	14.6
3.2.2 Max. Incremental gm/inch	1.17	1.17
3.2.3 Reactivity Loss - \$	0.76	0.70
3.2.4 Relative bore deterioration at max. incremental gm/in. (item 3.2.2)		
3.2.4.1 Normalized grams carbon removed per unit bore area	1.0	1.0
3.2.4.2 Percent of carbon mass lost from nominal internal web	19.7	17.2
3.3 Assumed Mid-Band Loss⁽⁹⁾		
3.3.1 Grams per element	10	10.0
3.3.2 Reactivity loss - \$	0.78	0.78
3.4 Relative interelement hydrogen leakage based on geometry (11)	1.0	1.0

(4) Represents element across flats dimension divided by matrix thermal conductivity (at 4000°R) normalized to reference. This is a measure of the temperature gradient to conduct a fixed amount of heat per unit area across one element.

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TABLE A-4.3-1 (CONTINUED)

Page 4 of 4

- (5) Assumes radial and axial maximum/average nuclear power peaking factors are not affected by changes in fuel and core geometry.
- (6) Relative thermal stress function defined by Appendix A-1, Section A-1.9.1.
- (7) For PEWEE-2 development coating profile on WNCO GEM-HED elements, as measured on 72 elements. See Figure A-4.0-3.
- (8) Defect-free hot end corrosion based on carbon diffusion only. Corrosion rate based on Figure A-4.0-2, independent of coating and matrix materials.
- (9) Mid-band corrosion proportional to bore surface area and fuel length with a ten(10) gram loss assumed for the reference element. Possible differences in mid-band performance for different materials are not considered.
- (10) Assuming interelement gap thickness constant for all alternates.

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Figure A-4.3-1. (CRD) Temperature Versus Axial Station for 19 Hole Hybrid Element (U)

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for more than ~2 hours are available. Special coating and/or corrosion problems may exist at the joint. The possibility exists to coat each section individually to more closely match the coating temperature to the axial temperature profile in the reactor.

(CRD) The consequences of a joint failure are more serious than for a joint at the aft end. However, the joint itself could possibly be more reliable than a matrix against cracking. A close match of the CTE's between sections is desirable to minimize interstitial gaps.

(U) The discussion regarding composite element corrosion given in Section A-4.5.2.2 is pertinent.

A-4.4.(CRD)³⁷-HOLE HYBRID FUEL ELEMENT, DWG. 947C816 (U)

A-4.4.0 (CRD) Introduction and Summary (U)

(CRD) This alternate was selected initially on the basis of simplified parametric ratios in Appendix A-1. More exact calculations using MCAP verify that the flow and thermal performance are satisfactory to meet R-1 requirements. The relative geometric factor for bore corrosion indicates approximately a 40 percent increase over the standard geometry. There is no corrosion test data available and no manufacturing history to indicate possible effects of geometry changes on cyclic performance or basic material properties. A comparison with the reference high expansion graphite element (Alternate 2) is given in Table A-4.4-1.

A-4.4.1 (CRD) Flow and Thermal Characteristics (U)

A-4.4.1.1 Flow

(CRD) Assuming the minimum effective impedance diameter is 0.0755 inches based on subtracting twice the coating thickness plus an additional 4.1 mils from the nominal reamed diameter, the core ΔP is 161 psi. For the experimental thick hot end coating defined by Drawing 939J709 for PW-2, the core ΔP is 177 psi. This high void element does not allow sufficient margin to add substantially thicker bore coatings, and the

(CRD) COMPARISON OF FLOW, THERMAL, AND CORROSION PERFORMANCE FOR
37 HOLE HYBRID ELEMENT, ALTERNATE 4 (U)

	Alternate Dwg. 947C816	Reference Alternate 2 (Dwg. 947C813)
0. DESCRIPTION		
0.1 Matrix Material	35 in. AXM 17 in. composite	AXM
0.2 Coating Material	ZrC	75 gm Nb, ZrC
0.3 Reamed Diameter - Inch	0.084	0.102
0.4 Hole Pitch - Inch	0.1309	0.173
0.5 Across Flats - Inch	0.799	0.753
0.6 Void Fraction	0.373	0.314
0.7 Core Length - Inch	52	52
0.8 Core Fueled Diameter - Inch	37.1	37.1
0.9 Number of Channels Per Element	37	19
0.10 Number of Channels Per Core	61050	35682
1. FLOW		
1.1 Equivalent Impedance Diameter-in.		
1.1.1 Min. Based on PW-2 Avg. Coating (2.18 mils)	0.0755	0.0935
1.1.2 Min. Based on Experimental PW-2 Thick Hot End Coating (Dwg. 939J709-G01)	0.0734	0.0914
1.2 Core Pressure Drop ⁽¹⁾ - Psi		
1.2.1 Based on diameter in 1.1.1	161	162
1.2.2 Based on diameter in 1.1.2	177	176

(1) Based on 1.1 power factor, except Alternate 6 (single channel - carbide), which is based on 1.0 power factor. Neglects taper in bore due to varying coating thickness.

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TABLE A-4. 4-1 (CONTINUED)

Page 2 of 4

	Alternate Dwg. 947C816	Reference Alternate 2 Dwg. 947C813
2. THERMAL		
2.1 (T_W) Max. Channel Wall Temp ⁽²⁾ °R		
2.1.1 Nominal	4353	4389
2.1.2 Hot Channel ⁽³⁾ (3 σ)	4701	4737
2.1.3 Difference from reference	-36	---
2.1.4 Axial Station - Inch	50	48
2.2 (T_F) Max. Fuel Hot Spot Temp ⁽²⁾ °R		
2.2.1 Nominal	4378	4463
2.2.2 Hot Channel ⁽³⁾ (3 σ)	4726	4811
2.2.3 Difference from reference	-85	---
2.2.4 Axial Station - Inch	50	46
2.3 Temperature Differences		
2.3.1 Max. ΔT ⁽²⁾ within nominal Element $(T_F - T_W)$ - °R	88	173
2.3.2 Statistical uncertainty in element-to-element average exit temp. ⁽³⁾ (3 σ) - °R	106	106
2.3.3 Normalized temp. gradient per unit heat flux ⁽⁴⁾	1.42	1.0
2.4 Max. Volumetric Heating Rate ⁽⁵⁾ Btu/sec-inch ³	72.5	66.4
2.5 Max. Thermal Stress Function ^(5,6) ° - Btu/sec-inch	0.037	0.0714

(2) Isolated channel calculation by MCAP for equilateral triangular array of coolant holes and 1.1 power factor (except alternate 6, single channel - carbide).

(3) Represents value for core periphery after one power test, neglecting control drum effects, but including $\pm 63^\circ\text{R}$ uncertainty in mean fuel exit gas temperature.

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TABLE A-4. 4-1 (CONTINUED)

Page 3 of 4

Alternate Dwg.
947C816

Reference Alternate 2
Dwg. 947C813

3.0 CORROSION BASED ON CORE SIZE AND
FUEL ELEMENT GEOMETRY

3.1	Avg. Carbon Reactivity Coefficient $\frac{\text{g}}{\text{KG}}$	4.28	3.90
3.2	Hot end diffusion loss for defect-free coating(7, 8) using corrosion rate relationship in Fig. A-4.0-2		
3.2.1	Grams per element	20.6	14.6
3.2.2	Max. Incremental gm/inch	1.66	1.17
3.2.3	Reactivity Loss - \$	1.02	0.70
3.2.4	Relative bore deterioration at max. incremental gm/in. (item 3.2.2)		
3.2.4.1	Normalized grams carbon removed per unit bore area	0.87	1.0
3.2.4.2	Percent of carbon mass lost from nominal internal web	26.0	17.2
3.3	Assumed Mid-Band Loss ⁽⁹⁾		
3.3.1	Grams per element	16.0	10.0
3.3.2	Reactivity loss - \$	1.17	0.78
3.4	Relative interelement hydrogen leakage based on geometry (11)	0.94	1.0

(4) Represents element across flats dimension divided by matrix thermal conductivity (at 4000°R) normalized to reference. This is a measure of the temperature gradient to conduct a fixed amount of heat per unit area across one element.

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TABLE A-4. 4-1 (CONTINUED)

Page 4 of 4

- (5) Assumes radial and axial maximum/average nuclear power peaking factors are not affected by changes in fuel and core geometry.
- (6) Relative thermal stress function defined by Appendix A-1, Section A-1.9.1.
- (7) For PEWEE-2 development coating profile on WNCO GEM-HED elements, as measured on 72 elements. See Figure A-4.0-3.
- (8) Defect-free hot end corrosion based on carbon diffusion only. Corrosion rate based on Figure A-4.0-2, independent of coating and matrix materials.
- (9) Mid-band corrosion proportional to bore surface area and fuel length with a ten(10) gram loss assumed for the reference element. Possible differences in mid-band performance for different materials are not considered.
- (10) Assuming interelement gap thickness constant for all alternates.

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potential effect of coating roughness is greater than for standard element geometry.

(CRD) A-4.4.1.2 Thermal (U)

(CRD) Because of the finer mesh geometry, the nominal maximum channel wall temperature is reduced to 4353°R from 4389°R for the reference element. The maximum fuel hot spot temperature is 4378°R versus 4463°R for the reference element. The maximum fuel hot spot-to-channel wall ΔT is 88°R . The normalized temperature gradient per unit heat flux across the element is 1.42, indicating the highest sensitivity to radial power gradients of any alternate. The brazed joint temperature at Station 35 inches is approximately 4000°R , considered within brazing technology capability. The discussion given in Section A-4.5.1 regarding composite material temperatures and limits is pertinent. The axial temperature profile is shown in Figure A-4.4-1.

(CRD) The maximum volumetric heat generation rate is increased from the reference design to $72.5 \text{ BTU/sec-in}^3$. Because of the change in geometry, the stress function $Q\Psi$ is reduced to $0.037 \text{ BTU/sec-inch}^2$, the lowest of any alternate. A net reduction in thermal stress due to heat generation would occur only if there is no significant change in basic matrix properties in going from a 19 to a 37 hole geometry.

(U) The estimated 37 element-to-element ΔT is the same as for the reference element, 106°R at the core periphery.

A-4.4-2 (CRD) Corrosion and Reactivity Loss (U)

(CRD) A-4.4-2.1 Geometric Factor for Defect-Free Hot End Corrosion

(CRD) The maximum grams per inch increases from 1.17 to 1.66 with the normalized weight of carbon per unit of bore area being reduced from 1.0 to 0.87. This indicates relatively less hot end coating deterioration for the same total weight loss. However, the 1.66 grams per inch represents 26 percent of the nominal web (versus 17.2 percent for the standard high expansion element) indicating greater tendency to pinhole.

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Figure A-4. 4-1. (CRD) Temperature Versus Axial Station for 37 Hole Hybrid Element (U)

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A-4.4-2.2 (U) Cyclic Performance Capability - Influence of Coating Defects (U)

(U) There is no available assessment on the possible cyclic performance of this alternate. Assuming there are no effects due to geometry changes, the discussions given under A-4.3.2.2 for the 19-hole hybrid element and A-4.5.2.2 for the composite element are pertinent.

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A-4.5 (CRD) ALL COMPOSITE ELEMENT, DWG 947C809 (U)

A-4.5.0 (CRD) Introduction and Summary (U)

(CRD) The thermal and flow characteristics of this fuel element alternate are almost identical to those for the other fuel element alternates with "standard geometry," (Alternates 1, 2, and 3). The corrosion weight loss based on element geometry if the hot end coating remains defect-free for the 10 hours' operation is also identical. Because of a slightly higher reactivity worth of carbon, this element has a higher reactivity loss with the same carbon mass loss.

Detailed comparisons of the flow, thermal, corrosion, and reactivity performance characteristics are given in Table A-4.5-1 for comparison with the reference high expansion graphite element, (Alternate 2).

(CRD) The composite fuel element can be considered less highly developed than the graphite matrix elements at the present time. They have not had the benefit of years of independent coating and matrix experimentation and testing at different laboratories plus reactor test experience. All manufacturing and electrical corrosion testing to date has been carried out at LASL. Although corrosion rates appear to increase with cycles, several different types of evidence indicate that the composite matrix is more resistant to defect corrosion than the graphite matrix. The present 30 v/o composite elements show significant mid-band corrosion due to coating cracks and this problem must be solved; however, the long-range outlook is that they have better potential for meeting 10-hour requirements than the graphite matrices. A comparison with the reference high expansion graphite element is given in Table A-4.5-1.

A-4.5.1 (CRD) Flow and Thermal Performance (U)

A-4.5.1.1 Flow

(U) The core pressure drop for the composite element is identical to that of the other alternates with standard 19-hole geometry in the reference core size.

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TABLE A-4.5-1

Page 1 of 4

(CRD) COMPARISON OF FLOW, THERMAL, AND CORROSION PERFORMANCE FOR
STANDARD GEOMETRY COMPOSITE ELEMENT, ALTERNATE 5 (U)

	Alternate Dwg. 947C809	Reference Alternate 2 (Dwg. 947C813)
0. DESCRIPTION		
0.1 Matrix Material	Composite	AXM
0.2 Coating Material	ZrC	75 gm Nb, ZrC
0.3 Reamed Diameter - Inch	0.102	0.102
0.4 Hole Pitch - Inch	0.173	0.173
0.5 Across Flats - Inch	0.753	0.753
0.6 Void Fraction	0.314	0.314
0.7 Core Length - Inch	52	52
0.8 Core Fueled Diameter - Inch	37.1	37.1
0.9 Number of Channels Per Element	19	19
0.10 Number of Channels Per Core	35682	35682
1. FLOW		
1.1 Equivalent Impedance Diameter-in.		
1.1.1 Min. Based on PW-2 Avg. Coating (2.18 mils)	0.0935	0.0935
1.1.2 Min. Based on Experimental PW-2 Thick Hot End Coating (Dwg. 939J709-G01)	0.0914	0.0914
1.2 Core Pressure Drop ⁽¹⁾ - Psi		
1.2.1 Based on diameter in 1.1.1	162	162
1.2.2 Based on diameter in 1.1.2	176	176

(1) Based on 1.1 power factor, except Alternate 6 (single channel - carbide), which is based on 1.0 power factor. Neglects taper in bore due to varying coating thickness.

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	Alternate Dwg. 947C809	Reference Alternate 2 Dwg. 947C813
2. THERMAL		
2.1 (T_W) Max. Channel Wall Temp ⁽²⁾ °R		
2.1.1 Nominal	4389	4389
2.1.2 Hot Channel ⁽³⁾ (3 σ)	4737	4737
2.1.3 Difference from reference	0	—
2.1.4 Axial Station - Inch	48	48
2.2 (T_F) Max. Fuel Hot Spot Temp ⁽²⁾ °R		
2.1.1 Nominal	4473	4463
2.1.2 Hot Channel ⁽³⁾ (3 σ)	4822	4811
2.1.3 Difference from reference	+11	—
2.1.4 Axial Station - Inch	46	46
2.3 Temperature Differences		
2.3.1 Max. ΔT ⁽²⁾ within nominal Element $(T_F - T_W)$ - °R	177	173
2.3.2 Statistical uncertainty in element-to-element average exit temp. ⁽³⁾ (3 σ) - °R	106	106
2.3.3 Normalized temp. gradient per unit heat flux ⁽⁴⁾	1.33	1.0
2.4 Max. Volumetric Heating Rate ⁽⁵⁾ Btu/sec-inch ³	66.4	66.4
2.5 Max. Thermal Stress Function ^(5,6) Q - Btu/sec-inch	0.0714	0.0714

(2) Isolated channel calculation by MCAP for equilateral triangular array of coolant holes and 1.1 power factor (except alternate 6, single channel - carbide).

(3) Represents value for core periphery after one power test, neglecting control drum effects, but including $\pm 63^\circ\text{R}$ uncertainty in mean fuel exit gas temperature.

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TABLE A-4.5-1 (CONTINUED)

Page 3 of 4

	Alternate Dwg. 947C809	Reference Alternate 2 Dwg. 947C813
3.0 CORROSION BASED ON CORE SIZE AND FUEL ELEMENT GEOMETRY		
3.1 Avg. Carbon Reactivity Coefficient $\frac{\text{g}}{\text{kg}}$	4.30	3.90
3.2 Hot end diffusion loss for defect-free coating(7, 8) using corrosion rate rela- tionship in Fig. A-4.0-2		
3.2.1 Grams per element	14.6	14.6
3.2.2 Max. Incremental gm/inch	1.17	1.17
3.2.3 Reactivity Loss - \$	0.78	0.70
3.2.4 Relative bore deterioration at max. incremental gm/in. (item 3.2.2)		
3.2.4.1 Normalized grams carbon removed per unit bore area	1.0	1.0
3.2.4.2 Percent of carbon mass lost from nominal internal web	19.7	17.2
3.3 Assumed Mid-Band Loss ⁽⁹⁾		
3.3.1 Grams per element	10.0	10.0
3.3.2 Reactivity loss - \$	0.85	0.78
3.4 Relative interelement hydrogen leak- age based on geometry (11)	1.0	1.0

(4) Represents element across flats dimension divided by matrix thermal conductivity (at 4000°R) normalized to reference. This is a measure of the temperature gradient to conduct a fixed amount of heat per unit area across one element.

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TABLE A-4.5-1 (CONTINUED)

- (5) Assumes radial and axial maximum/average nuclear power peaking factors are not affected by changes in fuel and core geometry.
- (6) Relative thermal stress function defined by Appendix A-1, Section A-1.9.1.
- (7) For PEWEE-2 development coating profile on WNCO GEM-HED elements, as measured on 72 elements. See Figure A-4.0-3.
- (8) Defect-free hot end corrosion based on carbon diffusion only. Corrosion rate based on Figure A-4.0-2, independent of coating and matrix materials.
- (9) Mid-band corrosion proportional to bore surface area and fuel length with a ten(10) gram loss assumed for the reference element. Possible differences in mid-band performance for different materials are not considered.
- (10) Assuming interelement gap thickness constant for all alternates.

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(CRD) A-4.5.1.2 Thermal (U)

(CRD) The axial temperature profile for the composite element is given in Figure A-4.5-1. Thermal performance is summarized in Table A-4.0-2 for comparison with other alternates. The only difference in maximum temperature levels from Alternates 1 and 2 is a slightly higher maximum matrix temperature due to the lower thermal conductivity (see Appendix A-1, Figure A-1.4-19) of the 30 v/o composite. Because the thermal conductivity is strongly affected by the percent of theoretical density in the matrix, these conductivity values and maximum temperature levels can shift slightly with changes in the manufacturing process. Matrix temperatures are higher than for graphite elements all along the element and are more strongly affected near the mid-band region than at the aft end. Channel wall temperatures are the same as for Alternates 1, 2, and 3.

(CRD) The temperature for the first appearance of the liquid phase in UC-ZrC-C composites was plotted as a function of fuel loading for different compositions and presented recently by LASL (see Figure A-4.5-2). The curves were based on experimental data up to 500 mg/cc loading and are extrapolated beyond this loading. For 700 mg/cc, the 30 v/o composite matrix has a melting point limit at 2600°C (5170°R) versus 4822°R for the 3 σ fuel matrix temperature. No limit has been established for the ZrC coating on composite elements, but preliminary data from Y-12* indicates a limit may exist for ZrC coating on 630 mg/cc fueled graphite at 2400°C (4810°R). If confirmed for ZrC coating on 30 v/o composite, the 4810°R can be compared with the 3 σ channel wall temperature after one power test of 4737°R .

(U) The estimated 3 σ uncertainty in an average element exit gas temperature is 106°R , the same as for all other alternates except the UC-ZrC carbide. The normalized relative temperature gradient across the element per unit of heat flux is 1.33, indicating greater sensitivity than the graphite element to imposed transverse variations in heat generation.

*See Table A-4.1-1.

(CRD) Figure A-4.5-1 Temperature Versus Axial Station for 30 v/o Composite Element (CRD)

Figure A-4.5-2. (CRD) First Appearance of Liquid Phase: UC-ZrC-C (U)

A-4.5.2 (CRD) Corrosion and Reactivity Loss (U)

A-4.5.2.1 (CRD) Geometric Factor for Defect Free Hot End Coating (U)

(U) Assuming the coating remains defect free during operation, the geometric factors affecting bore and surface corrosion are the same as for the standard element geometry.

A.4.5.2.2 (CRD) Cyclic Performance - Influence of Coating Defects (U)

(CRD) Because of differences in the LASL corrosion test facility, test conditions and instrumentation, a good comparison of LASL composite and WANL graphite test results cannot be made. However, two composite elements have shown long-time capability in the furnace which has not been attempted using graphite elements. These 10-hour composite element tests cannot be considered conclusive for many reasons, listed below:

1. One of the tests was single cycle, 1×600 .
2. The temperature was relatively low, $\sim 4000^{\circ}\text{R}$ at the Station 45-inch drilled hole--corresponding to about 3900°R exit gas temperature. Evidence discussed in Section A-4.1.2 suggests that not only the carbon diffusion rate, but the coating defect formation rate are strongly affected by temperature level.
3. Both elements were from an unusually good performing coating batch, B6-286. We do not know whether or not the best graphite elements would perform equally well under the same conditions.
4. The majority of the composite elements tested to date exhibit a severe mid-band corrosion problem which, for unknown reasons, was not present in the two ten-hour tests. Figure A-4.5-3 shows the axial weight loss profiles for the two elements.

(CRD) Figure A-4.5-3 Incremental Weight Loss Comparison
LASL Composite Element Ten Hour Electric Tests
1 x 600 and 60 x 10 (U)

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5. The tests were modified as follows by removing the elements periodically from the furnace. Pin holes would develop at the hot end chuck causing loss of the helium atmosphere. At such time the element would be removed from the furnace and a piece cut off the hot end to allow the pinhole to be in the hot end chuck contact area. Since the drilled hole could not be seen the test was continued by controlling from power based on the power measurement before shut-down. This happened three (3) or four (4) times on the long tests. The shortening of the element changes the axial temperature distribution along the element.

(CRD) The following statements are quotations from LASL Biweekly Status Reports or else paraphrased quotations from meetings.

The following general observations have been noted and are quoted from LASL Biweekly Report Number 20 (N-1-2464):

"Midrange corrosion is still a function of the presence of coating cracks.

"Extended cyclic life tests are usually terminated due to breakage in the mid-range region. Midrange corrosion of about 0.5 gm/inch weakens the element. Local changes in current density due to the midrange corrosion probably causes the element to break by a combination of thermal and mechanical stresses.(1)

"Corrosion losses up to 1.0 gm/inch are very difficult to detect visually and corrosion defects are extremely difficult to see.

"The adherence between coating and matrix remains excellent after many test cycles.

