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**WANL-TME-2758
FEBRUARY 1971**

DATA ITEM S-101

**NUCLEAR ANALYSIS REPORT
NUCLEAR SUBSYSTEM (U)**

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DATA ITEM S-101

NUCLEAR ANALYSIS REPORT NUCLEAR SUBSYSTEM (U)

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1.0 INTRODUCTION

1.1 PURPOSE OF REPORT

The purpose of this report is to document the Nuclear Analysis of the NUCLEAR SUBSYSTEM (NSS) and its Engineering Critical Components (ECC). This report constitutes the primary source of all steady state and transient nuclear data. The detailed Nuclear Analysis consisted of the following:

Support for the derivation and establishment of appropriate performance and design requirements for the NSS and its ECCs,
Establishment and substantiation of the appropriate parameters of the selected design.

The following detailed information is contained in this report:

- 1) Development of the absolute reactivity prediction for the NSS and appropriate reactivity compensation techniques to account for the uncertainty in reactivity.
- 2) Development of a multizoned fuel loading distribution.
- 3) Development of a corresponding power distribution.
- 4) Development of the control drum span.
- 5) The effect of the ECCs upon absolute reactivity and the radial and axial power distributions.
- 6) Development of the cold-to-hot reactivity feedback models and the corresponding cold-to-hot power distribution models.
- 7) The key environmental conditions for all operating and non-operating conditions.
- 8) Development of the uncertainties in the analytical methods and procedure. A description of the PAX/R-1 critical assembly, which provides a basis for most of this uncertainty analysis, is also included in this report. Material reactivity coefficients, drum span, and radial and axial power distribution measurements for the PAX/R-1 assembly are presented with their corresponding predictions.

1.2 CONTENTS OF REPORT

This report presents nuclear results for both composite and graphite fueled cores. Nuclear data are presented for the Nuclear Subsystem (NSS) as a whole (Chapter 2) and for the following Type A Engineering Critical Components (ECC):

- 1) Support Plate and Plena, EC-677561 (Chapter 6),
- 2) Internal Shield, EC-677562 (Chapter 7), and
- 3) Reflector Assembly, EC-677559 (Chapter 8).

Nuclear data for the following Type C ECCs are not presented because these ECCs are involved in Trade Studies whose results are to be reported for NSS PDR:

- 1) Fuel Element, EC-677566 (Chapter 3),
- 2) Cluster Hardware, EC-677558 (Chapter 4),
- 3) Core Periphery, EC-677564 (Chapter 5),
- 4) The Lateral Support portion of the Reflector Assembly, EC-677562 (Chapter 8), and
- 5) NSS Instrumentation, EC-677565 (Chapter 11).

The nuclear data for these ECCs are reported in the appropriate S-54 Data Item as follows:

- 1) Fuel Element--Trade Study No. 772,
- 2) Cluster Hardware--Trade Study No. 774,
- 3) Core Periphery--Trade Study No. 778,
- 4) Lateral Support--Trade Study No. 782, and
- 5) NSS Instrumentation--Trade Study No. 792.

Nuclear data are not presented for the following chapters for the reasons given:

- 1) Control Drum Drive Assembly, EC-677585 (Chapter 9)--Stuck drum analysis will be presented in an update of the Data Item.
- 2) Structural Support Coolant Assembly, EC-677575 (Chapter 10)--Due to the location of the SSCA, nuclear effects are negligible.

In general, this Data Item contains the following information for the NSS and each ECC with both composite and graphite fuel in the core:

- 1) Description of the component,
- 2) Summary of results and conclusions,
- 3) The requirements towards which the information is directed,
- 4) The method of analysis which includes
 - a) a description of the analytical procedures employed,
 - b) a description of the analytical models used,
 - c) a description of the status/maturity of the input data upon which the results are based, and
 - d) a discussion of the validity of the analysis, including comparisons with experimental results and the derivation of the uncertainties in the nuclear analysis.
- 5) The results, analysis, and interpretation of the data which contains the following technical information for both steady state and transient operation of the NSS and its components as appropriate:
 - a) NSS absolute reactivity and corresponding fuel loading distribution,
 - b) radial and axial fission distributions,
 - c) total control drum span,
 - d) effects of components upon NSS reactivity and fission distribution, and
 - e) cold-to-hot reactivity feedback.