From Trip Report 6284 on meeting at SNPO-C on Dec., 16, 1969:

"Significant coating cracking is occurring which penetrates into the matrix on the 30 v/o composite elements.

"Midband corrosion of the lower volume percent composites is significant, potentially due to this cracking, and must be solved to reach a 10-hour endurance.

(1) The behavior of three elements from coating batch B4-328 is an apparent contradiction to this observation. These elements had very poor coating adherence and high weight losses in 6x10 tests at 4600°R exit gas temperature. Maximum incremental weight losses were 2.17, 4.25, and 4.56 gm/inch at approximately station 40 inches for the three elements. All

[REDACTED]

"Higher volume percentages of carbide in the composite appear to help midband corrosion but hurt thermal stress capability which is already a major concern.

"Higher loadings show fewer midband cracks than lower loadings in the 30 v/o composite.

"One can consider reducing operating temperatures to accommodate corrosion while not increasing thermal stress problems.

"Eutectic temperatures of the UC-ZrC in the composites are a function of loading level and volume percentage of UC-ZrC in matrix. Low volume percentages limit loadings to lower levels for same eutectic temperature.

From LASL Biweekly Report No. 25, N-1-2522, Dec., 4, 1969:

"The factor of two greater weight loss in the 60x10 test than in the 1x600 test is probably caused by cycling. Earlier data had suggested that the hot end losses of ZrC composite elements were not noticeably sensitive to cyclic testing up to 12 ten-minute cycles. Additional cyclic testing of 20 to 50 cycles is required to learn more about the influence of cycling on hot end and midrange loss rates."

(CRD) Considering the relatively early stage of development in composite element production and coating, projections for their future capability are that the potential for 10 hours' operation is greater than for graphite elements, because:

1. Limited tests show qualitatively better corrosion resistance for bare, poorly coated, or intentionally damaged matrix surfaces for composites than for graphite.
2. Although mid-band corrosion is now a problem, the fact that it was also a problem for graphite elements at one time, gives reason to believe that a cure can also be found for composites.
3. The fact that the cyclic effects of accelerating corrosion did not become significant after 6 cycles in LASL tests, yet does become significant in WANL tests on graphite matrices. (However, differences in test conditions must still be

three elements completed the 6x10 test successfully and the two with highest losses parted at the weakened section after the test. Graphite elements in the WANL electrical test normally do not survive if incremental weight losses exceed 1.0 gm/inch. The two most severely corroded elements (peak incremental losses of 4.25 and 4.56 gm/inch) were tested with Helium under pressure of -32 psi and -17 psi at station 20.

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considered.)

4. The general appearance and coating adherence of the slit open elements after testing. A low number of corrosion pockets, pinholes, and characteristic ring type corrosion even on poorly coated specimens. (Again, differences in test conditions must be considered.)

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A-4.6 (CRD) CARBIDE FUEL ELEMENT (DWG 947C817) (U)

A-4.6.0 (CRD) Introduction and Summary

(CRD) The thermal, hydraulic and corrosion performance of this alternate are acceptable provided the small segments can be assembled and retained in position adequately. None of these elements have been made or tested, but the assumption has been made that zero corrosion and reactivity loss would occur. A comparison with the high expansion graphite element (Alternate 2) is given in Table A-4.6-1.

A-4.6.1 (CRD) Flow and Thermal (U)

A-4.6.1.1 Flow

(U) Because of the high void design and absence of coating, the core pressure drop is very low, about 50 psi. All channels are assumed to generate the same average power because of the potential for very fine loading graduations - both radially and axially.

(U) The "spaghetti" concept for the solid solution UC-ZrC element represents a radical departure from previous reactor design experience. A complete analysis and design effort similar to that which has been performed in developing the thermal and fluid flow performance of NRX type fuel would be required on the carbide fuel.

(CRD) The spaghetti fuel concept offers both advantages and potential problems from the flow standpoint. The principal advantage is that the fine graduations in uranium loading (both axially and radially) eliminates the need for flow channel orificing. Also, the consequences of radial flow redistribution due to element cracks and breaks are minimized.

(CRD) On the other hand, one must look carefully at the possibility of flow channel misalignment at the mating surfaces of the spaghetti elements. The resulting step changes in flow area would restrict flow in an unpredictable manner and result in an additional hot-channel factor.

TABLE A-4.6-1

Page 1 of 4

(CRD)COMPARISON OF FLOW, THERMAL, AND CORROSION PERFORMANCE FOR
SINGLE CHANNEL CARBIDE ELEMENT, ALTERNATE 6(U)

	Alternate Dwg. 947C 817	Reference Alternate 2 (Dwg. 947C813)
0. DESCRIPTION		
0.1 Matrix Material	ZrC - UC	AXM
0.2 Coating Material	None	75 gm Nb, ZrC
0.3 Reamed Diameter - Inch	0.120	0.102
0.4 Hole Pitch - Inch	0.170	0.173
0.5 Across Flats - Inch	0.170	0.753
0.6 Void Fraction	0.453	0.314
0.7 Core Length - Inch	52	52
0.8 Core Fueled Diameter - Inch	37.1	37.1
0.9 Number of Channels Per Element	1	19
0.10 Number of Channels Per Core	36843	35682
1. FLOW		
1.1 Equivalent Impedance Diameter-in.		
1.1.1 Min.	0.120	0.0935
1.1.2 Min. Based on Experimental PW-2 Thick Hot End Coating (Dwg. 939J709-G01)	(No coating) NA	0.0914
1.2 Core Pressure Drop ⁽¹⁾ - Psi		
1.2.1 Based on diameter in 1.1.1	~ 50	162
1.2.2 Based on diameter in 1.1.2	NA	176

(1) Based on 1.1 power factor, except Alternate 6 (single channel - carbide), which is based on 1.0 power factor. Neglects taper in bore due to varying coating thickness.

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Page 2 of 4

TABLE A-4.6-1 (Continued)

		Alternate Dwg. 947C 817	Reference Alternate 2 Dwg. 947C813
2.	THERMAL (12)		
2.1	(T_W) Max. Channel Wall Temp ⁽²⁾ °R		
2.1.1	Nominal	4417	4389
2.1.2	Hot Channel ⁽³⁾ (3 σ)	4765	4737
2.1.3	Difference from reference	+ 28	—
2.1.4	Axial Station - Inch	46 - 48	48
2.2	(T_F) Max. Fuel Hot Spot Temp ⁽²⁾ °R		
2.2.1	Nominal	4449 ⁽¹³⁾	4463
2.2.2	Hot Channel ⁽³⁾ (3 σ)	4797 ⁽¹³⁾	4811
2.2.3	Difference from reference	-14	—
2.2.4	Axial Station - Inch	46	46
2.3	Temperature Differences		
2.3.1	Max. ΔT ⁽²⁾ within nominal Element ($T_F - T_W$) - °R	94	173
2.3.2	Statistical uncertainty in element-to-element average exit temp. ⁽³⁾ (3 σ) - °R	~ 100	106
2.3.3	Normalized temp. gradient per unit heat flux ⁽⁴⁾	0.30	1.0
2.4	Max. Volumetric Heating Rate ⁽⁵⁾ Btu/sec-inch ³	75.3	66.4
2.5	Max. Thermal Stress Function ^(5, 6) Q - Btu/sec-inch	0.0447	0.0714

(2) Isolated channel calculation by MCAP for equilateral triangular array of coolant holes and 1.1 power factor (except alternate 6, single channel - carbide).

(3) Represents value for core periphery after one power test, neglecting control drum effects, but including $\pm 63^\circ\text{R}$ uncertainty in mean fuel exit gas temperature.

(See Next Page for Additional Footnotes.)

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(CRD) TABLE A-4.6-1 (Continued)

Page 3 of 4

Alternate Dwg.
947C817

Reference Alternate 2
Dwg. 947C813

3.0 CORROSION BASED ON CORE SIZE AND
FUEL ELEMENT GEOMETRY

3.1	Avg. Carbon Reactivity Coefficient $\frac{g}{kg}$	5.24	3.90
3.2	Hot end diffusion loss for defect-free coating(7, 8) using corrosion rate relationship in Fig. A-4.0-2		
3.2.1	Grams per element	NA	14.6
3.2.2	Max. Incremental gm/inch	NA	1.17
3.2.3	Reactivity Loss - \$	NA	0.70
3.2.4	Relative bore deterioration at max. incremental gm/in. (item 3.2.2)		
3.2.4.1	Normalized grams carbon removed per unit bore area	NA	1.0
3.2.4.2	Percent of carbon mass lost from nominal internal web	NA	17.2
3.3	Assumed Mid-Band Loss ⁽⁹⁾		
3.3.1	Grams per element	NA	10.0
3.3.2	Reactivity loss - \$	NA	0.78
3.4	Relative interelement hydrogen leakage based on geometry (11)	1.16	1.0

(4) Represents element across flats dimension divided by matrix thermal conductivity (at 4000°R) normalized to reference. This is a measure of the temperature gradient to conduct a fixed amount of heat per unit area across one element.

(See Next Page for Additional Footnotes.)

(CRD) TABLE A-4.6-1 (Continued)

- (5) Assumes radial and axial maximum/average nuclear power peaking factors are not affected by changes in fuel and core geometry.
- (6) Relative thermal stress function defined by Appendix A-1, Section A-1.9.1.
- (7) For PEWEE-2 development coating profile on WNCO GEM-HED elements, as measured on 72 elements. See Figure A-4.0-3.
- (8) Defect-free hot end corrosion based on carbon diffusion only. Corrosion rate based on Figure A-4.0-2, independent of coating and matrix materials.
- (9) Mid-band corrosion proportional to bore surface area and fuel length with a ten(10) gram loss assumed for the reference element. Possible differences in mid-band performance for different materials are not considered.
- (10) Assuming interelement gap thickness constant for all alternates.
- (11) Assuming axial distribution of power is the same for carbide fuel as reference.
- (12) Thermal conductivity is 90% theoretical density material.

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(CRD) A-4.6.1.2 Thermal (U)

(CRD) Preliminary data on the thermal conductivity of UC-ZrC carbides indicates much lower values than for graphite fuels. Assuming a composition with 90 percent of maximum theoretical density, maximum temperature levels are given in Tables A-4.0-2 and A-4.6-1. The maximum ΔT from hot spot to channel wall is 94°R versus 173°R for the high expansion graphite element (Alternate 2). The axial temperature profile is given in Figure A-4.6-1.

(U) The normalized temperature gradient per unit heat flux across the element is 0.30 (versus 1.0), the lowest of any alternate (because of the small size).

(CRD) LASL⁽¹⁾ has estimated that these elements can run within $\pm 100^{\circ}\text{R}$ (assumed 3σ) of the average with reasonable production tolerances on loading and hole diameter. This is virtually the same as for the other alternates.

(CRD) The maximum volumetric heat generation rate is $75.3 \text{ Btu/sec-in}^3$ and the corresponding relative thermal stress factor, Q , is $0.0447 \text{ Btu/sec-in}^2$. At very high temperatures (above the level anticipated for R-1 nominal conditions), there is some evidence that the pure carbide would be reduced to metal by the hydrogen. The exact time-temperature relationship of this chemical reaction has not been studied in sufficient detail to form positive conclusions. However, the absolute temperature limit of the pure carbide is defined by the lowest temperature where the carbon stoichiometry cannot be maintained for the time required by the mission.

A-4.6.2 (CRD) Corrosion and Reactivity Loss (U)

(CRD) For temperatures below the reaction rate for reducing carbide to metal (see above), no corrosion or reactivity loss would occur.

A-4.7 (CRD) ZIG-ZAG COMPOSITE ELEMENT, DWG 947C807 (U)

(CRD) This element is identical to the standard geometry composite element (Alternate 5) in regard to flow, thermal and corrosion performance except for two differences:

(1) Reference: LASL Document AN-519, "Long Life Reactor Task Group Status Report, 8/1/69."

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1) The outer perimeter of the fuel places the holes in a true equilateral triangular array at the junction between elements. Therefore, the slightly higher temperature levels at the outer webs of the reference element caused by a rectangular hole arrangement are reduced to the same level as for the interior webs.

The maximum axial tensile thermal stress is proportional to the difference between the average matrix temperature and the minimum matrix temperature. The minimum, at the channel wall, is not affected by this change in geometry. The average fuel temperature, using consistent assumptions, is estimated to be reduced by, at most, 10°R . The change in the fuel average-to-minimum ΔT represented by this 10°R is about 10 percent.

2) The surface area of the fuel element increases by 15.3 percent over the reference element which represents a corresponding increase in interstitial surface corrosion. The above assumes that the average gap thickness between elements can be held the same as the reference, but the practicality of doing this on a production basis is very remote.

(U) A comparison with the reference high expansion graphite element, (Alternate 2) is given in Table A-4.7-1. Otherwise, all comments and figures given for the standard geometry composite element in Section A-4.5 apply and the corresponding sections under A-4.7 are omitted.

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TABLE A-4.7-1

Page 1 of 4

(CRD) COMPARISON OF FLOW, THERMAL, AND CORROSION PERFORMANCE FOR
ZIG-ZAG COMPOSITE ELEMENT, ALTERNATE 7 (U)

	Alternate Dwg. 947C807	Reference Alternate 2 (Dwg. 947C813)
0. DESCRIPTION		
0.1 Matrix Material	Composite	AXM
0.2 Coating Material	ZrC	75 gm Nb, ZrC
0.3 Reamed Diameter - Inch	0.102	0.102
0.4 Hole Pitch - Inch	0.173	0.173
0.5 Across Flats - Inch	0.753*	0.753
0.6 Void Fraction	0.314	0.314
0.7 Core Length - Inch	52	52
0.8 Core Fueled Diameter - Inch	37.1	37.1
0.9 Number of Channels Per Element	19	19
0.10 Number of Channels Per Core	35682	35682
1. FLOW		
1.1 Equivalent Impedance Diameter-in.		
1.1.1 Min. Based on PW-2 Avg. Coating (2.18 mils)	0.0935	0.0935
1.1.2 Min. Based on Experimental PW-2 Thick Hot End Coating (Dwg. 939J709-G01)	0.0914	0.0914
1.2 Core Pressure Drop ⁽¹⁾ - Psi		
1.2.1 Based on diameter in 1.1.1	162	162
1.2.2 Based on diameter in 1.1.2	176	176

(1) Based on 1.1 power factor, except Alternate 6 (single channel - carbide), which is based on 1.0 power factor. Neglects taper in bore due to varying coating thickness.

* This is the difference between the "mean lines" of the opposite sides.

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TABLE A-4.7-1 (Continued)

Page 2 of 4

Alternate Dwg.
947C807

Reference Alternate 2
Dwg. 947C813

2. THERMAL

2.1	(T_W) Max. Channel Wall Temp ⁽²⁾ °R		
2.1.1	Nominal	4389	4389
2.1.2	Hot Channel ⁽³⁾ (3 σ)	4737	4737
2.1.3	Difference from reference	0	—
2.1.4	Axial Station - Inch	48	48
2.2	(T_F) Max. Fuel Hot Spot Temp ⁽²⁾ °R		
2.2.1	Nominal	4473	4463
2.2.2	Hot Channel ⁽³⁾ (3 σ)	4822	4811
2.2.3	Difference from reference	+11	—
2.2.4	Axial Station - Inch	46	46
2.3	Temperature Differences		
2.3.1	Max. ΔT ⁽²⁾ within nominal Element $(T_F - T_W)$ - °R	177	173
2.3.2	Statistical uncertainty in element-to-element average exit temp. ⁽³⁾ (3 σ) - °R	106	106
2.3.3	Normalized temp. gradient per unit heat flux ⁽⁴⁾	1.33	1.0
2.4	Max. Volumetric Heating Rate ⁽⁵⁾ Btu/sec-inch ³	66.4	66.4
2.5	Max. Thermal Stress Function ^(5, 6) ϕ - Btu/sec-inch	0.0714	0.0714

(2) Isolated channel calculation by MCAP for equilateral triangular array of coolant holes and 1.1 power factor (except alternate 6, single channel - carbide).

(3) Represents value for core periphery after one power test, neglecting control drum effects, but including $\pm 63^\circ\text{R}$ uncertainty in mean fuel exit gas temperature.

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Alternate Dwg.
947C807Reference Alternate 2
Dwg. 947C8133.0 CORROSION BASED ON CORE SIZE AND
FUEL ELEMENT GEOMETRY

3.1	Avg. Carbon Reactivity Coefficient $\frac{g}{kg}$	4.30	3.90
3.2	Hot end diffusion loss for defect-free coating(7, 8) using corrosion rate relationship in Fig. A-4.0-2		
3.2.1	Grams per element	14.6	14.6
3.2.2	Max. Incremental gm/inch	1.17	1.17
3.2.3	Reactivity Loss - \$	0.78	0.70
3.2.4	Relative bore deterioration at max. incremental gm/in. (item 3.2.2)		
3.2.4.1	Normalized grams carbon removed per unit bore area	1.0	1.0
3.2.4.2	Percent of carbon mass lost from nominal internal web	19.7	17.2
3.3	Assumed Mid-Band Loss ⁽⁹⁾		
3.3.1	Grams per element	10.0	10.0
3.3.2	Reactivity loss - \$	0.85	0.78
3.4	Relative interelement hydrogen leakage based on geometry (11)	1.153	1.0

(4) Represents element across flats dimension divided by matrix thermal conductivity (at 4000°R) normalized to reference. This is a measure of the temperature gradient to conduct a fixed amount of heat per unit area across one element.

(See Next Page for Additional Footnotes.)

(CRD) TABLE A-4.771 / (Continued)

- (5) Assumes radial and axial maximum/average nuclear power peaking factors are not affected by changes in fuel and core geometry.
- (6) Relative thermal stress function defined by Appendix A-1, Section A-1.9.1.
- (7) For PEWEE-2 development coating profile on WNCO GEM-HED elements, as measured on 72 elements. See Figure A-4.0-3.
- (8) Defect-free hot end corrosion based on carbon diffusion only. Corrosion rate based on Figure A-4.0-2, independent of coating and matrix materials.
- (9) Mid-band corrosion proportional to bore surface area and fuel length with a ten(10) gram loss assumed for the reference element. Possible differences in mid-band performance for different materials are not considered.
- (10) Assuming interelement gap thickness constant for all alternates.

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A-4.8 (CRD) HIGH VOID COMPOSITE ELEMENT LONG FUEL, DWG 947C814 (U)

A-4.8.0 (CRD) Introduction and Summary (U)

(CRD) The high void fuel element requires the second lowest core pressure drop of any alternate 141 psi, with the same coating profile as used in PW-2. The flow and power per element are the same as the reference element, but temperatures are reduced slightly because of the longer core. Otherwise, flow and thermal performance are very similar to the reference element. The geometric factor for defect-free hot end coating indicates a slightly lower value of maximum grams per inch than the reference element (1.13 gm/in versus 1.17). The corresponding mass of carbon removed per unit bore area is correspondingly less, but represents a higher percentage (19.7 percent versus 17.2 percent) of the carbon in the nominal web.

(U) The normalized relative interstitial leakage (see Table A-4.8-1) is 0.70, the lowest of any alternate. There is no corrosion test data or manufacturing history available for high void composite elements to indicate possible effects of geometry changes on cyclic performance or basic material properties. A comparison with the reference high expansion graphite element is given in Table A-4.8-1.

A-4.8.1 (U) Flow and Thermal Characteristics (U)

A-4.8.1.1 (U) Flow (U)

(U) This element and core design has the lowest core pressure drop of any coated alternate and, therefore, has greater potential for thicker coatings to enhance corrosion performance. Also, the minimum equivalent impedance diameter is less likely to be affected by increased coating roughness or extreme taper in the coating caused by thicker aft end coating. See Tables A-4.0-1 and A-4.8-1.

TABLE A-4.8-1

Page 1 of 4

(CRD) COMPARISON OF FLOW, THERMAL, AND CORROSION PERFORMANCE FOR
HIGH VOID COMPOSITE ELEMENT IN LONG CORE, ALTERNATE 8 (U)

	Alternate Dwg. 947C814	Reference Alternate 2 (Dwg. 947C813)
0. DESCRIPTION		
0.1 Matrix Material	Composite	AXM
0.2 Coating Material	ZrC	75 gm Nb, ZrC
0.3 Reamed Diameter - Inch	0.108	0.102
0.4 Hole Pitch - Inch	0.173	0.173
0.5 Across Flats - Inch	0.753	0.753
0.6 Void Fraction	0.354	0.314
0.7 Core Length - Inch	62	52
0.8 Core Fueled Diameter - Inch	37.1	37.1
0.9 Number of Channels Per Element	19	19
0.10 Number of Channels Per Core	35682	35682
1. FLOW		
1.1 Equivalent Impedance Diameter-in.		
1.1.1 Min. Based on PW-2 Avg. Coating (2.18 mils)	0.0995	0.0935
1.1.2 Min. Based on Experimental PW-2 Thick Hot End Coating (Dwg. 939J709-G01)	0.0974	0.0914
1.2 Core Pressure Drop ⁽¹⁾ - Psi		
1.2.1 Based on diameter in 1.1.1	141	162
1.2.2 Based on diameter in 1.1.2	150	176

(1) Based on 1.1 power factor, except Alternate 6 (single channel - carbide), which is based on 1.0 power factor. Neglects taper in bore due to varying coating thickness.

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Page 2 of 4

TABLE A-4.8-1 (Continued)

Alternate Dwg.
947C814

Reference Alternate 2
Dwg. 947C813

2. THERMAL

2.1 (T_W) Max. Channel Wall Temp⁽²⁾ °R

2.1.1	Nominal	4371	4389
2.1.2	Hot Channel ⁽³⁾ (3 σ)	4719	4737
2.1.3	Difference from reference	-18	—
2.1.4	Axial Station - Inch	59.5	48

2.2 (T_F) Max. Fuel Hot Spot Temp⁽²⁾ °R

2.1.1	Nominal	4434	4463
2.1.2	Hot Channel ⁽³⁾ (3 σ)	4782	4811
2.1.3	Difference from reference	-29	—
2.1.4	Axial Station - Inch	57	46

2.3 Temperature Differences

2.3.1	Max. ΔT ⁽²⁾ within nominal Element $(T_F - T_W)$ - °R	136	173
2.3.2	Statistical uncertainty in element-to-element average exit temp. ⁽³⁾ (3 σ) - °R	106	106
2.3.3	Normalized temp. gradient per unit heat flux ⁽⁴⁾	1.33	1.0

2.4 Max. Volumetric Heating Rate⁽⁵⁾
Btu/sec-inch³

58.9 66.4

2.5 Max. Thermal Stress Function^(5, 6)
Q - Btu/sec-inch

0.054 0.0714

(2) Isolated channel calculation by MCAP for equilateral triangular array of coolant holes and 1.1 power factor (except alternate 6, single channel - carbide).

(3) Represents value for core periphery after one power test, neglecting control drum effects, but including $\pm 63^\circ\text{R}$ uncertainty in mean fuel exit gas temperature.

(See Next Page for Additional Footnotes.)

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(CRD) TABLE A-4.8-1 (Continued)

Page 3 of 4

Alternate Dwg.
947C 814Reference Alternate 2
Dwg. 947C813

3.0 CORROSION BASED ON CORE SIZE AND
FUEL ELEMENT GEOMETRY

3.1	Avg. Carbon Reactivity Coefficient $\frac{\text{g}}{\text{KG}}$	4.30	3.90
3.2	Hot end diffusion loss for defect-free coating(7, 8) using corrosion rate relationship in Fig. A-4.0-2		
3.2.1	Grams per element	17.0	14.6
3.2.2	Max. Incremental gm/inch	1.13	1.17
3.2.3	Reactivity Loss - \$	0.86	0.70
3.2.4	Relative bore deterioration at max. incremental gm/in. (item 3.2.2)		
3.2.4.1	Normalized grams carbon removed per unit bore area	0.92	1.0
3.2.4.2	Percent of carbon mass lost from nominal internal web	19.7	17.2
3.3	Assumed Mid-Band Loss ⁽⁹⁾		
3.3.1	Grams per element	12.6	10.0
3.3.2	Reactivity loss - \$	1.08	0.78
3.4	Relative interelement hydrogen leakage based on geometry (11)	0.70	1.0

(4) Represents element across flats dimension divided by matrix thermal conductivity (at 4000°R) normalized to reference. This is a measure of the temperature gradient to conduct a fixed amount of heat per unit area across one element.

(See Next Page for Additional Footnotes.)

(CRD) TABLE A-4.8-1 (Continued)

- (5) Assumes radial and axial maximum/average nuclear power peaking factors are not affected by changes in fuel and core geometry.
- (6) Relative thermal stress function defined by Appendix A-1, Section A-1.9.1.
- (7) For PEWEE-2 development coating profile on WNCO GEM-HED elements, as measured on 72 elements. See Figure A-4.0-3.
- (8) Defect-free hot end corrosion based on carbon diffusion only. Corrosion rate based on Figure A-4.0-2, independent of coating and matrix materials.
- (9) Mid-band corrosion proportional to bore surface area and fuel length with a ten(10) gram loss assumed for the reference element. Possible differences in mid-band performance for different materials are not considered.
- (10) Assuming interelement gap thickness constant for all alternates.

A-4.8.1.2 (U) Thermal (U)

(U) Both channel wall temperatures and fuel hot spot temperatures are lower than for the reference element at the same relative axial position. A plot of the axial temperature profile is given in Figure A-4.8-1. The maximum fuel hot spot-to-channel wall ΔT is 136°R . The normalized temperature gradient per unit heat flux is higher than for the reference high expansion graphite element, indicating greater sensitivity to imposed transverse power variations. The estimated element-to-element temperature difference (3σ) is $\pm 106^{\circ}\text{R}$ for the core periphery, the same as the other alternates. The maximum volumetric heating rate is $58.9 \text{ Btu/sec-in}^3$ (versus 66.4) and the corresponding relative heat generation thermal stress function is $0.054 \text{ Btu/sec-in}^2$ (versus 0.0714).

A-4.8.2 (CRD) Corrosion and Reactivity Loss (U)

(CRD) A-4.8.2.1 Geometric Factors for Defect-Free Hot End Coating (U)

(CRD) Assuming defect-free hot end coating for the 10-hour operation, the maximum incremental corrosion is 1.13 grams per inch versus 1.17 for the reference high expansion graphite element. Because of the increased void fraction and length, the weight loss per element is increased to 17 grams for the hot end diffusion corrosion (versus 14.6) and 12.6 grams for the assumed mid-band. The normalized maximum grams of carbon per unit bore area is 0.92 which represents 19.7 percent of the carbon mass in the nominal web.

A-4.8.2.2 (U) Cyclic Performance - Influence of Coating Defects (U)

(U) There is no available data on possible cyclic effects for this alternate. Assuming there are no changes in performance caused by the different geometry, the discussion given under A-4.5.2.2 for the all-composite element is pertinent.

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High Void Composite Long Fuel, Alternate 8, Drawing 947C814

PF = 1.1

Channel Diameter = 0.0995 Inch

Figure A-4.8-1 (CRD) Temperature Versus Axial Station for High Void Composite Element in Long Core (U)

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A-4.9(CRD) HIGH EXPANSION GRAPHITE ELEMENT - LONG FUEL, DWG 947C815 (U)

A-4.9.0 (CRD) Introduction and Summary (U)

(U) This fuel element and core design has about the same core pressure drop as the reference, but reduced flow and power per element. Fuel temperatures are reduced from the levels in the reference high expansion graphite, but not dramatically so, because the channel wall-to-gas film temperature drop is only a weak function of power. The maximum nominal channel wall temperature is 4340°R versus 4389°R for Alternate 2, the maximum fuel hot spot temperature is 4373°R versus 4463°R . The maximum fuel channel wall ΔT is 94°R , almost half the value for Alternate 2. The volumetric heating rates and relative thermal stress due to heat generation are reduced inversely in proportion to core volume.

(CRD) Relative corrosion rates per unit of bore length and area are the lowest of any alternate indicating less severe damage to the element for defect-free corrosion.

(CRD) It is assumed that cyclic performance capability and resistance to defect corrosion would be similar to that of the reference (high expansion graphite) element, Alternate 2. Possibly, some slight improvement may be noticeable because of the reduced temperatures and power density.

(U) A comparison with the reference high expansion graphite element (Alternate 2) is given in Table A-4.9-1.

A-4.9.1 (U) Flow and Thermal Characteristics (U)

A-4.9.1.1 (U) Flow (U)

(U) Core pressure drop is 158 psi assuming the same average coating thickness as PW-2 (Figure A-4.0-3). The core size was picked from parameter studies to maintain the same core ΔP as the reference R-1 design. Flow and power per element are reduced in proportion to the number of elements.

TABLE A-4.9-1

Page 1 of 4

(CRD) COMPARISON OF FLOW, THERMAL, AND CORROSION PERFORMANCE FOR
HIGH EXPANSION GRAPHITE ELEMENT IN LARGE CORE, ALTERNATE 9 (U)

	Alternate Dwg. 947C815	Reference Alternate 2 (Dwg. 947C813)
0. DESCRIPTION		
0.1 Matrix Material	AXM	AXM
0.2 Coating Material	84% NbC	75 gm Nb, ZrC
0.3 Reamed Diameter - Inch	0.102	0.102
0.4 Hole Pitch - Inch	0.173	0.173
0.5 Across Flats - Inch	0.753	0.753
0.6 Void Fraction	0.314	0.314
0.7 Core Length - Inch	76	52
0.8 Core Fueled Diameter - Inch	40.6	37.1
0.9 Number of Channels Per Element	19	19
0.10 Number of Channels Per Core	42864	35682
1. FLOW		
1.1 Equivalent Impedance Diameter-in.		
1.1.1 Min. Based on PW-2 Avg. Coating (2.18 mils)	0.0935	0.0935
1.1.2 Min. Based on Experimental PW-2 Thick Hot End Coating (Dwg. 939J709-G01)	0.0914	0.0914
1.2 Core Pressure Drop ⁽¹⁾ - Psi		
1.2.1 Based on diameter in 1.1.1	158	162
1.2.2 Based on diameter in 1.1.2	174	176

(1) Based on 1.1 power factor, except Alternate 6 (single channel - carbide), which is based on 1.0 power factor. Neglects taper in bore due to varying coating thickness.

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Page 2 of 4

TABLE A-4.9-1 (Continued)

Alternate Dwg.
947C815Reference Alternate 2
Dwg. 947C813

2. THERMAL

2.1	(T_W) Max. Channel Wall Temp ⁽²⁾ °R		
2.1.1	Nominal	4340	4389
2.1.2	Hot Channel ⁽³⁾ (3 σ)	4688	4737
2.1.3	Difference from reference	-49	—
2.1.4	Axial Station - Inch	73	48
2.2	(T_F) Max. Fuel Hot Spot Temp ⁽²⁾ °R		
2.2.1	Nominal	4373	4463
2.2.2	Hot Channel ⁽³⁾ (3 σ)	4721	4811
2.2.3	Difference from reference	-90	—
2.2.4	Axial Station - Inch	73	46
2.3	Temperature Differences		
2.3.1	Max. ΔT ⁽²⁾ within nominal Element $(T_F - T_W)$ - °R	94	173
2.3.2	Statistical uncertainty in element-to-element average exit temp. ⁽³⁾ (3 σ) - °R	106	106
2.3.3	Normalized temp. gradient per unit heat flux ⁽⁴⁾	1.0	1.0
2.4	Max. Volumetric Heating Rate ⁽⁵⁾ Btu/sec-inch ³	45.4	66.4
2.5	Max. Thermal Stress Function ^(5, 6) Q - Btu/sec-inch	0.040	0.0714

(2) Isolated channel calculation by MCAP for equilateral triangular array of coolant holes and 1.1 power factor (except alternate 6, single channel - carbide).

(3) Represents value for core periphery after one power test, neglecting control drum effects, but including $\pm 63^\circ\text{R}$ uncertainty in mean fuel exit gas temperature.

(See Next Page for Additional Footnotes.)

Alternate Dwg.
947C 815Reference Alternate 2
Dwg. 947C8133.0 CORROSION BASED ON CORE SIZE AND
FUEL ELEMENT GEOMETRY

3.1	Avg. Carbon Reactivity Coefficient $\frac{\text{g}}{\text{kg}}$	3.21	3.90
3.2	Hot end diffusion loss for defect-free coating(7, 8) using corrosion rate relationship in Fig. A-4.0-2		
3.2.1	Grams per element	15.8	14.6
3.2.2	Max. Incremental gm/inch	0.96	1.17
3.2.3	Reactivity Loss - \$	0.68	0.70
3.2.4	Relative bore deterioration at max. incremental gm/in. (item 3.2.2)		
3.2.4.1	Normalized grams carbon removed per unit bore area	0.84	1.0
3.2.4.2	Percent of carbon mass lost from nominal internal web	14.4	17.2
3.3	Assumed Mid-Band Loss ⁽⁹⁾		
3.3.1	Grams per element	14.6	10.0
3.3.2	Reactivity loss - \$	1.12	0.78
3.4	Relative interelement hydrogen leakage based on geometry (11)	0.78	1.0

(4) Represents element across flats dimension divided by matrix thermal conductivity (at 4000°R) normalized to reference. This is a measure of the temperature gradient to conduct a fixed amount of heat per unit area across one element.

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Page 4 of 4

(CRD) TABLE A-4.9-1 (Continued)

- (5) Assumes radial and axial maximum/average nuclear power peaking factors are not affected by changes in fuel and core geometry.
- (6) Relative thermal stress function defined by Appendix A-1, Section A-1.9.1.
- (7) For PEWEE-2 development coating profile on WNCO GEM-HED elements, as measured on 72 elements. See Figure A-4.0-3.
- (8) Defect-free hot end corrosion based on carbon diffusion only. Corrosion rate based on Figure A-4.0-2, independent of coating and matrix materials.
- (9) Mid-band corrosion proportional to bore surface area and fuel length with a ten(10) gram loss assumed for the reference element. Possible differences in mid-band performance for different materials are not considered.
- (10) Assuming interelement gap thickness constant for all alternates.

[REDACTED]

A-4.9.1.2 (U) Thermal (U)

(U) The axial temperature profile is given in Figure A-4.9-1. Because of the reduced heating rates, radial ΔT 's (in fuel and film drop) are reduced significantly. The maximum volumetric heating rate is 45.4 Btu/sec-in³ (versus 66.4) and the corresponding thermal stress function is 0.040 Btu/sec-in². The element has the same estimated element-to-element temperature uncertainty ($\pm 106^{\circ}\text{R}$) as the other alternates. Also, the resistance to imposed power generation difference across the element (given as normalized $^{\circ}\text{R}$ /unit heat flux) is the same as the reference. (See Table A-4.9-1.)

A-4.9.2 (CRD) Corrosion and Reactivity Loss (U)

(CRD) A-4.9.2.1 Geometric Factors for Defect-Free Hot End Coating (U)

(CRD) Assuming defect-free hot end coating for 10 hour's operation, the maximum incremental carbon loss per inch due to hot-end diffusion is 0.96 gm/in. versus 1.17 gm/inch for Alternate 2. The corresponding normalized grams per unit bore area is 0.84 versus 1.0, which represents 14.4 percent of the carbon in the nominal web. Because of the longer fuel elements (and longer surface area exposed to high temperatures), the total weight loss per element is higher than Alternate 2, 15.8 grams hot end versus 14.6 and 14.6 grams mid-band versus 10.0.

(CRD) A-4.9.2.2 Cyclic Performance - Influence of Coating Defects (U)

(CRD) There is no data to assess cyclic performance for the longer fuel element, but electrical corrosion tests (see Section A-4.1-2) indicate that lower temperatures and reduced heating rates may tend to result in less noticeable acceleration in corrosion rates. With that qualification in mind, the discussion given for the standard length high expansion graphite element in Section A-4.2.2.2 is believed to be generally applicable.

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Figure A-4.9-1 (CRD) Temperature Versus Axial Station for High Expansion Graphite Element in Large Core (U)

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APPENDIX A-5

STRUCTURAL DESIGN SUPPORTING DATA
TRADE STUDY FOR NERVA FUEL ELEMENT
DESIGN CONCEPTS

TRADE STUDY NO. 759

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A-5.0 INTRODUCTION (U)

(U) This appendix contains the results of structural analyses on the nine (9) alternates of fuel element designs for the Fuel Element Conceptual Design Trade Study. The results of the analysis are summarized in Table A-5-1, and are obtained from Section A-1.3.1.

A-5.1 DESIGN CONCEPT 2.1 (INTERMEDIATE CTE FUEL ELEMENT - DRAWING 947C812) (U)

(C-RD) Fuel element alternate 2.1 is a standard 19 hole, .753 across flats fuel element with an intermediate expansion graphite matrix (GLFI) with duplex coated bores (HED ZrC and GEM NbC). The reamed bore diameter is .102, and the average bore coat thickness is 2.2 mils. The assumed coating distribution is shown on Figure A-1.3-10.

A-5.1.1 Bundling Load (U)

(C-RD) The bundling load on a 37.1" fuel envelope is approximately 30 psi. This load produces negligible compressive stresses on the GLFI fuel elements as compared to heat generation and coating-matrix interaction stresses.

A-5.1.2 Axial Load (U)

(U) The axial load on the GLFI element is maximum for the bearing or contact design and is approximately equal to the core ΔP . This axial stress is negligible when compared to the heat generation and coating matrix interaction stresses.

A-5.1.3 Radial Expansion (U)

(C-RD) The radial expansion of this fuel element is (Table A-5-1) 1.146% as compared to a R-1 capability of accommodating 1.665%.

A-5.1.4 Corrosion (U)

(C-RD) The corrosion capability of the GLFI is expected to be unacceptable from an endurance and cyclic standpoint.

(C-RD) The brazed joint strength will probably be decreased after cycling without any corrosion present.

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(C-RD) GLFI element strength will be degraded because of the corrosion from the cycles and from the long reactor operating times.

A-5.1.5 Interface Requirements (U)

(U) No additional interface requirements other than those in the reference periphery are required.

(U) A core composed of GLFI fuel elements will be compatible with the present (reference) core periphery.

(U) A core composed of GLFI fuel elements will be compatible with the present (reference) axial support.

(U) A core composed of GLFI fuel elements will be compatible with the present (reference) lateral support.

A-5.1.6 Structural Criteria (U)

(C-RD) The maximum combined axial stress in the GLFI fuel element is calculated to exceed the structural adequacy limit described in the following paragraph (see Table A-5-1).

(U) According to the provisions of the "NERVA Program Structural Design Requirements," SNPO-C-1, the fuel elements should be treated as brittle materials and analyzed by fracture mechanics or Weibull techniques. Lack of fracture toughness and Weibull constant experimental data precludes performing a brittle analysis. An alternative criterion of structural adequacy was chosen which is a maximum stress approach with a margin. The criterion chosen for allowable stress is:

$$\sigma_{\max} + 2.3 \sigma_{\text{ms}} \leq 0.8 (\sigma_{\text{ult}} - 2.3 \sigma_{\text{us}})$$

where:

- $\sigma_{\max}$ = Maximum calculated stress in nominal analysis
 σ_{ms} = Standard deviation of maximum calculated stress
 σ_{ult} = Mean ultimate strength
 σ_{us} = Standard deviation of ultimate strength

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This approach parallels a rule in SNPO-C-1 intended for ductile material applications.

(C-RD) The portion of the maximum combined axial stress in the GLFI fuel element that is caused by bundling friction and a 2.3σ interelement temperature difference of 81°R is 500 psi. The sensitivity of the GLFI element to bundling friction (defined as the ratio of friction stress to combined allowable axial stress) is .227.

(C-RD) The axial heat generation stress in the GLFI fuel element is 488 psi. The 2.3σ variation in the heat generation stress due to a 2.3σ variation in the material properties component dimensions, and nuclear heating is 307 psi.

(C-RD) The axial coating-matrix interaction stress in the matrix is 1470 psi. This coating-matrix interaction stress is at best an estimate, judged to be conservative. The coating and matrix are assumed to be free of interaction stress at the ZrC deposition temperature (see Figure A-1.3-10), perfectly adherent, and behaving elastically. These same assumptions regarding the coating-matrix interaction stress calculations are applied to all fuel alternates except the Single Channel-Carbide Element, alternate 2.6, which is uncoated. The combined axial stress is 2543 psi. This is greater than the $.8 (\sigma_{ult} - 2.3 \sigma_{us})$ minimum tensile strength of 2200 psi for the GLFI fuel element. The matrix-bead interaction stress in the matrix is negligible when compared to the heat generation stress and the coating-matrix interaction stress. The matrix-bead interaction stress occurs at temperatures greater than the graphitization temperature. If the graphitization temperature is greater than the maximum fuel temperature, then no bead-matrix interaction will occur. In R-1, the graphitization temperature is assumed to be about 4450°R as compared to a maximum fuel temperature of 4460°R .

(C-RD) The sensitivity of the GLFI element to a transverse temperature gradient is about 260 psi per 100°R temperature difference across the flats of the element.

A-5.2 DESIGN CONCEPT 2.2 (HIGH CTE FUEL ELEMENT - DRAWING 947C813) (U)

(C-RD) Fuel element alternate 2.2 is a standard 19 hole, .753" across flats fuel element with a high expansion graphite matrix (AXM) with duplex coated bores (ZrC and NbC). The reamed bore diameter is .102 in., and the average bore coating thickness is

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2.2 mils. The assumed coating distribution is shown on Figure A-1.3-10.

A-5.2.1 Bundling Load (U)

(C-RD) The bundling load on a 37.1" fuel envelope is approximately 30 psi. This load produces negligible compressive stresses on the AXM fuel elements as compared to heat generation and coating-matrix interaction stresses.

A-5.2.2 Axial Load (U)

(U) The axial load on the AXM element is maximum for the bearing or contact design and is approximately equal to the core ΔP . This axial stress is negligible when compared to the heat generation and coating-matrix interaction stresses.

A-5.2.3 Radial Expansion (U)

(C-RD) The radial expansion of the AXM fuel element is (Table A-5-1) 1.48% as compared to an R-1 capability of accommodating 1.665%.

A-5.2.4 Corrosion (U)

(C-RD) The corrosion capability of the AXM is expected to be unacceptable from an endurance and cyclic standpoint.

(C-RD) The brazed joint strength will probably be decreased after cycling without any corrosion present.

(C-RD) AXM element strength will be degraded because of the corrosion from the cycles and from the long reactor operating times.

A-5.2.5 Interface Requirements (U)

(U) No additional interface requirements other than those in the reference periphery are required.

(U) A core composed of AXM fuel elements will be compatible with the present (reference) core periphery.

(U) A core composed of AXM fuel elements will be compatible with the present (reference) axial support.

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(U) A core composed of AXM fuel elements will be compatible with the present (reference) lateral support.

A-5.2.6 Structural Criteria (U)

(C-RD) The AXM fuel element does not meet the maximum stress structural adequacy criterion described in Section A-5.1.6 (see Table A-5-1).

(C-RD) The axial stress in the AXM element that is caused by bundling friction and a 2.3σ interelement temperature difference of 81°R is 500 psi. The sensitivity of the AXM element to bundling friction is .256.

(C-RD) The axial heat generation stress in the AXM fuel element is 821 psi. The 2.3σ variation in the heat generation stress due to a 2.3σ variation in the material properties, component dimensions, and nuclear heating is 518 psi.

(C-RD) The axial coating-matrix interaction stress in the matrix is calculated to be 501 psi based on the interaction stress assumptions given in Section A-5.1.6. The combined axial stress is 2042 psi. This is greater than the $.8(\sigma_{\text{ult}} - 2.3\sigma_{\text{us}})$ minimum tensile strength of 1950 psi for the AXM fuel element.

(C-RD) The matrix-bead interaction stress in the matrix is negligible when compared to the heat generation stress and the coating-matrix interaction stress. The matrix-bead interaction stress occurs at temperatures greater than the graphitization temperature. If the graphitization temperature is greater than the maximum fuel temperature, no bead-matrix interaction occurs. In R-1, the graphitization temperature is assumed to be about 4450°R as compared to a maximum fuel temperature of 4460°R .

(C-RD) The sensitivity of the AXM element to a transverse temperature gradient is about 400 psi per 100°R temperature difference across the flats of the element.

A-5.3 DESIGN CONCEPT 2.3 (HYBRID FUEL - STANDARD GEOMETRY -
DRAWING 947C810) (U)

(C-RD) Fuel element alternate 2.3 is a standard 19 hole, .753" across flats fuel element. Station 0 to station 35 is composed of AXM high expansion matrix. Station 35 to

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52 is composed of 30 V/O UC-ZrC-C composite matrix. The reamed bore diameter is .102 in. and the average coating thickness is 2.2 mils of ZrC. The assumed coating distribution is shown in Figure A-1.3-10.

A-5.3.1 Bundling Load (U)

(C-RD) The bundling load on a 37.1" fuel envelope is approximately 30 psi. This load produces negligible compressive stresses on the hybrid element as compared to heat generation and coating-matrix interaction stresses.

A-5.3.2 Axial Load (U)

(U) The axial load on the hybrid element is maximum for the bearing or contact design and is approximately equal to the core ΔP . This axial stress is negligible when compared to the heat generation and coating-matrix interaction stresses.

A-5.3.3 Radial Expansion (U)

(C-RD) The radial expansion of the hybrid element is (Table A-5-1) 1.54% as compared to an R-1 capability of accommodating 1.665%.

A-5.3.4 Corrosion (U)

(C-RD) The corrosion capability of the hybrid is expected to be improved from an endurance and cyclic standpoint, as compared to the graphite elements.

(C-RD) The brazed joint strength will probably be improved as compared to the graphite elements after cycling without any corrosion present.

(C-RD) Hybrid element strength will be degraded because of the corrosion from the cycles and from the long reactor operating times.

A-5.3.5 Interface Requirements (U)

(U) No additional interface requirements other than those in the reference periphery are required.

(U) A core composed of hybrid fuel elements will be compatible with the present (reference) core periphery.

(U) A core composed of hybrid fuel elements will be compatible with the present (reference) axial support.

(U) A core composed of hybrid fuel elements will be compatible with the present (reference) lateral support.

A-5.3.6 Structural Criteria (U)

(C-RD) The hybrid fuel element does not meet the maximum stress structural adequacy criterion described in Section A-5.1.6 (see Table A-5-1). It is limited by the AXM portion.

(C-RD) The axial stress in the AXM portion of the hybrid element that is caused by bundling friction and a 2.3σ interelement temperature difference of 81°R is 525 psi. The sensitivity of the AXM portion to bundling friction is .280 psi.

(C-RD) The axial heat generation stress in the AXM fuel portion is 815 psi. The 2.3σ variation in the heat generation stress due to a 2.3σ variation in the material properties, component dimensions, and nuclear heating is 514 psi.

(C-RD) The axial coating-matrix interaction stress in the matrix is calculated to be 455 psi based on the interaction stress assumptions given in Section A-5.1.6. The combined axial stress is 2005 psi. This is greater than the $.8(\sigma_{ult} - 2.3\sigma_{us})$ minimum tensile strength of 1940 psi for the AXM portion of the hybrid fuel element.

(C-RD) The matrix-bead interaction stress in the matrix is negligible when compared to the heat generation stress and the coating-matrix interaction stress. The matrix-bead interaction stress occurs at temperatures greater than the graphitization temperature. If the graphitization temperature is greater than the maximum fuel temperature, no bead-matrix interaction occurs. In R-1, the graphitization temperature is assumed to be about 4450°R as compared to a maximum fuel temperature of 4460°R .

(C-RD) The sensitivity of the element to a transverse temperature gradient is about 1500 psi per 100°R temperature difference across the flats of the element (composite portion).

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A-5.4 DESIGN CONCEPT 2.4 (37 HOLE HYBRID ELEMENT - DRAWING 947C816) (C-RD)

(C-RD) Fuel element alternate 2.4 is a 37 hole-hybrid element. Station 0 to station 35 is composed of AXM high expansion matrix. Station 35 to station 52 is composed of 30 V/O UC-ZrC-C material. The assumed coating distribution is shown on Figure A-1.3-10.

A-5.4.1 Bundling Load (U)

(C-RD) The bundling load on a 37.1" fuel envelope is approximately 30 psi. This load produces negligible compressive stresses on the hybrid element as compared to heat generation and coating-matrix interaction stresses.

A-5.4.2 Axial Load (U)

(U) The axial load on the hybrid element is maximum for the bearing or contact design and is approximately equal to the core ΔP . This axial stress is negligible when compared to the heat generation and coating-matrix interaction stresses.

A-5.4.3 Radial Expansion (U)

(C-RD) The radial expansion of the 37 hole hybrid is (Table A-5-1) 1.505% as compared to a R-1 capability of accommodating 1.665%.

A-5.4.4 Corrosion (U)

(C-RD) The corrosion capability of the hybrid-AXM is expected to be improved from an endurance standpoint as compared to the graphite elements. The brazed joint strength will probably be improved as compared to the graphite elements after cycling without any corrosion present. Hybrid-AXM element strength will be degraded because of the corrosion from the cycles and from the long reactor operating times.

A-5.4.5 Interface Requirements (U)

(U) No additional interface requirements other than those in the reference periphery are required. A core composed of hybrid-AXM fuel elements will be compatible with the present (reference) core periphery. A core composed of hybrid-AXM fuel elements will be compatible with the present (reference) axial support. A core composed of hybrid-AXM fuel elements will be compatible with the present (reference) lateral support.

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A-5.4.6 Structural Criteria (U)

(C-RD) The 37 hole hybrid fuel element does meet the maximum stress structural adequacy criterion described in Section A-5.1.6 (see Table A-5.1).

(C-RD) The axial stress in the AXM and composite at station 35 element that is caused by bundling friction and a 2.3σ interelement temperature difference of 81°R is 525 psi. The sensitivity of the hybrid element to bundling friction is 0.280.

(C-RD) The axial heat generation stress in the AXM matrix is 342 psi. The 2.3σ variation in the heat generation stress due to 2.3σ variation in the material properties, component dimensions, and nuclear heating is 215 psi.

(C-RD) The axial coating-matrix interaction stress in the AXM matrix is calculated to be 869 psi based on the interaction stress assumptions given in Section A-5.1.6. The combined axial stress is 1777 psi. This is less than the $0.8 (\sigma_{ult} - 2.4\sigma_{us})$ minimum tensile strength of 1940 psi for the AXM portion of fuel element.

(C-RD) The matrix-bead interaction stress in the matrix is negligible when compared to the heat generation stress and the coating-matrix interaction stress. The matrix-bead interaction stress occurs at temperatures greater than the graphitization temperature. If the graphitization temperature is greater than the maximum fuel temperature, no bead-matrix interaction occurs. In R-1, the graphitization temperature is assumed to be about 4450°R as compared to a maximum fuel temperature of 4460°R .

(C-RD) The sensitivity of the hybrid element to a transverse temperature gradient is about 1500 psi per 100°R temperature difference across the flats of the element (composite part).

A-5.5 DESIGN CONCEPT 2.5 (30 V/O UC-ZrC-C - STANDARD FUEL ELEMENT - DRAWING 947C809) (C-RD)

(C-RD) Fuel element alternate 2.5 is a standard 19 hole, 0.753 in. across flats fuel element with a composite 30 V/O UC-ZrC matrix. The reamed bore diameter is 0.102 in. and the average coating thickness is 2.2 mils of ZrC. The assumed coating distribution is shown in Figure A-1.3-10, with, however, the NbC portion replaced by ZrC.

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A-5.5.1 Bundling Load (U)

(C-RD) The bundling load on a 37.1 in. fuel envelope is approximately 30 psi. This load produces negligible compressive stresses on the 30 V/O UC-ZrC-C fuel elements as compared to heat generation and coating-matrix interaction stresses.

A-5.5.2 Axial Load (U)

(C-RD) The axial load on the 30 V/O UC-ZrC-C element is maximum for the bearing or contact design and is approximately equal to the core ΔP . This axial stress is negligible when compared to the heat generation and coating-matrix interaction stresses.

A-5.5.3 Radial Expansion (U)

(C-RD) The radial expansion of the composite 19 hole element is 1.54% as compared to a R-1 capability of accommodating 1.665%.

A-5.5.4 Corrosion (U)

(C-RD) The brazed joint strength problem is eliminated because no composite tips are required on this element. Degradation of element strength due to corrosion is substantially less in this element as compared to graphite fuel elements.

A-5.5.5 Interface Requirements (U)

(U) A core composed of composite fuel elements will be compatible with the present (reference) core periphery. The increased weight of a core of composite fuel requires modification of the reference axial support. The increased mass of a core of composite fuel requires modification of the lateral support system.

A-5.5.6 Structural Criteria (U)

(C-RD) The composite fuel element does meet the maximum stress based on the interaction stress assumptions given in Section A-5.1.6.

(C-RD) The axial stress in the composite element that is caused by bundling friction and a 2.3 $^{\circ}$ interelement temperature difference of 81 $^{\circ}$ R is 450 psi. The sensitivity of the carbide element to bundling friction is 0.103.

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(C-RD) The axial heat generation stress in the composite fuel element is 2759 psi. The 2.3σ variation in the heat generation stress due to 2.3σ variation in the material properties, component dimensions, and nuclear heating is 1100 psi.

(C-RD) The axial coating-matrix interaction stress in the matrix is calculated to be 323 psi based on the interaction stress assumptions given in Section A-5.1.6. The combined axial stress is 4272 psi. This is less than $0.8(\sigma_{ult} - 2.3\sigma_{us})$ minimum tensile strength of 4350 psi for the composite fuel element.

(U) The matrix-bead interaction stress in the matrix does not exist. There are no fuel beads.

(C-RD) The sensitivity of the composite element to a transverse temperature gradient is about 1500 psi per 100°R temperature difference across the flats of the element.

A-5.6 DESIGN CONCEPT 2.6 (SINGLE CHANNEL-CARBIDE-FUEL ELEMENT-DRAWING 947C817) (C-RD)

(C-RD) Fuel element alternate 2.6 is a single flow channel, hexagonal element made entirely of UC-ZrC. The fuel element is made up of 5.2-in. lengths, 0.170 in. across flats with a bore diameter of 0.120. The element is not coated.

A-5.6.1 Bundling Load (U)

(C-RD) Preliminary calculations indicate that bundling stresses are essentially compressive and are about equal to the bundling pressure. The single hole carbide element is capable of withstanding the maximum total hydraulic bundling load of 30 psia. However, if the bundling is increased and if very thin minimum webs are employed (less than 0.010 in.), additional bending and wall collapse (elastic stability) may become a problem.

A-5.6.2 Axial Load (U)

(C-RD) At present, it seems that the single hole elements necessitate a contacting hot end support system. It appears that the axial load can be transmitted in an appropriate manner to the support hardware, since contacting designs have been used before in the NRX reactors and these elements are short length, uncoated pieces.

A-5.6.3 Radial Expansion (U)

(C-RD) The radial expansion of the carbide fuel element is 1.62% as compared to an R-1 capability of accommodating 1.665%.

A-5.6.4 Corrosion (U)

(C-RD) At present, no braze joints are anticipated with single hole carbides. This implies that there are no cyclic degradation problems. No significant degradation of carbide element strength due to corrosion is expected. The peak incremental weight loss will not exceed that required to assure that the element remains sufficiently intact to support the imposed axial load and prevent ejection of parts.

A-5.6.5 Interface Requirements (U)

(C-RD) The single hole elements, per se, are not compatible with the present (reference) core periphery because of their small size. Compatibility with the periphery would depend on how the single hole elements are assembled or clustered. At present, the single hole elements are not compatible with the reference axial support design in that they necessitate a contacting hot end support system which eliminates the use of the present standoff supports. The present redundant forward end support is also not compatible with single hole elements if short lengths are employed.

(C-RD) The single hole carbide elements are not compatible with the present lateral support system for the following reasons. The increased core weight associated with the denser carbide fuel will require modification (stiffening) of the lateral support system. Also, the transverse CTE is about 20% higher than high expansion graphite which may also require some modification in the lateral support system.

A-5.6.6 Structural Criteria (U)

(C-RD) The single hole element does not meet the maximum stress structural adequacy criterion described in Section A-5.1.6 (see Table A-5.1).

(C-RD) The axial stress in the carbide element that is caused by bundling friction and a 2.3σ interelement temperature difference is 3300 psi. The sensitivity of the

carbide element to bundling friction is .62.

(C-RD) The axial heat generation stress in the carbide matrix is 10,650 psi. The 2.3% variation in the heat generation stress due to a 2.3% variation in material properties, component dimensions, and nuclear heating is 4250 psi.

(U) The carbide fuel is uncoated. Therefore, there is no coating-matrix interaction stress. The carbide fuel does not contain fuel beads. Therefore, fuel bead-matrix interaction does not occur.

(C-RD) The sensitivity of the carbide element to a transverse temperature gradient is about 7500 psi per 100°R temperature difference across the flats of the element.

A-5.7 DESIGN CONCEPT 2.7 (ZIG-ZAG COMPOSITE-DRAWING 947C807) (U)

(C-RD) Fuel element alternate 2.7 is a 19 hole composite fuel element with a zig-zag boundary. The boundary geometry is determined by the geometrical stacking of 19 small hexagons such that the distance between the mean lines of two opposite sides is 0.753 in. The reamed channel diameter is 0.102 in. and the average coating thickness is 2.2 mils of ZrC. The assumed coating distribution is shown in Figure A-1.3-10, with, however, the NbC portion replaced by ZrC.

A-5.7.1 Bundling Load (U)

(C-RD) The bundling load on a 37.1 in. fuel envelope is approximately 30 psi. This load produces negligible compressive stresses on the zig-zag composite fuel elements as compared to heat generation and coating-matrix interaction stresses.

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A-5.7.2 Axial Load (U)

(U) The axial load on the zig-zag composite element is maximum for the bearing or contact design and is approximately equal to the core ΔP . This axial stress is negligible when compared to the heat generation and coating-matrix interaction stresses.

A-5.7.3 Radial Expansion (U)

(C-RD) The radial expansion of the zig-zag composite is 1.54% as compared to an R-1 capability of accommodating 1.665%.

A-5.7.4 Corrosion (U)

(C-RD) The brazed joint strength problem is eliminated because no composite tips are required on this element. Degradation of element strength due to corrosion is substantially less in this element as compared to graphite fuel elements.

A-5.7.5 Interface Requirements (U)

(U) A core composed of composite fuel elements will be compatible with the present (reference) core periphery. The increased weight of a core of composite fuel requires modification of the reference axial support. The increased mass of a core of composite fuel requires modification of the lateral support system.

A-5.7.6 Structural Criteria (U)

(C-RD) The zig-zag composite fuel element does meet the maximum stress structural adequacy criterion described in Section A-5.1.6 (See Table A-5-1).

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(C-RD) The axial stress in the zig-zag composite element that is caused by bundling friction and a 2.3σ interelement temperature difference of 81°R is 450 psi. The sensitivity of the carbide element to bundling friction is 0.103.

(C-RD) The axial heat generation stress in the zig-zag composite fuel element is 2698 psi. The 2.3σ variation in the heat generation due to a 2.3σ variation in the material properties, component dimensions, and nuclear heating is 1160 psi.

(C-RD) The axial coating-matrix interaction stress in the matrix is calculated to be 259 psi based on the interaction stress assumptions given in Section A-5.1.6. The combined axial stress is 4199 psi. This is less than the $0.8 (\sigma_{\text{ult}} - 2.3 \sigma_{\text{us}})$ minimum tensile strength of 4350 psi for the 30 V/O UC-ZrC fuel element.

(U) The matrix-bead interaction stress in the matrix does not exist. There are no fuel beads in this element.

(C-RD) The sensitivity of the zig-zag composite element to a transverse temperature gradient is about 1500 psi per 100°R temperature difference across the flats of the element.

A-5.8 DESIGN CONCEPT 2.8 (HIGH VOID COMPOSITE FUEL - LONG CORE - DRAWING 947C814) (U)

(C-RD) Fuel element alternate 2.8 is a standard 19 hole fuel element composed of 30 V/O UC-ZrC. The reamed channel diameter is 0.108 and the average coating thickness is 2.2 mils of ZrC. The core is 37.1 in. in diameter and 62 in. long. The assumed coating distribution is shown in Figure A-1.3-10, with, however, the NbC portion replaced by ZrC.

A-5.8.1 Bundling Load (U)

(C-RD) The bundling load on a 37.1 in. fuel envelope will be approximately 30 psi. This load produces negligible compressive stresses on the composite fuel elements as compared to heat generation and coating-matrix interaction stresses.

A-5.8.2 Axial Load (U)

(U) The axial load on the composite fuel element is maximum for the bearing or contact design and is approximately equal to the core ΔP . This axial stress is negligible when compared to the heat generation and coating-matrix interaction stresses.

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A-5.8.3 Radial Expansion (U)

(C-RD) The radial expansion for this fuel element is 1.53% as compared to the R-1 capability of accommodating 1.665%.

A-5.8.4 Corrosion (U)

(C-RD) The brazed joint strength problem is eliminated because no composite tips are required on this element. Degradation of element strength due to corrosion is substantially less in this element as compared to graphite fuel elements.

A-5.8.5 Interface Requirements (U)

(U) A core composed of composite fuel elements will be compatible with the present (reference) core periphery. The increased weight of a core of composite fuel requires modification of the reference axial support. The increased mass of a core of composite fuel requires modification of the lateral support system.

(U) It is expected that the additional core length in itself will require no more than a simple extension of the periphery, axial support, and lateral support components.

A-5.8.6 Structural Criteria (U)

(C-RD) The high void composite fuel element does meet the maximum stress structural adequacy criterion described in Section A-5.1.6 (See Table A-5-1).

(C-RD) The axial stress in the element that is caused by bundling friction and a 2.3σ interelement temperature difference of 81°R is 570 psi. The sensitivity of the element to bundling friction is .126.

(C-RD) The axial heat generation stress in the fuel element is 2397 psi. The 2.3σ variation in the heat generation due to a 2.3σ variation in the material properties, component dimensions, and nuclear heating is 1030 psi.

(C-RD) The axial coating-matrix interaction stress in the matrix is calculated to be 61 psi based on the interaction stress assumptions given in Section A-5.1.6. The combined axial stress is 3630 psi. This is greater than the $0.8(\sigma_{ult} - 2.3\sigma_{us})$ minimum tensile strength of 4530 psi for the fuel element.

(U) The matrix-bead interaction stress in the matrix is not existent. There are no fuel beads.

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(C-RD) The sensitivity of the element to a transverse temperature gradient is about 1500 psi per 100°R temperature difference across the flats of the element.

A-5.9 DESIGN CONCEPT 2.9 (HIGH EXPANSION FUEL (AXM) LARGE CORE - DRAWING 947C815) (U)

(C-RD) Fuel element alternate 2.9 is a standard 19 hole fuel element of high expansion AXM fuel. The reamed diameter is 102 in., across flats is 0.753 and the coating is an average of 2.2 mils of NbC. The fuel envelope is 40.6 in. and the core length is 76 in. The assumed coating distribution is shown on Figure A-1.3-10.

A-5.9.1 Bundling Load (U)

(C-RD) The bundling load on a 40.6 in. fuel envelope is approximately 30 psi. This load produces negligible compressive stresses on the AXM fuel elements as compared to heat generation and coating-matrix interaction stresses.

A-5.9.2 Axial Load (U)

(U) The axial load on the AXM element is maximum for the bearing or contact design and is approximately equal to the core ΔP . This axial stress is negligible when compared to the heat generation and coating-matrix interaction stresses.

A-5.9.3 Radial Expansion (U)

(C-RD) The radial expansion of this element is 1.44% as compared to the R-1 capability of accommodating 1.665%.

A-5.9.4 Corrosion (U)

(C-RD) The corrosion capability of the AXM is expected to be unacceptable from an endurance and cyclic standpoint.

A-5.9.5 Interface Requirements (U)

(U) The increased core diameter will require modification of the reference core periphery. The increased core length requires modification of the reference axial support. The increased size of the core requires modification of the lateral support system.

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A-5.9.6 Structural Criteria (U)

(C-RD) This large core fuel element does meet the maximum stress structural adequacy criterion described in Section A-5.1.6 (See Table A-5-1).

(C-RD) The axial stress in the AXM element that is caused by bundling friction and a 2.3σ interelement temperature difference of 81°R is 730 psi. The sensitivity of the AXM element to bundling friction is 0.370.

(C-RD) The axial heat generation stress in the AXM fuel element is 604 psi. The 2.3σ variation in the heat generation due to a 2.3σ variation in the material properties, component dimensions, and nuclear heating is 380 psi.

(C-RD) The axial coating-matrix interaction stress in the matrix is calculated to be 441 psi based on the interaction stress assumptions given in Section A-5.1.6. The combined axial stress is 1867 psi. This is less than the $0.8 (\sigma_{\text{ult}} - 2.3 \sigma_{\text{us}})$ minimum tensile strength of 1970 psi for the fuel element.

(C-RD) The matrix-bead interaction stress in the matrix is negligible when compared to the heat generation stress and the coating-matrix interaction stress. The matrix-bead interaction stress occurs at temperatures greater than the graphitization temperature. If the graphitization temperature is greater than the maximum fuel temperature, no bead-matrix interaction occurs. In R-1, the graphitization temperature is assumed to be about 4450°R as compared to a maximum fuel temperature of 4460°R .

(C-RD) The sensitivity of the element to a transverse temperature gradient is about 400 psi per 100°R temperature difference across the flats of the element.

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TABLE A-5-1

STRUCTURAL ANALYSIS RESULTS OF 9 FUEL ELEMENT ALTERNATES

	(1)		(2)		(3)		(4)		(5)		(6)		(7)		(8)**		(9)**			
	19-Hole Std.		19-Hole Std.		Hybrid 19-Hole Std.		37-Hole Hybrid		30 V/O UC-ZrC-C 19-Hole Std.		Single Channel Carbide		30 V/O UC-ZrC-C Zig-Zag		30 V/O UC-ZrC-C High Void Long Core 62"		AXM-19 Hole Std. Large Core 40.6" Dia x 76" Long			
Structural					(AXM Limited)		AXM		30 V/O UC-ZrC-C											
Axial Matrix Stress	36		36		35		35		35		36		38		36		41			
Combined Stress	nom 2.3σ		nom 2.3σ		nom 2.3σ		nom 2.3σ		nom 2.3σ		nom 2.3σ		nom 2.3σ		nom 2.3σ		nom 2.3σ			
Max. at STA																				
Friction (psi) (2.3σ ΔT = 81.4°R)	0.	500*	0	500*	0	525*	0	525*	0	525*	0	450*	0	3300*	0	450*	0	570*	0	730*
Heat Generation (psi)	488.	307*	821	518*	815*	514*	342	215*	1275	548*	2759	1100*	10650	4250	2698	1160	2397	1030	604	380*
Coating-Matrix Inter- action (psi) (Begins at ZrC dep. Temp)	1470		501		455		869		702		323				259		61		441	
Total nom (psi)	1958	585	1322	720	1270	735	1211	566	2067	711	3082	1190	10650	5390	2957	1242	2450	1172	1045	822
Combined Stress (psi)	2543		2042		2005		1777		2788		4272		16040		4199		3630		1867	
Allowable 0.8 (2.3σ min. Tensile Str.) psi	2200		1950		1940		1940		4350		4350		5300		4350		4530		1970	
Radial Expansion (%) (Calculated as the Product of Fuel CTE and Fuel Tempera- ture Less 540°R at the Station of Maximum Fuel Temperature)	1.46		1.48		1.54				1.505		1.54		1.62		1.54		1.53		1.44	
Sensitivity to Bundling Friction (Ratio of Friction Stress/Allowable Str.)	0.227		0.256		0.280		0.280				0.103		0.62		0.103		0.131		0.370	

* Statistically combined.

** Estimated stresses. Detailed analysis in progress

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APPENDIX A-6
RELIABILITY AND QUALITY ASSURANCE SUPPORTING DATA
TRADE STUDY FOR NERVA FUEL ELEMENT
DESIGN CONCEPT
TRADE STUDY NO. 759

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A-6a RELIABILITY SUPPORTING DATA

A-6a.0 (U) INTRODUCTION - Reliability Considerations in the Long Life Reactor Fuel Element Trade Study

(U) Reliability is listed as a functional requirement (Section 1.3.3) for the purposes of this trade study. In reality "Reliability" is an expression of the probability of fulfillment of the other functional and technical requirements, listed in Sections 1.3 and 1.4, to the extent that mission requirements are met. The manner in which these requirements might fail to be met is the appropriate concern of reliability analysis in general, and of failure mode analysis in particular. Thus, the following discussion will review the significant failure modes and mechanisms applicable to the various design concepts.

(U) It will be useful to appreciate the complex way in which the functional requirements and the failure modes/mechanisms in the fuel elements are interrelated:

(CRD) Fuel elements employing a bead loaded graphite matrix are subject to corrosion, which is most severe at the hot end. The flow channels of such elements are coated with carbide to impede the corrosion mechanism, i.e., the reaction of carbon from the matrix with hydrogen. Significant disruption of this coating is thus an important failure mode, having as its effect severe corrosion. Corrosion of the uncoated (hex-) flat surfaces of the elements is normally limited by the very low flow rates of core interstitial gases, which thus tend to be saturated with carbon. Breakage of fuel elements in a way to allow fresh (unsaturated) hydrogen to flow from the bores to the core interstices would drastically alter that situation. (This could be construed as a failure to meet the flow characteristics of Section 1.3.2). While the structural failure in itself could be tolerated within limits, thanks to the redundant support provided by bundling and/or CHESH, the severe corrosion which would result might compromise the criticality, reactivity, and limited corrosion requirements of Sections 1.3.1, 1.4.5, and 1.4.6, respectively.

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(U) The requirements of load support capabilities specified in Sections 1.4.1 and 1.4.2 are expected to be compromised if elements are weakened by extensive corrosion. We must await data on strength measurements of corroded elements, however, before we can say whether breakdown of these requirements is more significant than the corrosion which must occur to cause it. With effective redundant support, the prime result of such breakdown would be increased corrosion.

(U) Because of the complex interrelations between the failure mechanisms just described and the requirements of Sections 1.3 and 1.4, the following discussion will not address itself to those requirements on a paragraph by paragraph basis under each concept. Instead, the requirements concerned in the discussion under each concept are indicated parenthetically in the respective heading.

(U) Alternates 1 through 9 are treated in Sections A-6a.1 through A-6a.9, respectively. The key points of comparison are summarized in Table A-6.1 and conclusions are presented in Section A-6a.10. (Table A-6.1 is presented in lieu of detailed failure mode analyses, since it is not practical to complete such FMA's for conceptual studies.)

A-6a.1 (U) Design Concept 2.1 - Intermediate CTE Graphite - Standard Geometry
(Discussion pertinent to requirements of Sections 1.3.1, 1.4.5 and 1.4.6)

(CRD) The most serious failure mode for this fuel element is reactivity loss through corrosion. The most important mechanisms contributing to this are hot end diffusion and degradation of the coating which is the main barrier to corrosion at the hot end. Elements having an intermediate CTE matrix would be expected to suffer more severe midband corrosion, such as was experienced in the NRX-A5 reactor, on account of gaps in the coating due to coating-matrix thermal expansion mismatch. General degradation of the coating at the hot end is expected from increased cycling. In view of the long duration requirement, this would render this design concept unacceptable.

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A-6a.2 (U) Design Concept 2.2 - High CTE Graphite - Standard Geometry
(Discussion pertinent to requirements of Sections 1.3.1, 1.4.5 and 1.4.6)

(CRD) High CTE matrices provide improved compatibility between matrix and coating at the expense of thermal shock resistance (which is inversely proportional to CTE). Considering the hexagonal 0.753 inch-cross-flats standard 19 hole design with a 66.4 BTU/in³-sec power requirement, if a high CTE matrix is used, the principle part of the reactivity loss could be attributable to corrosion along the interfaces of cracks caused by thermal stress. On the other hand, experimental data from recent electrical corrosion tests (December, 1969) shows that, even with high CTE fuel, the rate of corrosion weight loss increases several fold after six ten-minute operational cycles, indicating that significant degradation of the coating has occurred at the hot end. This might rule out the present matrix-coating combinations from consideration for the ten hour reactor. It should be noted that thermal stress cracking of the type described here is not observed in electrical corrosion tests because of the more uniform axial power distribution and absence of element to element interactions there as compared to the situation in the reactor.

A-6a.3 (U) Design Concept 2.3 - Hybrid Fuel Element - Standard Geometry
(Discussion pertinent to requirements of Sections 1.3.1, 1.3.2, 1.4.5 and 1.4.6)

(CRD) The hybrid fuel with the 19 hole configuration appears to retain a sensitivity to thermal stress cracking near midband since a high CTE matrix is used at the forward end to avoid both a compatibility problem at the braze joint and midband corrosion. With an aft end support system, failure of the braze joint would not be serious from the structural point of view, but would be quite serious if the presence of unsaturated hydrogen at the interstices is unacceptable (e.g., because of corrosion of graphite parts of the periphery). The same serious consequences could occur if cracks originating from thermal stress were to propagate from the bores to the hex flat surfaces, as is the case with all the alternates.

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A-6a.4 (U) Design Concept 2.4 - 37 Hole Hybrid Fuel Element
(Discussion pertinent to requirements of Sections 1.3.1, 1.3.2, 1.4.5, 1.4.6, 1.4.10)

(CRD) The hybrid fuel with the 37 hole configuration appears to alleviate the thermal stress difficulty. The reduced pitch of the flow channels should reduce the heat-generation-induced thermal stress to an acceptably low value even with the use of the high CTE matrix at midband, so that the risks of midband corrosion, thermal stress cracking, and braze joint failure are all significantly reduced. The major degradation mechanisms of the hybrid element with 37 hole configuration would appear to be surface and intergranular corrosion. While the corrosion rates (for uncoated material) in the graphite-solid solution composite used at the aft end are lower* than those in the historical graphite based matrix, the composites made so far are more porous than the latter matrix and this could result in an increased partial pressure of unsaturated hydrogen at the interstices. The excellent compatibility of the ZrC coating with the composite* is expected to offer adequate protection against hot end bore corrosion despite the increased bore surface area of the 37 hole configuration. There is some concern, however, that the thin web essential to this configuration might be seriously weakened by the corrosion normally expected to occur. This concern may be mitigated by the following considerations: (a) Coating defects, and therefore concentrated pockets of corrosion, are not likely to occur, so strength will not be lost rapidly, (b) the carbide network maintains some residual strength (~2,000 psi)* even when all of the graphite has been leached out of the composite, and (c) the stresses become lower towards the hot end where the weakening is greatest, thus the loss of strength there may be tolerable. At any rate, it is clear that we must obtain strength data on elements that have been through the corrosion test before we can dismiss web failure from further consideration.

*As indicated by preliminary tests at LASL.

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(CRD) Use of a slightly larger cross-flats dimension expected for this design entails an increase in the distance scale over which element to element loading adjustments are made, so that some increase in temperature gradients at the periphery could result, although this could be offset somewhat by orificing adjustments.

A-6a.5 (U) Design Concept 2.5 - All-Composite Fuel Element
(Discussion pertinent to requirements of Sections 1.3.1, 1.3.2, 1.4.5, 1.4.6 and 1.4.8)

(CRD) A fuel element consisting of a graphite-solid solution composite over its entire length has the advantage of complete freedom from joint failures (although it carries some weight penalty). Material properties data indicate that there would be a somewhat lower risk of thermal stress cracking than that expected when using the high CTE graphite matrix in the peak power zone. Some preliminary Q test results indicate that the 30 weight percent solid solution composite has lower thermal stress resistance than that deduced from its thermal and mechanical properties. This circumstance would have to be understood before any quantitative reliability predictions can be made. The full length composite element would be somewhat less desirable than the hybrid element with respect to any effects of the higher porosity of the composite. I.e., one would expect to find a higher partial pressure of unsaturated hydrogen in the interstices. The all-composite 19 hole design would not appear to be desirable because of the risk of thermal stress cracking. (An all composite element with a 37 hole configuration would seem to be acceptable from a reliability point of view, provided that the associated interstitial hydrogen does not result in unacceptably high corrosion rates for the periphery, and that the web maintains adequate strength, despite intergranular corrosion.)

(CRD) It is worth noting here one feature of the all-composite concept which would offer some reliability advantage in the core periphery if the porosity problem is overcome. This is its greater dimensional stability by virtue of the higher temperatures at which its final

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heat treatment can be carried out. (A matrix containing fuel beads cannot be given stabilizing heat treatments above 2400°C, while the composites can endure 2600°C.) Improved dimensional stability in the core allows the periphery to better maintain high impedance to axial flow, thanks to the more stable interface.

A-6a.6 (U) Design Concept 2.6 - All-Carbide Fuel Element
(Discussion pertinent to requirements of Sections 1.3.1, 1.3.2, 1.4.5, 1.4.6, 1.4.8, 1.4.10)

(CRD) The concept employing an element with no free graphite but a solid solution of uranium and zirconium carbides over the entire length appears attractive because corrosion of the fuel elements would be virtually eliminated. The solid solution of carbides has been shown to be rather stable in hydrogen at the temperatures of interest although some reaction could occur under reactor conditions. Further experimental data is needed in this area. To minimize the probability of thermal stress cracking, these fuel elements require a new geometry. This in turn requires a new core support design, which for the time being would introduce a host of new reliability problems. A key problem is that of flow channel misalignment between axial segments at assembly and the resulting flow and temperature perturbations. On the other hand one has the possibility of optimizing the axial loading distribution. All-carbide fuel also has the advantage of dimensional stability discussed in the last section. The porosity problems would be replaced by problems of leakage of propellant from the flow channels to the interstices at the joints. A major shortcoming of this design hinges on the consequences of thermal stress cracking. Should such cracking occur in any element, broken sections of wall could become lodged in a manner to partially obstruct the flow channel. A number of such occurrences would lead to serious perturbations of flow, pressure, and temperature in the core, which could result in considerable further damage.

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A-6a.7 (U) Design Concept 2.7 - Zig-Zag Composite
(Discussion pertinent to requirements of Sections 1.3.1, 1.3.2, 1.4.5, 1.4.6, and 1.4.8)

(CRD) The considerations with respect to this concept are the same as those discussed under Section A-6a.5 with the exception of a slight improvement in the thermal stress performance expected by virtue of modification of the pitch in the outer ring of flow channels. On the other hand, there is likely to be greater interstitial flow with this design because of the difficulty of matching the zig-zag interfaces. This disadvantage appears to outweigh the slight decrease in risk of cracking from thermal stress as compared to Concept 5.

A-6a.8 (U) Design Concept 2.8 - High Void Composite with Long Core
(Discussion pertinent to requirements of Sections 1.3.1, 1.3.2, 1.4.5, 1.4.6, and 1.4.8)

(CRD) This concept was conceived to overcome the thermal stress cracking problem which was indicated as presenting a high reliability risk in the all composite element of conventional geometry discussed in Section A-6a.5. The thermal stress is reduced by the order of 20% through the use of the larger bore diameter and lower power density. The concern over the thinner web is the same as that discussed under Concept No. 4 (the 37 hole hybrid element).

A-6a.9 (U) Design Concept 2.9 - Standard High CTE Graphite Element in the Large Core
(Discussion pertinent to requirements of Sections 1.3.1, 1.4.5 and 1.4.6)

(CRD) By virtue of the large core dimensions associated with this concept, the power density will be lower and therefore the risk of thermal stress cracking reduced. On the other hand the problem of hot end corrosion, as discussed under Section A-6a.2 would remain. This would probably rule out this concept from consideration for the 10 hour reactor.

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A-6a.10 (U) Conclusions

(CRD) In the light of the foregoing discussion, the high void composite element appears to offer the most promise for meeting reliability goals, with the 37 hole hybrid design a close second. Both designs reduce the risk of thermal stress cracking to an acceptably low level by virtue of their relatively larger surface area for the flow channel walls. This increases the corrosion related reactivity losses somewhat, the more so in the 37 hole configuration. However, the improved compatibility of the ZrC coating with the composite matrix ultimately expected should allow this corrosion loss to be reliably limited. The ranking of the nine design concepts discussed above from the point of view of Reliability, only, is presented in the lower right of Figure A-6.1, where the failure modes/mechanisms and the criticality of their effects for the various alternates are qualitatively compared.

The overall failure estimates, the overall severity estimates, and the emphasis factors used in Table 3-1 (Reliability Considerations, Comparison Matrix) were derived from the probabilities of failure listed in Figure A-6.1 as follows: For each concept, the probabilities of occurrence of failure via the tabulated (dominant) failure modes were summed to give an overall failure estimate. An overall severity estimate was obtained by multiplying the probability of occurrence of the various failure modes by a weighting factor, according to the relative severity category in each case, and totalling the weighted probabilities. The weighting factors used are .6, .3, .1 and 0 for severity Categories A, B, C and D, respectively. The emphasis factors are related to the logarithms of the reciprocals of the respective overall failure or severity estimates, so as to obtain the numerically highest ratings for the concepts having lowest probabilities of failure and lowest severity estimates. The emphasis factors are obtained from the logarithms by subtracting a common integer from the logarithm associated with each concept, so as to make the lowest resulting term fall between 0 and 1. Each term is then divided by the sum of the terms for all concepts to yield the normalized emphasis factor.

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A-6b QUALITY ASSURANCE SUPPORTING DATA

(U) A-6b.0 INTRODUCTION - Quality Engineering Evaluation of Fuel Element Conceptual Trade Study

The following is the Quality Engineering evaluation of the fuel element conceptual design layout drawings obtained by Quality Assurance on January 6, 1970. In general, the detail available on these layout drawings is not sufficient to permit a qualitative or quantitative (R_Q) rating. The applicable detail FMECA's are not available which further complicates any meaningful ranking endeavor. The evaluations supplied are a narrative description of anticipated quality problems concentrating on the areas of nondestructive testing, specifications and inspectability where sufficient detail exists to make these judgments.

(U) A-6b.1 Intermediate Expansion Graphite Element - 947C812

Determining homogeneity of loading within the matrix as well as uniformity of coating application continues to present inspection problems. Work must be done to refine these inspection procedures from the standpoint of more accurate determination of the absolute values for these variable characteristics.

Maintaining coating integrity throughout the assembly and testing operations is difficult to achieve due to the ease with which handling damage may occur.

(U) A-6b.2 High Expansion Graphite Element - 947C813

This alternative, having the same geometry and basic matrix, presents no additional identifiable inspection problem over alternate A-6b.1.

(CRD) A-6b.3 Hybrid Element, 19 hole - 947C810

Understanding the variables involved in brazing composite to graphite to secure a joint that will withstand cyclic conditions must be achieved. Capability to determine fuel

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loading values in two adjacent dissimilar matrix must be developed. Additional NDT techniques for evaluation of matrix and coating integrity need development.

(CRD) A-6b.4 Hybrid Element, 37 hole - 947C816

Comments for 19 hole hybrid elements apply. In addition, the complexity introduced by the additional holes increases the probability of rejection due to discrepancies in channel coating. Increase in the number of holes presents more opportunities for rejection for coating and dimensional characteristics. NDT techniques would have to be developed for this geometry.

(U) Substantial modification and revision of all inspection equipment will be required to provide the capability to inspect elements having this size cross section. The change in bore size requires modifications of all NDT equipment. New indexing heads on all existing inspection devices will be needed to be compatible with the increased number of bores. In addition to this, the data system must be completely reprogrammed to accept the additional data points resulting from the increase in the number of bores for this alternative.

(U) Corrosion test facilities will require a complete rebuild to accept this element configuration.

(CRD) A-6b.5 All Composite Element, 19 hole - 947C809

Inspection capability in the NDT areas of x-ray, beta backscatter and gamma counting require development for an all composite element. The well known standard geometry should present no additional problems for the dimensional inspection requirements.

(U) A-6b.6 Carbide, Single Channel, Element - 947C817

Dimensional inspection should present no real problems. The requirement of true position within .012 inch will permit rather severe projections within the flow channel from one element segment to the next segment. The assembly problems as far as stacking would

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require development. The absence of coating should reduce the losses due to handling damage. NDT procedures for this geometry require development.

(U) The quantity of pieces needed for a total core would require the duplication of many pieces of inspection and test equipment to avoid a production delay at inspection.

(U) Corrosion test facilities will require extensive modification.

(U) A-6b.7 Zig-Zag Composite - 947C807

A full scale development program for inspection of machined characteristics as well as NDT techniques to be applicable to this geometry would require development. Fitup requirements to reduce interstitial H_2 flow would require extremely tight dimensional tolerances which, in turn, compound inspection difficulties.

(U) A-6b.8 High Void Composite - 947C814

The high void feature in itself should not present inspection problems. NDT methods and techniques would require development to correlate the composite and high void to some known standard values. Inspection of dimensional characteristics should be rather straightforward. The additional length may pose problems if existing inspection equipment is to be made applicable to this design.

(U) Corrosion test facilities will require modification to accept the additional element length.

(U) A-6b.9 High Expansion Graphite, Long - 947C815

This alternate, being similar to alternate A-6b.2, except for additional length, should present no additional problems other than those associated with the increased length. Assembly techniques and fixturing would all have to be revised to accommodate the increase in element length.

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(U) Facilities may not be adequate when adapting existing inspection techniques to elements of this length. All handling equipment would probably require modification as well as all storage vaults to permit acceptance of the additional length.

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FIGURE A-6.1 THE PRINCIPLE FAILURE MODES/MECHANISMS; THEIR PROBABILITY OF OCCURRENCE AND SEVERITY OF EFFECT

<div> <div>Concepts</div> <div>Failure Modes Mechanisms</div> </div>	1	2	3	4	5	6	7	8	9
	Standard Geometry Intermediate Expan-Graphite	Standard Geometry High Expansion Graphite	Standard Geometry Hybrid	37 Hole Hybrid	Standard Geometry Composite	Single Channel Carbide	Zig Zag Composite	High Void Composite Long Core	Standard Geometry Graphite Large Core
Reactivity Loss Via Excessive Hot End Corrosion									
a. By gross coating degradation	10^{-2} C	10^{-2} C							10^{-2} C
b. By diffusion			10^{-3} C	2×10^{-3} C	10^{-3} C	10^{-6} C	10^{-3} C	1.5×10^{-3} C	
Corrosion at Cracks in Channel Coating ("Mid Band" Type)	10^{-1} C	10^{-4} C	10^{-4} C	10^{-4} C	10^{-4} C	N.A.	10^{-4} C	10^{-4} C	10^{-4} C
Corrosion at Interfaces of Thermal Stress Cracks in Matrix	10^{-4} C	10^{-2} C	10^{-2} C	10^{-3} C	10^{-2} C	10^{-6} C	10^{-2} C	10^{-3} C	10^{-3} C
Surface Corrosion	10^{-3} C	10^{-3} C	10^{-4} C	10^{-4} C	10^{-4} C	10^{-6} C	2×10^{-4} C	10^{-4} C	10^{-3} C
Intergranular Corrosion (weakening due to loss of binder carbon)	10^{-3} C	10^{-3} C	10^{-3} C	2×10^{-3} C	10^{-3} C	N.A.	10^{-3} C	1.5×10^{-3} C	10^{-3} C
Channel to Surface Breakthrough Via Thermal Stress Cracking*	10^{-5} C	10^{-3} C	10^{-3} C	$3 \times 10^{-4**}$ C	3×10^{-4} C	10^{-2} B***	3×10^{-4} C	10^{-4} C	10^{-4} C

Calibration of Qualitative Judgement for Estimate of Probability of Occurrence

Highly Probable $\sim 10^{-1}$
Moderately Probable $\sim 10^{-2}$
Not Unlikely $\sim 10^{-3}$
Unlikely $\sim 10^{-4}$
Very Unlikely $\sim 10^{-5}$
Extremely Unlikely $\sim 10^{-6}$

*Graphite periphery endangered by unsaturated interstitial hydrogen

**Thermal stress cracks unlikely, but cracking at the braze joint is considered here

***Fragmentation from thermal stress cracking can obstruct flow channels and perturb flow, pressure and temperature distribution

Severity of Effect Categories

Category	Definition
A	"Catastrophic" - Complete loss of NSS capability
B	"Major" - Substantial degradation of NSS capability; prompt action required to prevent catastrophic effects
C	"Minor" - Slight degradation of NSS capability; probability of further degradation increased; action may be required
D	"Negligible" - No loss of NSS capability

Ranking In Order of Reliability

1st-Concept No. 8 - High Void Composite - Large surface area of bores gives more corrosion, webs may weaken
2nd-Concept No. 4 - 37 Hole Hybrid - Large surface areas of bores gives more corrosion, webs may weaken; potential braze joint failure
3rd-Concept No. 5 - All-Composite - Subject to thermal stress cracking
4th-Concept No. 7 - Zig Zag Composite - Subject to thermal stress cracking; surface corrosion at bundling gaps
5th-Concept No. 3 - 19 Hole Hybrid - Subject to thermal stress cracking plus potential braze joint failure
6th-Concept No. 9 - High CTE Graphite Large Core - Severe hot end corrosion results from coating degradation
7th-Concept No. 6 - Single Channel Carbide - Possible flow channel obstruction by fragments dislodged via thermal stress cracking
8th-Concept No. 2 - High CTE Standard Geometry Graphite - Severe corrosion results from hot end coating degradation and from moderately probable thermal stress cracking of matrix
9th-Concept No. 1 - Intermediate CTE - Severe corrosion at coating defects from mismatch of CTE with matrix

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APPENDIX A-7

MATERIALS SUPPORTING DATA
TRADE STUDY FOR NERVA FUEL ELEMENT
DESIGN CONCEPTS
TRADE STUDY NO. 759

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A-7 Materials Supporting Data (U)

(U) Materials supporting data for fuel element trade studies generally are considered to be those materials properties which are needed to assess the design adequacy of a given alternate design concept. The materials properties of most significance are those attendant to compositional variations. In the present trade study, four (4) compositional vintages are under consideration, i.e., 1) beaded fuel in intermediate expansion graphite, 2) beaded fuel in high expansion graphite, 3) UC-ZrC solid solution in graphite composite, and 4) UC-ZrC solid solution. In general, materials property data for the UC-ZrC solid solution vintage is non-existent for the reference design. A summary comparison of the materials properties significant in assessing design adequacy is given for the other three (3) compositional vintages in Table A-7.10.1.

(U) For the rest of this appendix, each alternate fuel element design concept (reference Section 2.0) has been reviewed with respect to a limited number of technical requirements which are believed to be pertinent. No attempt has been made in this appendix to discuss the materials supporting data used in assessing thermal stress stability since the data and calculations have been covered adequately in Appendix A-4. Likewise, the status of corrosion test data with respect to reactivity and corrosion losses are not discussed in this appendix since this subject is discussed in detail in Appendix A-4.

(U) The pertinent requirements and types of material data considered in support of these requirements are as follows:

Bundling load - flexure strength and elasticity data

Axial load - flexure and compressive strength data

Radial expansion - thermal expansion data

Environmental resistance - hydrolysis data

(U) For convenience, the status of the materials supporting data for each requirement and each alternate design concept is given by the following code:

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Code A - Data available to meet specifications

Code B - Limited data available. Additional data effort needed to meet specifications.

Code C - Data not available. Tests and/or data to be developed when specification issued.

(U) Where applicable, references are given to additional materials property data shown in this Appendix, A-7.10.

(U) A-7.1 Standard Geometry - Intermediate Expansion Graphite (947C812)

(U) A-7.1.1 Bundling load (Para. 1.4.1)

(U) Code A - Fuel element strength levels set forth in specifications have been verified through flexure testing and modulus measurements (see Table A-7.10.1).

(U) A-7.1.2 Axial load (Para. 1.4.2)

(U) Code A - Fuel element strength levels set forth in specifications have been verified through tests for flexure and compressive strengths (see Table A-7.10.1).

A-7.1.3 Radial expansion (Para. 1.4.3)

(C-RD) Code A - The transverse and longitudinal expansion requirements have been verified by measurements between 25°C and 2000°C as required by specification (see Table A-7.10.1).

(U) A-7.1.4 Environmental resistance (Para. 1.4.12)

(U) Code B - Accelerated hydrolysis testing has shown elements of A6 vintage to be somewhat sensitive to hydrolysis. Additional study needed on effects of coatings, leaching, and fuel bead stability.

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(U) A-7.2 Standard Geometry - High Expansion Graphite (947C813)

(U) A-7.2.1 Bundling load (Para. 1.4.1)

(U) Code A - Fuel element strength levels set forth in specifications have been verified through flexure testing and modulus measurements (see Table A-7.10.1).

(U) A-7.2.2 Axial load (Para. 1.4.2)

(U) Code A - Fuel element strength levels set forth in specifications have been verified through tests for flexure and compressive strengths (see Table A-7.10.1).

(U) A-7.2.3 Radial expansion (Para. 1.4.3)

(C-RD) Code A - The transverse and longitudinal expansion requirements have been verified by measurements between 25°C and 2000°C as required by specification (see Table A-7.10.1).

(U) A-7.2.4 Environmental resistance (Para. 1.4.12)

(U) Code B - Accelerated hydrolysis testing has shown elements of A6 vintage to be somewhat sensitive to hydrolysis. Additional studies needed.

(U) A-7.3 Standard Geometry - Hybrid (947C810)

(U) A-7.3.1 Bundling load (Para. 1.4.1)

(U) Code B - Specifications needed for joint limits and additional testing needed for composite. See comments A-7.5.1.

(U) A-7.3.2 Axial load (Para. 1.4.2)

(U) Code B - Additional testing needed for composite fuel (see comments A-7.5.2).

(U) A-7.3.3 Radial expansion (Para. 1.4.3)

(U) Code A - Data indicates requirements could be met in both

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composite and graphite, hybrid effect believed to be not pertinent (see comments A-7.5.3 for thermal expansion data on composite fuel).

(U) A-7.3.4 Environmental resistance (Para. 1.4.12)

(U) Code B - Accelerated hydrolysis testing at LASL has shown composite elements to be insensitive to hydrolysis. The graphite-composite element would be expected to require some data effort to evaluate joint-braze effect.

(C-RD) A-7.4 37-Hole Hybrid (947C816)

(U) A-7.4.1 Bundling load (Para. 1.4.1)

(U) Code C - Specifications need to be developed considering geometry implications and presence of joint. Additional data effort needed on composite fuel (see A-7.5.1).

(U) A-7.4.2 Axial load (Para. 1.4.2)

(U) Code C - Specifications need to be developed considering geometry implications. Additional data needed on composite fuel (see A-7.5.2).

(U) A-7.4.3 Radial expansion (Para. 1.4.3)

(U) Code B - No data available on this geometry, but expansion data on composite and graphite sections indicates no significant problems.

(U) A-7.4.4 Environmental resistance (Para. 1.4.12)

(U) Code B - No data available, however, geometry considerations will probably not effect hydrolysis sensitivity beyond that described in A-7.3.4.

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- (U) A-7.5 Standard Geometry - Composite (947C809)
- (U) A-7.5.1 Bundling load (Para. 1.4.1)
- (U) Code B - While specifications are not available for the 700 mgms/cc
aim loadings, the current LASL flexure and modulus data for variable
loadings (see Tables A-7.10.1 and A-7.10.2 and Figure A-7.10.1)
indicate that strength requirements can probably be achieved.
Additional data effort needed, however, as a function of loading.
- (U) A-7.5.2 Axial load (Para. 1.4.2)
- (U) Code B - Comments given in A-7.5.1 apply. Compressive strength
data obtained at LASL is given in Tables A-7.10.1 and A-7.10.2.
Additional data effort needed for 700 mgms/cc loading.
- (U) A-7.5.3 Radial expansion (Para. 1.4.3)
- (U) Code B - LASL data on the expansion characteristics of composite
fuel with respect to volume percent carbide variation, anisotropy
and fuel loading variation indicate that thermal expansion require-
ments can be met (see Tables A-7.10.1, A-7.10.3 and A-7.10.4
and Figures A-7.10.2 and A-7.10.3). Additional data effort
needed, however.
- (U) A-7.5.4 Environmental resistance (Para. 1.4.12)
- (U) Code B - Accelerated hydrolysis testing at LASL has shown the
composite elements to be relatively insensitive to hydrolysis.
Additional data study needed on higher fuel loadings.
- (U) A-7.6 Single Channel - Carbide (947C817)
- (U) A-7.6.1 Bundling load (Para. 1.4.1)
- (U) Code C - Specifications not available nor strength data on 2000
mgms/cc loadings.

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- (U) A-7.6.2 Axial load (Para. 1.4.2)
- (U) Code C - Specifications not available nor strength data on 2000 mgms/cc loadings.
- (U) A-7.6.3 Radial expansion (Para. 1.4.3)
- (U) Code C - Specifications not available nor expansion data on 2000 mgms/cc loadings.
- (U) A-7.6.4 Environmental resistance (Para. 1.4.12)
- (U) Code C - No data available; it is believed that this material will be more resistant to hydrolysis than the composite, however, based on limited LASL studies.
- (U) A-7.7 Zig-Zag Composite (947C807)
- (U) A-7.7.1 Bundling load (Para. 1.4.1)
- (U) Code C - Specifications need to be developed considering geometry implications. See comments A-7.5.1 for composite fuel strengths.
- (U) A-7.7.2 Axial load (Para. 1.4.2)
- (U) Code C - Specifications need to be developed. See comments A-7.5.2 for composite fuel strengths.
- (U) A-7.7.3 Radial expansion (Para. 1.4.3)
- (U) Code B - See comments A-7.5.3.
- (U) A-7.7.4 Environmental resistance (Para. 1.4.12)
- (U) Code B - No data available on this geometry, but composite fuel has shown insensitivity to hydrolysis as described in Para. A-7.5.7.

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- (U) A-7.8 High Void - Composite - Long Core (947C814)
- (U) A-7.8.1 Bundling load (Para. 1.4.1)
- (U) Code B - See comments A-7.5.1. Effect of hole size needs some additional study.
- (U) A-7.8.2 Axial load (Para. 1.4.2)
- (U) Code B - Specifications not available. See comments A-7.5.1 and A-7.8.1.
- (U) A-7.8.3 Radial expansion (Para. 1.4.3)
- (U) Code B - See comments A-7.5.3.
- (U) A-7.8.4 Environmental resistance (Para. 1.4.12)
- (U) Code B - See comments A-7.5.4, added length and hole size should not influence hydrolysis.
- (U) A-7.9 Standard Geometry - Graphite - Large Core (947C815)
- (U) A-7.9.1 Bundling load (Para. 1.4.1)
- (U) Code A - See Para. A-7.2.1.
- (U) A-7.9.2 Axial load (Para. 1.4.2)
- (U) Code A - See Para. A-7.2.2.
- (U) A-7.9.3 Radial expansion (Para. 1.4.3)
- (U) Code A - See Para. 7.2.3.
- (U) A-7.9.4 Environmental resistance (Para. 1.4.12)
- (U) Code B - See Para. A-7.2.7.

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TABLE A-7.10.1

PROPERTIES OF SOLID SOLUTION UC-ZrC COMPOSITE FUEL ELEMENTS COMPARED TO
INTERMEDIATE AND HIGH EXPANSION GRAPHITE FUEL ELEMENTS

<u>Property</u>	<u>Graphite</u>	<u>Carbide Components</u> ⁽³⁾
1. Flexure Strength (RT)	3,500 to 5,000 psi	8,000 to 9,000 psi
Flexure Strength (2000°C)	7,000 to 10,000 psi ⁽¹⁾	8,000 to 9,000 psi
2. Compressive Strength (RT)	14,000 to 16,000 psi	17,000 to 19,000 psi
Compressive Strength (2000°C)		17,000 to 19,000 psi
3. Modulus (psi)	1.0 to 2.0 x 10 ⁶⁽²⁾	8 x 10 ⁶
4. CTE - Transverse (2000°C) (in/in/°C x 10 ⁻⁶)	AXM - 5.7 to 6.4 GL - 4.3 to 5.3	6.95
CTE - longitudinal (2000°C) (in/in/°C x 10 ⁻⁶)	AXM - 5.6 to 6.4 GL - 3.5 to 4.6	6.4
5. Thermal Stress (BTU in ³ /sec)	GL - 120 AXM - 85	10 v/o 120 - 200 20 v/o 120 - 150 25 v/o 90 - 110 30 v/o 76 - 84 50 v/o 26 - 34
6. Leak Rate (cc/min)	Less than 170 with correct flour	300 to 500 mg/cc - 300 100 to 200 mg/cc - 200
7. Thermal Conductivity (BTU/hr.ft-°F)	60 - 120 (RT) 12 - 40 (high temp.)	17 - 20 (RT) 8 - 10 (high temp.)

(1) At 2200°C

(2) Sonic Modulus

(3) Properties are for 30 v/o except where noted.

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TABLE A-7.10.2

MECHANICAL PROPERTIES. THE AVERAGE ROOM TEMPERATURE STRENGTH OF 20 AND
30 v/o UC-ZrC COMPOSITE FUEL ELEMENTS

UC-ZrC Content	Loading (mgU/cc)	Density (gm/cc)	Strength	
			Flexural (psi)	Compressive (psi)
20 v/o	151	2.812*	9,155*	16,518 $\pm$ 1,235
20 v/o	102	2.817 $\pm$.033	8,775*	16,820 $\pm$ 1,880
20 v/o	70	2.720 $\pm$.035	7,050*	15,438 $\pm$ 900
30 v/o	151	3.237*	9,390*	21,425 $\pm$ 1,900
30 v/o	102	3.232 $\pm$.046	9,720*	21,227 $\pm$ 1,580
30 v/o	70	3.151 $\pm$.054	8,885*	19,095 $\pm$ 1,755
20 v/o	297	2.828 $\pm$.002	6,950 $\pm$ 210	14,150 $\pm$ 265
30 v/o	297	3.255 $\pm$.008	7,690 $\pm$ 235	17,130 $\pm$ 1,500

* Only two samples; other averages represent three samples.

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TABLE A-7.10.3

ANISOTROPY RATIOS $\frac{\text{TRANSVERSE CTE}}{\text{LONGITUDINAL CTE}}$

PEWEE 2, UNCOATED, 30 v/o UC-ZrC-C

<u>Loading</u>	<u>1000°C</u>	<u>2000°C</u>
102 mgU/cm ³	1.07	1.11
151 mgU/cm ³	1.05	1.08
200 mgU/cm ³	1.06	1.09
267 mgU/cm ³	1.09	1.11

(C-RD)

TABLE A-7.10.4

UC-ZrC-C COMPOSITE THERMAL EXPANSION,
LONGITUDINAL DIRECTION, URANIUM LOADING VARIATION

$$(\alpha) = \text{in./in.}^{\circ}\text{C} \times 10^{-6}$$

	<u>20-1000°C</u>	<u>20-2000°C</u>
30 v/o ZrC-Carbon 0 mgU/cm ³	5.93	6.19
30 v/o UC-ZrC-Carbon 70 mgU/cm ³	5.82	6.08
30 v/o UC-ZrC-Carbon 150 mgU/cm ³	5.83	6.28
30 v/o UC-ZrC-Carbon 200 mgU/cm ³	5.93	6.22
30 v/o UC-ZrC-Carbon 300 mgU/cm ³	5.87	6.17
Average	5.876	6.188
Standard dev, σ	.053	.073

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ELASTIC MODULUS, lb/in.² x 10⁶

Figure A-7.10.1. Variation of Temperature Dependence of Elastic Modulus with Content of UC-ZrC in Composite Fuel.

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FIGURE A-7.10.3

THERMAL EXPANSION OF UC-ZrC-CARBON EXTRUSIONS, VOLUME PERCENT
CARBIDE VARIATION, LONGITUDINAL DIRECTION

FIGURE A-7.10.2

THERMAL EXPANSION OF UC-ZrC - CARBON EXTRUSIONS, VOLUME PERCENT CARBIDE
VARIATION, TRANSVERSE DIRECTION

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APPENDIX A-8

MANUFACTURING SUPPORTING DATA
TRADE STUDY FOR NERVA FUEL ELEMENT
DESIGN CONCEPTS
TRADE STUDY NO. 759

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A-8 Manufacturing Supporting Data (U)

(U) A-8.0 Introduction

(U) Each alternate fuel element design concept (reference Section 2.0) is discussed with respect to the potential of current manufacturing technology to produce an element to meet pertinent functional, technical and non-technical requirements (reference Section 1.0).

(U) The pertinent requirements and the manufacturing criteria used in this appendix to establish the manufacturing potentials are as follows:

(U) Nuclear Criticality - Element can be produced to specified fuel loadings and tolerances. Surveillance technology is in evidence to verify loadings and assure uniformity.

(U) Flow and Thermal Characteristics - Element can be produced with coolant channel and hole locations to specified dimensions and tolerances in both uncoated and coated states. Surveillance technology is in existence to verify dimensions and coating thicknesses.

(U) Reliability - Quality control surveillance technology (procedures and equipment) are in existence to qualify uncoated or coated elements to all specifications and to assure production uniformity.

(U) Corrosion - Corrosion test technology (test equipment, test procedures, evaluation equipment and procedures) is in existence to qualify element to specifications.

(U) Safety - Facility, procedures and equipment are adequate to meet pertinent specifications and regulations on processing, handling, storage and shipping of the element.

(U) Interface Requirements - Element can be produced with external dimensions and tolerances to meet all interface specifications.

(U) Environmental Resistance - Facilities, procedures and equipment are in

existence to prevent environmental contamination (principally hydrolysis in this case).

(U) Development and Qualification - The state of manufacturing technology (facilities, equipment, processes and surveillance) is such that the fuel element can be produced to meet all pertinent manufacturing functional and technical requirements without further development efforts and within budgetary limits.

(U) Schedule - The present status of WNCO manufacturing technology is such as to permit the production of the fuel element within the time schedule specified; NSS-PDR (9/30/70) and R-1 reactor test (12/31/72).

(U) For convenience, the manufacturing potentials established for each of the above requirements and for each alternate fuel element design concept has been coded to permit a quick comparative analysis. The coding used is as follows:

(U) A - High potential to meet requirement criteria. Extensive manufacturing experience has been accumulated.

(U) B - Moderate potential to meet requirement criteria. Limited manufacturing experience. Requires some development and/or modification efforts.

(U) C - Low potential to meet requirement criteria. Little or no manufacturing experience. Extensive development and/or modification efforts needed.

(U) Principal development or modification efforts which are believed to be necessary are also given where applicable.

- (U) A-8.1 Standard Geometry - Intermediate Expansion Graphite (947C812)
- (U) A-8.1.1 Nuclear Criticality (Para. No. 1.3.1)
Potential is A.
- (U) A-8.1.2 Flow and Thermal Characteristics (Para. No. 1.3.2)
Potential is A.
- (U) A-8.1.3 Reliability (Para. No. 1.3.3)
Potential is A.
- (U) A-8.1.4 Corrosion (Para. No. 1.4.6)
Potential is A.
- (U) A-8.1.5 Safety (Para. No. 1.4.7)
Potential is A.
- (U) A-8.1.6 Interface Requirements (Para. No. 1.4.8)
Potential is A.
- (U) A-8.1.7 Environmental Resistance (Para. No. 1.4.12)
Potential is B. Protective procedures have been instituted to minimize hydrolysis during manufacture, storage and shipping phases.
- (U) A-8.1.8 Development and Qualification (Para. No. 1.5.1)
Potential is A. A complete manufacturing technology exists for beaded type graphite elements of standard geometry and extensive experience has been gained in producing this type of element.

- (U) A-8.1.9 Schedule (Para. No. 1.5.2)
- (U) Potential is A. The existing WNCO manufacturing technology will permit the production of this type of element well within the time schedule limits specified.
- (U) A-8.2 Standard Geometry - High Expansion Graphite (947C813)
- (U) A-8.2.1 Nuclear Criticality (Para. No. 1.3.1)
- Potential is A.
- (U) A-8.2.2 Flow and Thermal Characteristics (Para. No. 1.3.2)
- Potential is A.
- (U) A-8.2.3 Reliability (Para. No. 1.3.3)
- Potential is A.
- (U) A-8.2.4 Corrosion (Para. No. 1.4.6)
- Potential is A.
- (U) A-8.2.5 Safety (Para. No. 1.4.7)
- Potential is A.
- (U) A-8.2.6 Interface Requirements (Para. No. 1.4.8)
- Potential is A.
- (U) A-8.2.7 Environmental Resistance (Para. No. 1.4.12)
- Potential is B. See comments A-8.1.7.
- (U) A-8.2.8 Development and Qualification (Para. No. 1.5.1)
- Potential is A. See comments A-8.1.8.

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(U) A-8.2.9 Schedule (Para. No. 1.5.2)

Potential is A. See comments A-8.1.9.

(U) A-8.3 Standard Geometry - Hybrid (947C810)

(U) A-8.3.1 Nuclear Criticality (Para. No. 1.3.1)

(U) Potential is A/B. Adaptation and development of LASL procedures are needed at WNCO to produce composite sections of specified loadings (up to and including 700 mgms/cc) and tolerances.

(U) A-8.3.2 Flow and Thermal Characteristics (Para. No. 1.3.2)

(U) Potential is A/B. Adaptation and development of LASL extrusion, heat treatment and coating procedures for composite fuel are needed at WNCO. Development effort needed on brazing and alignment of joint.

(U) A-8.3.3 Reliability (Para. No. 1.3.3)

(U) Potential is A/B. LASL surveillance technology on composite fuel must be adapted to WNCO equipment and procedures. Standards needed for composite fuel in all phases. Procedures and equipment to verify joint integrity are needed.

(U) A-8.3.4 Corrosion (Para. No. 1.4.6)

(U) Potential is B. Effects of joint and hybrid on test procedures and interpretation of test data needs study. MULE standards needed for interpretation of weight losses in composite fuel section.

(U) A-8.3.5 Safety (Para. No. 1.4.7)

Potential is A.

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- (U) A-8.3.6 Interface Requirements (Para. No. 1.4.8)
- (U) Potential is A/B. Some development effort needed in machining of composite fuel (probably in both tooling and procedures) to assure meeting specified external dimensions and tolerances.
- (U) A-8.3.7 Environmental Resistance (Para. No. 1.4.12)
- (U) Potential is B. See comments A-8.1.7 (Note: With respect to hydrolysis, the graphite section will be the most susceptible. LASL data indicates that the resistance of the composite fuel to hydrolysis is high).
- (U) A-8.3.8 Development and Qualification (Para. No. 1.5.1)
- (U) Potential is B/C. Current plans call for WNCO manufacturing technology to be developed on composite fuel to meet specifications in CY 70-72. The hybrid, however, may require two (2) parallel production facilities unless manufacturing technology is completely interchangeable.
- (U) A-8.3.9 Schedule (Para. No. 1.5.2)
- (U) Potential is B/C. It is believed that the time required to develop a joint braze technique will be a limiting factor considering NSS PDR (9/30/70). The time to establish two (2) parallel production facilities (if necessary) also is a limiting factor.
- (C-RD) A-8.4 37 Hole - Hybrid (947C816)
- (U) A-8.4.1 Nuclear Criticality (Para. No. 1.3.1)
- (U) Potential is A/B. See comments A-8.3.1 concerning composite fuel loadings. Standards needed for gamma scan surveillance of larger cross section and increased number of channels.

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- (U) A-8.4.2 Flow and Thermal Characteristics (Para. No. 1.3.2)
- (U) Potential is C. Considering external size and internal geometry development of extrusion, heat treatment, handling fixtures and coating procedures and equipment are required for both graphite and composite sections. See also comments on standard geometry hybrid A-8.3.2.
- (U) A-8.4.3 Reliability (Para. No. 1.3.3)
- (U) Potential is C. Extensive development of equipment and procedures (plus standards) for all phases of surveillance is required considering geometry. See also comments on standard geometry hybrid A-8.3.3.
- (U) A-8.4.4 Corrosion (Para. No. 1.4.6)
- (U) Potential is C. See comments A-8.3.4. Study of effect of 37-hole cross-section on test procedures needed. Larger cross-section requires chuck development. MULE standards needed for weight loss interpretations.
- (U) A-8.4.5 Safety (Para. No. 1.4.7)
- (U) Potential is B. Existing storage, handling and transport facilities will require modification to accommodate larger cross-section element.
- (U) A-8.4.6 Interface Requirements (Para. No. 1.4.8)
- (U) Potential is B. See comments A-8.3.6 concerning composite element machining development. Larger cross-section will require some modification of machining fixtures and procedures.
- (U) A-8.4.7 Environmental Resistance (Para. No. 1.4.12)
- (U) Potential is B. See comments A-8.1.7. Increased number of coolant channels will probably increase susceptibility of the graphite section to hydrolysis.

- (U) A-8.4.8 Development and Qualification (Para. No. 1.5.1)
- (U) Potential is C. See comments A-8.3.8. Attaining the desired state of manufacturing technology will be more difficult than for the standard geometry hybrid due to added development efforts needed to produce the complex internal geometry and assure its uniformity.
- (U) A-8.4.9 Schedule (Para. No. 1.5.2)
- (U) Potential is C. See comments A-8.3.9 and A-8.4.8. Manufacturing technology will not be available to produce this element on the time schedule specified under existing budgetary restrictions.
- (U) A-8.5 Standard Geometry - Composite (947C809)
- (U) A-8.5.1 Nuclear Criticality (Para. No. 1.3.1)
- (U) Potential is B. Development efforts and modifications consist of adapting LASL manufacturing technology to all phases of WNCO manufacturing including blending, extrusion, heat treatment, machining, coating and quality surveillance.
- (U) A-8.5.2 Flow and Thermal Characteristics (Para. No. 1.3.2)
- Potential is B. See comments A-8.5.1.
- (U) A-8.5.3 Reliability (Para. No. 1.3.3)
- Potential is B. See comments A-8.5.1. Standards for all phases needed.
- (U) A-8.5.4 Corrosion (Para. No. 1.4.6)
- (U) Potential is A. LASL experience in corrosion testing of composite fuel of standard geometry indicates no significant problems involved in testing in existing equipment and using existing test procedures. MULE standards needed.

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- (U) A-8.5.5 Safety (Para. No. 1.4.7)
Potential is A.
- (U) A-8.5.6 Interface Requirements (Para. No. 1.4.8)
Potential is B. See comments A-8.5.1.
- (U) A-8.5.7 Environmental Resistance (Para. No. 1.4.12)
(U) Potential is B. See comments A-8.1.7 and A-8.3.7 with respect to the increased resistance of the composite fuel to hydrolysis.
- (U) A-8.5.8 Development and Qualification (Para. No. 1.5.1)
(U) Potential is B. See comments A-8.3.8. This requirement has not been given a potential because the change-over to composite fuel production at WNCO is still some 6-9 months away from this writing. Budget and schedules geared to this element type for CY 70-72.
- (U) A-8.5.9 Schedule (Para. No. 1.5.2)
Potential is B. See comments A-8.5.8.
- (C-RD) A-8.6 Single Channel - CARBIDE * (947C817)
- (U) A-8.6.1 Nuclear Criticality (Para. No. 1.3.1)
(U) Potential is C. All phases of manufacturing technology (facilities, equipment, processes and surveillance) need extensive development effort. LASL manufacturing experience is limited especially considering the desired loading (2000 mgms/cc).
- (U) A-8.6.2 Flow and Thermal Characteristics (Para. No. 1.3.2)
(U) Potential is C. See comments A-8.6.1. While restrictions

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on maintaining coating tolerances are removed, the lack of manufacturing experience will still require extensive development efforts in extrusion, heat treatment or machining to produce channel diameters and web thicknesses to meet specifications and assure uniformity.

(U) A-8.6.3 Reliability (Para. No. 1.3.3)

(U) Potential is C. See comments A-8.6.1. All phases of surveillance need development in equipment and procedures, especially in areas of development of adequate standards.

(U) A-8.6.4 Corrosion (Para. No. 1.4.6)

(U) Potential is C. See comments A-8.6.1. The corrosion testing of this fuel element will require development of new test equipment and procedures. Evaluation techniques will require development and modification and MULE standards need to be developed.

(U) A-8.6.5 Safety (Para. No. 1.4.7)

(U) Potential is C. See comments A-8.6.1. Numbers and sizes of elements will mean major revision of storage, handling and transport procedures and equipment.

(U) A-8.6.6 Interface Requirements (Para. No. 1.4.8)

(U) Potential is C. Support structure requirements need to be established in order to specify the external dimensional tolerances which will be required. Extensive development of machining procedures is foreseen especially if the tolerances are small or "extrude-to-size" concepts are not realistic.

(U) A-8.6.7 Environmental Resistance (Para. No. 1.4.12)

(U) Potential is B. There is a lack of data on hydrolysis tendencies, but this element may be more resistant

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to this effect than composite.

(U) A-8.6.8 Development and Qualification (Para. No. 1.5.1)

(U) Potential is C. The change in geometry and attendant drastic changes in overall manufacturing technology require development efforts which definitely fall outside of current budget limits.

(U) A-8.6.9 Schedule (Para. No. 1.5.2)

Potential is C. See comments A-8.6.8.

(C-RD) *

Recent design considerations indicate a thermal stress problem for the carbide reference fuel. To alleviate, smaller elements with web thicknesses of 15 mils or less are proposed. The considered opinion from manufacturing standpoint is that element of 15 mil web thickness can be fabricated. However, it is believed that elements of this type with web thicknesses of 10 mils or less cannot be fabricated as reactor grade with any degree of confidence. This opinion is based on consideration of 1) problems in fabricating and maintaining element tolerances 2) production losses which will be attendant to all phases of processing and 3) the large number of fuel element components needed (of the order of 1-3 million). Barring a breakthrough in carbide fabrication technology, the attainment of manufacturing capability for producing the 15 mil web element will require drastic increases in facilities and manpower.

(U) A-8.7 Zig-Zag Composite (947C807)

(U) A-8.7.1 Nuclear Criticality (Para. No. 1.3.1)

(U) Potential is B. Notwithstanding the complex geometry, there appears to be no large problems associated with obtaining desired loadings (see comments A-8.5.1).

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Some development effort needed for gamma scan surveillance plus standards.

- (U) A-8.7.2 Flow and Thermal Characteristics (Para. No. 1.3.2)
- (U) Potential is C. Complex external geometry needs development of equipment, procedures and standards for extrusion, coating processes and Beta-BS to assure internal dimensional tolerances.
- (U) A-8.7.3 Reliability (Para. No. 1.3.3)
- (U) Potential is C. The complex geometry will require extensive development and/or modification of all surveillance equipment and procedures. Standards must be developed for all phases.
- (U) A-8.7.4 Corrosion (Para. No. 1.4.6)
- (U) Potential is C. Major modification of test equipment and procedures to adapt to complex external geometry. MULE standards needed.
- (U) A-8.7.5 Safety (Para. No. 1.4.7)
- (U) Potential is B. Existing storage, handling and transport processes and equipment need some modification.
- (U) A-8.7.6 Interface Requirements (Para. No. 1.4.8)
- (U) Potential is C. Tolerances will probably tighten due to complex fit to interfaces; resulting in some extensive development effort to obtain more precise machining methods and/or accurate measuring methods of external dimensions.
- (U) A-8.7.7 Environmental Resistance (Para. No. 1.4.12)
- (U) Potential is B. Increased surface area will probably not

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seriously adversely effect the resistance of this element type to hydrolysis.

(U) A-8.7.8 Development and Qualification (Para. No. 1.5.1)

(U) Potential is C. The external geometry and attendant changes in overall manufacturing technology will require development efforts which will probably fall outside of present budget limits.

(U) A-8.7.9 Schedule (Para. No. 1.5.2)

Potential is C. See comments A-8.7.8.

(U) A-8.8 High Void - Composite - Long Core (947C814)

(U) A-8.8.1 Nuclear Criticality (Para. No. 1.3.1)

(U) Potential is B. See comments A-8.5.1 concerning composite fuel manufacturing technology development. The additional length should offer no serious problems in obtaining the desired loadings other than some modification of gamma scan surveillance equipment and procedures.

(U) A-8.8.2 Flow and Thermal Characteristics (Para. No. 1.3.2)

(U) Potential is C. Modification of handling, graphitizing, coating and Beta-BS equipment to accommodate longer length elements. Previous comments on composite fuel technology still apply (see comments A-8.5.1).

(U) A-8.8.3 Reliability (Para. No. 1.3.3)

(U) Potential is C. Modification required of all surveillance equipment and procedures to accommodate longer length element. Previous comments on composite fuel technology still apply.

- (U) A-8.8.4 Corrosion (Para. No. 1.4.6)
- (U) Potential is C. The added length will require considerable modification of existing corrosion test equipment.
- (U) A-8.8.5 Safety (Para. No. 1.4.7)
- (U) Potential is C. The added length will require considerable modification of handling, storage and shipping equipment and procedures.
- (U) A-8.8.6 Interface Requirements (Para. No. 1.4.8)
- (U) Potential is B. Added length should not seriously effect the maintaining of external dimensions and tolerances. Machining development contingent with composite fuel still is necessary, however.
- (U) A-8.8.7 Environmental Resistance (Para. No. 1.4.12)
- Potential is B.
- (U) A-8.8.8 Development and Qualification (Para. No. 1.5.1)
- (U) Potential is C. The development of composite fuel technology plus the modification problems associated with the added length mean overall development efforts which appear to fall outside budget limits.
- (U) A-8.8.9 Schedule (Para. No. 1.5.2)
- (U) Potential is C. The time required for development efforts appear to be in excess of NSS PDR (9/30/70).
- (U) A-8.9 Standard Geometry - Graphite - Large Core (947C815)
- (U) A-8.9.1 Nuclear Criticality (Para. No. 1.3.1)
- (U) Potential is A. Manufacturing experience for graphite

exists. Added length should not offer problems in maintaining loadings and tolerances.

(U) A-8.9.2 Flow and Thermal Characteristics (Para. No. 1.3.2)

(U) Potential is C. While graphite technology exists, the added length will require modifications in handling, graphitizing, coating and Beta-BS equipment and procedures.

(U) A-8.9.3 Reliability (Para. No. 1.3.3)

(U) Potential is C. Modification required on all surveillance equipment and procedures due to added length.

(U) A-8.9.4 Corrosion (Para. No. 1.4.6)

(U) Potential is C. See comments A-8.8.4 per modifications due to added length.

(U) A-8.9.5 Safety (Para. No. 1.4.7)

Potential is C. See comments A-8.8.5.

(U) A-8.9.6 Interface Requirements (Para. No. 1.4.8)

(U) Potential is A. Added length should not affect the maintaining of internal dimensions and tolerances on graphite fuel.

(U) A-8.9.7 Environmental Resistance (Para. No. 1.4.12)

Potential is B.

(U) A-8.9.8 Development and Qualification (Para. No. 1.5.1)

(U) Potential is B. Development efforts to accommodate to longer length could fall within budget limits since graphite manufacturing technology exists.

(U)

A-8.9.9 Schedule (Para. No. 1.5.2)

(U)

Potential is C. The time required in development efforts to verify longer length technology probably will fall outside of NSS PDR (9/30/70).

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APPENDIX A-9

SAFETY SUPPORTING DATA
TRADE STUDY FOR NERVA FUEL ELEMENT
DESIGN CONCEPTS
TRADE STUDY NO. 759

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A-9 SAFETY SUPPORTING DATA (U)

A-9.0 INTRODUCTION (U)

(U) The fuel element operational safety requirements are:

- 1) The ability to complete the emergency mission (reduced rating capability producing 10^8 lb-sec impulse) in the event of a malfunction which leads to mission abort.
- 2) Reduction to a minimum of the possibility of fission product contamination of the earth from space.
- 3) Reduction to a minimum of the possibility of astronaut exposure to fission products released from the reactor during either normal or accident conditions.
- 4) Reduction of the radiological problems associated with engine operation in the vicinity of space stations.
- 5) Reduction of the fission product source term associated with ground development and qualification testing.

(U) In addition there is one safety requirement for pre-operational conditions:

- 1) Reduction to a minimum of the nuclear criticality hazard during manufacturing, storage, handling, and transportation.

(U) The operational safety requirements are defined by four basic criteria. These are 1) maintaining the fuel element integrity at higher than normal temperatures, 2) the temperature differential between normal operating conditions and the fuel element degradation threshold, 3) sensitivity to thermal stress cracking, and 4) minimization of fission product release.

(U) The first criterion is based upon the elements' capability to survive a severe over-temperature or a loss-of-coolant without loss of structural integrity through sublimation of graphite or melting of the matrix. This is of importance in reducing the potential amount

of debris released to space. It is desirable even though an objective of the engine design is to reduce the probability of a loss-of-coolant to a negligible value and not require specific design features to accommodate the potential accident. Safety requirements 1, 2, and 3 are partially dependent upon this criterion.

(U) The second criterion is related to the maximum allowable reaction time between the occurrence of a malfunction and appropriate remedial action if operation in the emergency mode is to be successful. The 3 σ normal operating temperatures are from Section A-4, Thermal and Corrosion Evaluation, Table A-4.0-2. Safety requirements 1, 2, and 3 partially depend on this criterion.

(U) Sensitivity to thermal stress cracking comprises the third criterion. This governs the probability of losing portions of elements from the engine under both normal and abnormal temperature conditions. The ranking of the alternate fuel element designs for this criterion are based on Section A-5, Structural Design, Table A-5.1. Safety requirements 1, 2, and 3 are partially dependent upon this criterion.

(U) The definition of the emergency mode operation (safety requirement 1) is to provide 10^8 lb-sec impulse in a reduced thrust operating mode in the event of a malfunction which necessitates mission abort and if possible in a manner which will maintain the ability for full thrust operation after maintenance. The major sensitivity of fuel to malfunctions, that has been identified, is the effect of control drum failures which will either cause power peaking in the periphery if a drum fails to the full out position or fuel element cracking if a control drum fails to the full in position. In either case, it is expected that the emergency mode operation can be completed with a low chamber temperature (1850°R is the nominal chamber temperature selected for the emergency mode) but subsequent operation at rated conditions in the event of fuel element cracking is questionable.

(U) The fourth item, fission product release, has two effects on operation. High release reduces the coolant required for decay heat removal. This is not significant for the release fractions estimated for these fuel designs. Low release is advantageous from the

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standpoint of the "source term" and potential contamination of space equipment. For ground testing, a scrubber to remove the fission products from the effluent stream has been proposed and is under active investigation. It is not expected that increased fission product releases will significantly alter the design or operation of such a system. No significant difference in fission product release is expected between the proposed fuel designs in the event of a loss-of-coolant. The pyrocarbon coating on the UC_2 beads would be destroyed by the high temperatures and so the composition of the graphite matrix, beaded fuel would be similar to that of the composite and carbide fuels. Safety requirements 3, 4, and 5 partially depend on this criterion. Requirement 2 does not, since earth contamination primarily results from fuel element debris falling from orbit. Gaseous fission products have essentially infinite orbit lifetimes.

(U) The pre-operational safety requirement is a fifth criterion. The basic concern is personnel safety, which is directly related to nuclear criticality control. Maintaining individual fuel elements, clusters of elements, and the NSS subcritical involves controls over the number of elements allowed in a given configuration and nuclear poisoning of the assembled reactor. Although all the elements can be made safe, criticality control is simpler for some designs than for others. This is dependent upon the absolute reactivity of the elements* (ρ /element) and the ease of poisoning individual elements (number of poison wires required per element and the difficulty of inserting the wires).

(U) The alternate fuel element designs are compared against the above five criteria. Section A-9.10 contains a summary of the relative safety ranking of the alternate designs. The ratings of the Section 3.0, Safety Comparison Matrix were derived from the material in this Appendix.

A-9.1 STANDARD GEOMETRY-INTERMEDIATE EXPANSION GRAPHITE (U) (WANL Drawing Number 947C812)

* In this section reactivity is defined to be proportional to the potential hydrogen reactivity insertion (ρ /kgmH x void volume).

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A-9.1.1 Fuel Characteristics at High Temperatures (U)

(C-RD) This fuel material, UC_2 pyrocoated beads in a graphite matrix, is very refractory. Even though the UC_2 of the bead may form a solid solution with the pyrocoating at a temperature below sublimation temperature, the element is not expected to lose its structural integrity until after extensive sublimation of the matrix occurs. Sublimation, which occurs at about $5800^{\circ}R$, is very effective in removing decay heat from the core, and essentially maintains the fuel at that temperature. If bundling through the lateral support system is effective, it is expected that the un-sublimed fuel will remain within the core. On failure of the lateral support, it can be expected that the fuel elements or remaining fragments would disperse in relatively large pieces.

A-9.1.2 Temperature Margin (U)

(C-RD) The temperature margin is $989^{\circ}R$ and is the difference between the graphite sublimation temperature of $5800^{\circ}R$ (cf. A-9.1.1) and the 3σ fuel temperature of $4811^{\circ}R$.

A-9.1.3 Thermal Stress Cracking (U)

(U) This design does not meet the stress criteria in that the computed stresses are higher than the allowable. This design is more susceptible to cracking than is the Standard Geometry - High Expansion Graphite.

A-9.1.4 Fission Product Release (U)

(C-RD) The release of fission products from this design during normal operations is well defined. A large amount of information has been developed on fission product diffusion from laboratory experiments and reactor tests. For a typical 50 minute full power run releases of 0.3% of the gross gamma and 1% of the iodine 131 activities are predicted. Corrosion of this fuel by the hydrogen coolant results in the release of particulate material containing fission products. This is attributed to the ability of the pyrocoated fuel beads to survive the exhaust plume. The size distribution ranges from particles of several hundred microns diameter

to those of sub-micron size. The amount of particulate material depends upon the severity of the corrosion, but potentially is of the order of 1000 grams.

A-9.1.5 Criticality Control (U)

(U) This reference design is considered the best of the alternates with respect to criticality control. It is the least reactive, allowing the most elements to be placed in any one array or requiring the least amount of poisoning, and the coolant channels are large and free of obstructions except for the brazed joint at the composite tip on the hot end.

A-9.2 STANDARD GEOMETRY - HIGH EXPANSION GRAPHITE (U)
(WANL Drawing Number 947C813)

A-9.2.1 Fuel Characteristics at High Temperatures (U)

(U) This design is considered equivalent to the Standard Geometry-Intermediate Expansion Graphite in this property.

A-9.2.2 Temperature Margin (U)

(U) This design is considered equivalent to the Standard Geometry-Intermediate Expansion Graphite in this property.

A-9.2.3 Thermal Stress Cracking (U)

(U) This design does not meet the stress criteria in that the computed stresses are higher than the allowable. It is less susceptible to cracking than is the Standard Geometry-Intermediate Expansion Graphite.

A-9.2.4 Fission Product Release (U)

(U) This design is considered equivalent to the Standard Geometry-Intermediate Expansion Graphite in this property.

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A-9.2.5 Criticality Control (U)

(U) This design is considered equivalent to the Standard Geometry-Intermediate Expansion Graphite in this property.

A-9.3 STANDARD GEOMETRY - HYBRID (U)
(WANL Drawing Number 947C810)

A-9.3.1 Fuel Characteristics at High Temperatures (U)

(C-RD) The hybrid element is composed of two partial elements brazed together. The forward (cooler) section is Standard Geometry-High Expansion Graphite and the aft (hotter) section is Standard Geometry-Composite. The two sections are brazed at 2700-2800°R (4860-5040°R) to form one element. The comments on its two components are applicable to this element.

A-9.3.2 Temperature Margin (U)

(C-RD) 5400°R (cf. A-9.5.1) - 4822°R = 578°R. The braze joint between the section will probably melt before the fuel maximum temperature increases 578°R.

A-9.3.3 Thermal Stress Cracking (U)

(U) This design does not meet the stress criteria in that the computed stresses are higher than the allowable. It is better than either of the two all graphite matrix designs, however.

A-9.3.4 Fission Product Release (U)

(U) The majority of the fission products which diffuse from the fuel elements come from the hottest 1/3 of the element. This design is thus similar to the Standard Geometry-Composite in its diffusion characteristics. Its corrosion behaviour is intermediate to that of the High Expansion Graphite and Composite designs.

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A-9.3.5 Criticality Control (U)

(C-RD) This design is more difficult to maintain in a non-critical condition than is the reference design, the Standard Geometry-Intermediate Expansion Graphite. It requires somewhat more poisoning and has a potential obstruction in a misalignment of the braze joint between the sections at station 35.

A-9.4 THIRTY-SEVEN HOLE - HYBRID (C-RD)
(WANL Drawing Number 947C816)

A-9.4.1 Fuel Characteristics at High Temperatures (U)

(U) This design is considered equivalent to the Standard Geometry-Hybrid in this property.

A-9.4.2 Temperature Margin (U)

(C-RD) 5400°R (cf. A-9.5.1) - 4726°R = 674°R . The braze joint between the sections will probably melt before the fuel maximum temperature increases 674°R .

A-9.4.3 Thermal Stress Cracking (U)

(U) This design meets the stress criteria in that the allowable stresses are greater than the computed ones. It is superior to all the standard geometry, regular core designs.

A-9.4.4 Fission Product Release (U)

(U) This design is considered equivalent to the Standard Geometry-Hybrid in this property.

A-9.4.5 Criticality Control (U)

(U) The remarks of Section A-9.3.5 are applicable to this design. In addition the coolant channels are ~20% smaller in diameter than those in the standard design, increasing the number of wires for the same amount of poison insertion and increasing the difficulty of inserting each individual poison wire.

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A-9.5 STANDARD GEOMETRY - COMPOSITE (U)
(WANL Drawing Number 947C809)

A-9.5.1 Fuel Characteristics at High Temperatures (U)

(C-RD) The composite fuel presently being considered consists of 30 v/o Zr and U carbides in a graphite matrix. The ZrC and UC form a eutectic at about 3000°K (5400°R) which is a slightly higher temperature than required for the UC-C eutectic of the beaded fuel. A considerably larger portion of the matrix is involved in the ZrC-UC eutectic formation however, and this is expected to weaken the fuel. As with the beaded fuels, sublimation of the matrix is effective for removal of the fission product decay power.

A-9.5.2 Temperature Margin (U)

(C-RD) 5400°R (cf. A-9.5.1) - 4822°R = 578°R

A-9.5.3 Thermal Stress Cracking (U)

(U) This design meets the stress requirement in that the computed stresses are less than the allowable. The allowable exceeds the computed by the smallest margin of any of the acceptable designs.

A-9.5.4 Fission Product Release (U)

(U) Relatively little is known about the fission product release from the composite fuel. One set of data is available for composite fuel without coating on the coolant channels. This data, and extrapolation of the reduction in diffusion due to coolant channel coating on beaded fuel, indicates that the fission product loss for a 50 minute full power run would be 9% for gross gamma activity and 9% for I-131. At temperatures near the fuel exit temperature, 15% of the activity will be released. The data for uncoated fuel indicates 50% at 2300°K (4150°R). These release fractions are roughly an order of magnitude greater than those for the graphite matrix, beaded fuel.

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(U) The amount of fission products released by corrosion is less for this material than for the beaded graphite material for the same reactivity loss. The nature of the material is such that uranium is retained in the fuel element even if extensive loss of the graphite matrix occurs. In addition what particulate material is released is expected to burn more readily in the exhaust plume than for the beaded fuel and be released either in gaseous form or as smaller particulates.

A-9.5.5 Criticality Control (U)

(C-RD) This design requires relatively little additional poison over that required for the reference design. The coolant channels are large and free of obstructions except for the brazed joint at the composite tip on the hot end.

A-9.6 SINGLE CHANNEL - CARBIDE (C-RD)
(WANL Drawing Number 947C817)

A-9.6.1 Fuel Characteristics at High Temperatures (U)

(C-RD) This design is made of a solid solution of uranium and zirconium carbides. This solid solution melts at about 3000°K (5400°R) and the heat of fusion is approximately 1% of the heat of sublimation of graphite. It can be expected that a large fraction of a core of this material would melt in the event of loss-of-coolant. This material is also subject to a slow depletion of carbon by interaction with the hydrogen coolant and the resultant metal rich phase would melt in the range of 2500°K (4500°R) (See Section A-4.6.1.2).

A-9.6.2 Temperature Margin (U)

(C-RD) 5400°R (cf. A-9.6.1) - $4817^{\circ}\text{R} = 583^{\circ}\text{R}$. This temperature margin is somewhat misleading since the carbide matrix material melts at 5400°R causing rapid damage rather than the relatively slow sublimation of the graphite matrix and composite fuels.

A-9.6.3 Thermal Stress Cracking (U)

(U) The computed stresses exceed the allowable by a larger margin (3.03) than for any of the alternates considered.

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A-9.6.4 Fission Product Release (U)

(U) No experimental information is available on potential fission product release from this material. Based on the physical nature of fission product diffusion it is felt that release values would be intermediate to the beaded graphite and composite materials.

A-9.6.5 Criticality Control (U)

(U) This design is the most difficult to place in a nuclear "safe" state. It is the most reactive and the many joints (the element is composed of 10 sections) present a large potential number of obstructions to the insertion of poison wires.

A-9.7 ZIG ZAG - COMPOSITE (U)
(WANL Drawing Number 947C807)

A-9.7.1 Fuel Characteristics at High Temperatures (U)

(U) This design is considered equivalent to the Standard Geometry-Composite in this property.

A-9.7.2 Temperature Margin (U)

(U) This design is considered equivalent to the Standard Geometry-Composite in this property.

A-9.7.3 Thermal Stress Cracking (U)

(U) This design is slightly less susceptible to cracking than is the Standard Geometry-Composite.

A-9.7.4 Fission Product Release (U)

(U) This design is considered equivalent to the Standard Geometry-Composite in this property.

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A-9.7.5 Criticality Control (U)

(U) This design is considered equivalent to the Standard Geometry-Composite in this property.

A-9.8 HIGH VOID - COMPOSITE - LONG CORE (U)
(WANL Drawing Number 947C814)

A-9.8.1 Fuel Characteristics at High Temperatures (U)

(U) This design is considered equivalent to the Standard Geometry-Composite in this property.

A-9.8.2 Temperature Margin (U)

(C-RD) 5400°R (cf. A-9.5.1) - $4782^{\circ}\text{R} = 618^{\circ}\text{R}$.

A-9.8.3 Thermal Stress Cracking (U)

(U) This design is the least susceptible to cracking of any of the alternates considered.

A-9.8.4 Fission Product Release (U)

(U) This design is considered equivalent to the Standard Geometry-Composite in this property.

A-9.8.5 Criticality Control (U)

(C-RD) This design requires more poison than does the Standard Geometry-Composite in that it has a larger void volume which can be filled with a moderating material. The coolant channels are slightly larger ($\sim 5\%$) than the reference design and are free of obstruction except for the brazed joint at the composite tip on the hot end.

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A-9.9 STANDARD GEOMETRY - GRAPHITE - LARGE CORE (U)

(WANL Drawing Number 947C815)

A-9.9.1 Fuel Characteristics at High Temperatures (U)

(U) This design is considered equivalent to the Standard Geometry-Intermediate Expansion Graphite in this property.

A-9.9.2 Temperature Margin (U)

(C-RD) 5800°R (cf. A-9.1.1) - 4721°R = 1079°R .

A-9.9.3 Thermal Stress Cracking (U)

(U) This design is less susceptible to cracking than is the Standard Geometry-Intermediate Expansion Graphite.

A-9.9.4 Fission Product Release (U)

(U) This design is considered equivalent to the Standard Geometry-Intermediate Expansion Graphite in this property.

A-9.9.5 Criticality Control (U)

(U) This design requires more poison than does the Standard Geometry-Intermediate Expansion Graphite in that it has a larger void volume which can be filled with a moderating material. The coolant channels are the same as for the Standard Geometry-Intermediate Expansion Graphite.

A-9.10 RELATIVE RATINGS OF THE ALTERNATE DESIGNS (U)

(U) A summary of the above data appears in Table A-9.1.

(U) A qualitative relative comparison of the alternate fuel elements with respect to the operational and pre-operational safety requirements is contained in Table A-9.2.

TABLE A-9.1

RELATIVE CRITERIA RANKING

(C-RD) Safety Requirements	Standard Geometry Intermediate Expansion Graphite	Standard Geometry High Expansion Graphite	Standard Geometry Hybrid	37 Hole Hybrid	Standard Geometry Composite	Single Channel Carbide	ZigZag Composite	High Void Composite Long Core	Standard Geometry Graphite Large Core
1) Ability to Complete Emergency Mode Operation Following Malfunction	High	High	Moderate	Moderate	Moderate	Low	Moderate	Moderate	Very High
2) Retention of Fission Products in Long- Lived Orbit in Space	Very High	Very High	High	High	High	Low	High	High	Very High
3) Retention of Fission Products within the Pressure Vessel/ Under Accident Conditions	High	High	Moderate	Moderate	Moderate	Low	Moderate	Moderate	High
4) Fission Product Contamination of Space Equipment for Normal Opera- tion	Low	Low	High	High	High	Moderate	High	High	Low
5) Fission Product Source Term for Ground Testing - Relative Release	1	1	10	10	10	5	10	10	1
6) Preoperational Safety (criticality control)	Very High	Very High	High	Moderate	High	Low	High	High	High
Overall Acceptability	High	High	Moderate	Moderate	Moderate	Low	Moderate	Moderate	High

TABLE A-9.2

RELATIVE SAFETY RANKING

Criteria (one is best)	Standard Geometry- Intermediate Expansion Graphite	Standard Geometry- High Expansion Graphite	Standard Geometry Hybrid	37 Hole Hybrid	Standard Geometry- Composite	Single Channel Carbide	Zig-Zag Composite	High Void Composite Long Core	Standard Geometry- Graphite Large Core
1. Fuel Characteristics at High Temperature	2	2	5	5	4	6	4	3	1
2. Temperature Margin	2	2	5	3	5	5	5	4	1
3. Thermal Stress Cracking	6	6	6	3	5	9	4	1	2
4. Fission Product Release	1	1	3	3	3	2	3	3	1
5. Criticality Control	1	1	3	5	2	6	2	4	4

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APPENDIX A-10

RADIATION AND SHIELDING SUPPORTING DATA
TRADE STUDY FOR NERVA FUEL ELEMENT
DESIGN CONCEPT
TRADE STUDY NO. 759

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A-10.0 INTRODUCTION

This section provides radiation heating rates and the radiation environment for the various fuel element concepts. This section also demonstrates how changes in the fuel element affect the shield weight and dose rates external to the Nuclear Subsystem.

The radiation heating rate data is provided as input to thermal design analysis in order that the Flow and Thermal Characteristics requirement may be evaluated. The radiation environment data is provided as input to radiation effects analysis in order that the Environmental Resistance requirement may be evaluated. The effect of the different fuel element concepts on the shield weight and dose rates external to the Nuclear Subsystem was determined in order to evaluate the Weight requirement.

In obtaining the radiation heating rates and radiation environment for the various fuel element concepts, it was assumed that the differences in the design did not significantly alter the neutron and photon fluxes in the core. Therefore, the heating rate and radiation environment data is the same for all concepts and will be reported and described for the first design concept only. This data will not be discussed in sections for the remaining design concepts to avoid repetition of the same information.

A-10.1 DESIGN CONCEPT 2.1 - STANDARD GEOMETRY - INTERMEDIATE EXPANSION GRAPHITE (DRAWING 711J596)

A-10.1.1 Flow and Thermal Characteristics Requirement

Figures A-10-1 and A-10-2 provide, respectively, the relative axial and relative radial distribution of radiation heating in the fuel elements. The heating rate at any location within the fuel element is determined in the following manner:

1. Determine the value of the relative radial distribution of heating for the position of interest,
2. Determine the value of the relative axial distribution of heating for the location of interest,
3. Determine the average value of the heating rate from Table A-10-1 for the appropriate region and loading code, and

4. The heating rate for the material at the location of interest is the product of the average heating rate, the value of the relative radial distribution, and the value of the relative axial distribution.

A-10.1.2 Environmental Resistance Requirement

Table A-10-2 provides the radiation environment to evaluate the Structural Reliability requirement. This data is provided at the following locations:

1. reactor centerline at the core inlet,
2. reactor centerline at the core midplane,
3. reactor centerline at the core outlet,
4. outer radius of core at the core inlet,
5. outer radius of core at the core midplane, and
6. outer radius of core at the core outlet.

This table provides the thermal neutron flux ($E < 0.4$ ev), the fast neutron flux ($E > 1.0$ Mev), and the gamma ray dose rate (Rads-C/hr). Neutron fluences and gamma doses are obtained for the longest single burn and the total integrated power by multiplying this data by the factors supplied on Table A-10-1. The factor for the longest single burn is based on the requirement for a one hour run during ground testing which will be longer than any single mission burn during flight. The factor for the total integrated power is based on ten hours of full power operation with minimal power being generated in the startup and shutdown ramps. At the present time, all of the requirements on the reactor operating profile have not been completely defined; therefore, as new information on the operating profile becomes available, the factors to be applied to the fluxes and dose rates to obtain fluences and doses will be modified to reflect the new information.

A-10.1.3 Weight Requirement

Table A-10-3 provides the estimated gamma ray and fast neutron dose rates at 30 feet from the core center on the reactor centerline for the fuels involved in this study. Table A-10-4 provides the estimated BATH shield weights and delta shield weights

which give a constant dose rate at 30 feet from the core center on the reactor centerline.

Design Concept 2.1, represents the reference design employed for shielding considerations. From Table A-10-4 it is noted that this fuel element configuration employs 3850 pounds of Reference Design (939J685E) BATH Shield.

A-10.2 DESIGN CONCEPT 2.2 - STANDARD GEOMETRY - HIGH EXPANSION GRAPHITE (DRAWING 711J596)

Weight Requirement

Table A-10-3 provides the estimated gamma ray and fast neutron dose rates at 30 feet from the core center on the reactor centerline for this fuel element concept. Table A-10-4 provides the estimated BATH shield weights and delta shield weights which give a constant dose rate at 30 feet from the core center on the reactor centerline.

Since the neutron and gamma ray dose rates are not affected in the same manner by core changes, the shield weights are based upon the shield which would achieve both the neutron and gamma dose rate criteria (i. e., the higher shield weight). Table A-10-4 indicates that this fuel element concept increases the BATH shield weight by 47 pounds relative to the shield weight required for the reference design fuel element (Design Concept 2.1, Standard Geometry - Intermediate Expansion Graphite).

A-10.3 DESIGN CONCEPT 2.3 - STANDARD GEOMETRY - HYBRID (DRAWING 711J596)

Weight Requirement

Table A-10-3 provides the estimated gamma ray and fast neutron dose rates at 30 feet from the core center on the reactor centerline for this fuel element concept. Table A-10-4 provides the estimated BATH shield weights and delta shield weights which give a constant dose rate at 30 feet from the core center on the reactor centerline.

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(U) Table A-10-4 indicates that this fuel element concept decreases the BATH shield weight by 212 pounds relative to the shield weight required for the reference design fuel element (Design Concept 2.1, Standard Geometry-Intermediate Expansion Graphite).

(U) However, it should be noted that the weight of the core for this fuel element concept increases by 1800 pounds. Therefore, while the shield weight decreases, the weight of the Nuclear Subsystem increases by approximately 1600 pounds.

A-10.8 DESIGN CONCEPT 2.8 - HIGH VOID-COMPOSITE-LONG CORE (U)

A-10.8.1 Flow and Thermal Characteristics Requirement (U)

(CRD) Figure A-10-1 provides the relative axial distribution of radiation heating for a 52-inch fuel element. By employing this data at the same relative axial location, the distribution given in Figure A-10-1 is applicable to a 62-inch long fuel element.

A-10.8.2 Weight Requirement (U)

(U) Table A-10-3 provides the estimated gamma ray and fast neutron dose rates at 30 feet from the core center on the reactor centerline for this fuel element concept. Table A-10-4 provides the estimated BATH shield weights and delta shield weights which give a constant dose rate at 30 feet from the core center on the reactor centerline.

(U) Table A-10-4 indicates that this fuel element concept decreases the BATH shield weight by 212 pounds relative to the shield weight required for the reference design fuel element (Design Concept 2.1, Standard Geometry-Intermediate Expansion Graphite).

(U) However, it should be noted that the weight of the core for this fuel element concept increases by 2600 pounds. Therefore, while the shield weight decreases, the weight of the Nuclear Subsystem increases by approximately 2400 pounds.

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A-10.9 DESIGN CONCEPT 2.9 - STANDARD GEOMETRY-GRAPHITE-LARGE CORE

A-10.9.1 Flow and Thermal Characteristics Requirement

The heating rate distribution, both relative axial and radial, will be similar to the information provided in Figures A-10-1 and A-10-2 at the same relative position within the core. However, detailed heating rate distributions can not be accurately obtained until the final uranium zone loadings have been defined.

A-10.9.2 Weight Requirement

Table A-10-3 provides the estimated gamma ray and fast neutron dose rates at 30 feet from the core center on the reactor centerline for this fuel element concept. Table A-10-4 provides the estimated BATH shield weights and delta shield weights which give a constant dose rate at 30 feet from the core center on the reactor centerline.

Since the neutron and gamma ray dose rates are not affected in the same manner by core changes, the shield weights are based upon the shield which would achieve both the neutron and gamma dose rate criteria (i. e., the higher shield weight). Table A-10-4 indicates that this fuel element concept decreases the BATH shield weight by 31 pounds relative to the shield weight required for the reference design fuel element (Design Concept 2.1, Standard Geometry-Intermediate Expansion Graphite).

It should be noted that since the shield extends to the outer edge of the reflector, the large diameter core requires a larger diameter shield which increases the

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shield weight. However, a longer core tends to reduce the dose rate on the reactor centerline external to the Nuclear Subsystem by providing more self-shielding of core sources and reducing the volumetric source strength at the inlet end of the core. Therefore, the longer core tends to require less shield weight to maintain the same dose rate external to the Nuclear Subsystem. These two effects tend to cancel one another resulting in a small decrease in shield weight of 31 pounds.

A-10.10 METHOD OF ANALYSIS AND ASSUMPTIONS (U)

(U) This analysis considers only the relative attenuating properties of the different cores analyzed. The neutron and photon fluxes internal to the Nuclear Subsystem were assumed to be the same as the data determined by two-dimensional discrete ordinates transport calculations for the reference fuel element design (Concept 2.1, Standard Geometry-Intermediate Expansion Graphite). Therefore, the radiation environment and heating rates are the same for all concepts.

(U) With the exception of the Hybrid and Zigzag Composite Fuels, the dose rates external to the Nuclear Subsystem were obtained from KAP-V point kernel analysis. This analysis made the following assumptions:

1. Core gamma ray source spectrum and magnitude is constant for all cores.
2. Secondary gamma ray sources generated outside the core are the same for all cases.
3. Support structure forward of the core is the same for all cores.

(CRD) The dose rates presented for both the 19 and 37 coolant channel hybrid fuel were assumed to be the same as the data for the high expansion graphite mixture of carbon and pyrocarbon coated UC₂ particles. This assumption is based on the fact that in the hybrid fuel the high density UC-ZrC fuel is located at the aft end of the reactor where it will not significantly affect the radiation leaving the forward end of the reactor.

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Therefore, the radiation dose rates and the shield weights would be virtually unaffected.

(U) The dose rates presented for the zigzag composite fuel were assumed to be the same as the data for the standard geometry-composite fuel. The change in geometry between the two fuels would not affect the radiation characteristics of the Nuclear Subsystem.

FIGURE A-10-1

(CRD) RELATIVE AXIAL DISTRIBUTION OF HEATING IN THE FUEL ELEMENT,
SEE TABLE A-10-1 FOR AVERAGE HEATING RATES (U)

FIGURE A-10-2

(CRD) RELATIVE RADIAL DISTRIBUTION OF HEATING IN THE FUEL ELEMENT,
SEE TABLE A-10-1 FOR AVERAGE HEATING RATES (U)

TABLE A-10-1

(CRD) AVERAGE HEATING RATES AND FUEL ELEMENT LOADING CODES FOR THE CORE (U)

REGION NUMBER	AVERAGE HEATING RATE (BTU/hr-in ³)	LOADING CODE	DENSITY (lb/in ³)
1. (0.0 to 3.1 in. radius)	1.643×10^5	12	0.08588
2. (3.1 to 5.7 in. radius)	1.637×10^5	13	0.08639
3. (5.7 to 7.1 in. radius)	1.617×10^5	14	0.08700
4. (7.1 to 7.9 in. radius)	1.617×10^5	15	0.08758
5. (7.9 to 8.9 in. radius)	1.622×10^5	16	0.08819
6. (8.9 to 9.6 in. radius)	1.626×10^5	17	0.08884
7. (9.6 to 10.4 in. radius)	1.637×10^5	18	0.08957
8. (10.4 to 11.6 in. radius)	1.625×10^5	19	0.09025
9. (11.6 to 12.8 in. radius)	1.573×10^5	20	0.09098
10. (12.8 to 16.1 in. radius)	1.547×10^5	21	0.09173
11. (16.1 to 16.7 in. radius)	1.390×10^5	17	0.08884
12. (16.7 to 17.2 in. radius)	1.464×10^5	16	0.08819
13. (17.2 to 17.8 in. radius)	1.440×10^5	12	0.08588
14. (17.8 to 18.7 in. radius)	1.396×10^5	11	0.08220

NOTE: Refer to Figures A-10-1 and A-10-2 for relative heating rate distributions

TABLE A-10-2

(CRD) FUEL ELEMENT RADIATION ENVIRONMENT AT OPERATING
CONDITIONS (1575 Mw) (U)

LOCATION	NEUTRON FLUXES ^(1,2) (N/CM ² -SEC)		GAMMA RAY DOSE RATE ^(1,3) (Rads-C/hr)
	THERMAL FLUX (E<0.4 ev)	FAST FLUX (E>1.0 Mev)	
1. Reactor Centerline at the Core Inlet	5.4 (13)	8.2 (14)	7.4 (9)
2. Reactor Centerline at the Core Midplane	4.8 (13)	4.5 (15)	3.9 (10)
3. Reactor Centerline at the Core Outler	7.4 (12)	5.4 (14)	4.6 (9)
4. Outer Radius of Core at the Core Inlet	1.5 (14)	3.2 (14)	3.5 (9)
5. Outer Radius of Core at the Core Midplane	6.4 (14)	2.1 (15)	1.8 (10)
6. Outer Radius of Core at the Core Outler	5.4 (13)	2.1 (14)	2.5 (9)

(1) Numbers in parentheses refer to powers of ten.

(2) The neutron fluences for the longest single burn at 1505 Mw are obtained by multiplying the data by 3.44 (3) while the fluences for the total integrated power are obtained by multiplying by 3.44 (4).

(3) The dose for the longest single burn at 1505 Mw is obtained by multiplying this data by 0.96 while the dose for the total integrate power is obtained by multiplying by 9.6.

TABLE A-10-3

(CRD) ESTIMATED GAMMA RAY AND FAST NEUTRON DOSE RATE AT 30 FEET FROM THE
CORE CENTER ON THE REACTOR CENTERLINE (U)

FUEL ELEMENT CONFIGURATION	ESTIMATED DOSE RATE (REM/HR) ⁽¹⁾	
	GAMMA	FAST NEUTRON
1 Standard Geometry-Intermediate Expansion Graphite	1.1 (5)	5.5 (5)
2 Standard Geometry-High Expansion Graphite	1.1 (5)	5.7 (5)
3 Standard Geometry-Hybrid	1.1 (5)	5.7 (5)
4 37 Hole-Hybrid	1.1 (5)	5.7 (5)
5 Standard Geometry-Composite	7.9 (4)	4.4 (5)
6 Single Channel-Carbide	5.7 (4)	4.3 (5)
7 Zigzag-Composite	7.9 (4)	4.4 (5)
8 High Void-Composite-Long Core	7.7 (4)	4.2 (5)
9 Standard Geometry-Graphite-Large Core	9.4 (4)	3.9 (5)

(1) NUMBERS IN PARENTHESES REFER TO POWERS OF TEN.

TABLE A-10-4

(CRD) ESTIMATED BATH SHIELD WEIGHTS AND DELTA SHIELDING WEIGHTS
WHICH GIVE A CONSTANT DOSE RATE⁽¹⁾ AT 30 FEET FROM THE
CORE CENTER ON THE REACTOR CENTERLINE (U)

FUEL ELEMENT CONFIGURATION	SHIELD WEIGHT (LBS)	DELTA SHIELD WEIGHT (LBS)
1 Standard Geometry-Intermediate Expansion Graphite	3850	---
2 Standard Geometry-High Expansion Graphite	3897	+ 47
3 Standard Geometry-Hybrid	3897	+ 47
4 37-Hole-Hybrid	3897	+ 47
5 Standard Geometry-Composite	3638	-212
6 Single Channel-Carbide	3630	-220
7 Zigzag-Composite	3638	-212
8 High Void-Composite-Long Core	3544	-306
9 Standard Geometry-Graphite-Large Core	3819	- 31

(1) THE CONSTANT DOSE RATE IS THE MEDIUM EXPANSION DESIGN LEVEL FOR EITHER GAMMA RAYS OR NEUTRONS

Fold
or Cut

Fold
or Cut

TITLE Trade Study for NERVA Fuel Element Design Concepts Trade Study No. 759 ✓	
DOCUMENT NO. WANL-TME-2674		DATE February 1970	
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