2.0 NUCLEAR SUBSYSTEM

2.1 INTRODUCTION

The NSS consists of a hydrogen cooled, graphite moderated, intermediate neutron energy reactor and its associated reactivity control devices consisting of neutron absorbing control drums with their actuation system and the Structural Support Coolant Assembly.

2.1.1 Description of Nuclear Subsystem

The NUCLEAR SUBSYSTEM includes a core, reflector, core support plate and associated piping and an internal shield. The core contains fuel element clusters. Each cluster, except for the irregular clusters in the periphery, consists of 6 fueled elements surrounding a central unfueled element. A counterflow support stem passes through each central element to the support plate. The support stems carry the core pressure drop load to the core support plate and also accommodate a varying hydrogen inventory for reactivity control.

The core support plate transmits the core pressure drop load to the reflector and also incorporates plenums for support stem hydrogen feed, exhaust and bypass.

The reflector surrounds the core and, as its name implies, returns neutrons to the core to sustain criticality. A lateral support system is also housed in the reflector. This supplies radial restraint for the core acting through the core periphery which consists of thermal and flow barriers backed up by filler strips which round the core into a cylindrical shape for interfacing with the lateral support. Eighteen control drums are contained in the reflector. Each control drum has a neutron absorbing plate covering approximately 120 degrees of its external surface. Appropriate synchronous positioning of the control drums by rotation provides the means then to establish reactor criticality.

An internal shield which provides radiation protection for engine components and NSS instrumentation complete the major components of the reactor.

The NSS is comprised of the following Engineering Critical Components (ECC):

- 1) Fuel Element (FE), EC-677566,
- 2) Cluster Hardware (CH), EC-677558,
- 3) Core Periphery (CP), EC-677564,
- 4) Support Plate and Plena (SPP), EC-677561,
- 5) Internal Shield (IS), EC-677562,
- 6) Reflector Assembly (RA), EC-677559,
- 7) Control Drum Drive Assembly (CDDA), EC-677585,
- 8) Structural Support Coolant Valve Actuator (SSCVA), EC-677576, and
- 9) Nuclear Subsystem Instrumentation (NSSI), EC-677565.

2.1.2 Content of Chapter

This chapter presents the results of nuclear analyses of both the composite and graphite candidate designs in support of the NSS for Data Item S-101. Section 2.2 summarizes the major nuclear characteristics of each design which provide a basis for design evaluation with respect to the NSS Specification Requirements itemized in Section 2.3. Section 2.4 describes both the analytical methods employed in this study and the results of an application of these methods to analyses of the PAX/R-1, NRX/EST, and PHOEBUS-2A reactors. These analyses address themselves to absolute reactivity, fuel loadings, power distribution, drum span, and cold-to-hot reactivity feedback. Based upon these analyses, the validity of application of these methods to the NSS design was determined. Finally this section investigates methods of reactivity adjustment to compensate for both cold and hot reactivity uncertainties. Section 2.5 presents in detail the results and analysis of the NSS.

2.2 SUMMARY OF RESULTS AND CONCLUSIONS

(U) This section presents the highlights of the nuclear analysis of the overall NSS. Analysis is discussed for both composite and graphite fueled reactors.

2.2.1 NSS Absolute Reactivity and Fuel Loadings

Composite Fueled Reactor

(U) The predicted NERVA maximum uranium loading for the NSS is 765 mg/cc for the composite core. Thirteen loading codes, ranging from a minimum of 463 mg/cc to the 765 mg/cc maximum loading, are utilized. These are based upon an ambient critical control drum position of 80 degrees, an initial \$1.5 poison shim content, and the reference NSS design given in Drawing 939J740.

(U) The predicted loadings are based upon R-θ EXTERMINATOR calculations. Initial verification of these predictions are planned in the near future utilizing LASL composite fuel in the PAX-G4 critical assembly.

(CRD) Significant reductions in the maximum composite fuel loading can only be achieved by increasing the reflector thickness and/or increasing the fuel graphite density from the present value of 1.3 g/cc.

Graphite Fueled Reactor

(CRD) Based upon the same calculational model, shim content, ambient critical drum angle, and reference NSS design, utilized for the composite fuel prediction, the predicted maximum uranium loading for graphite fuel is 620 mg/cc. These elements contain 43 grams of Nb in the NbC coating. This loading meets the 630 mg/cc loading requirement given in the Reflector Assembly Specification. Thirteen loading codes are also utilized in the graphite fueled reactor. They range from a minimum of 374 mg/cc to the maximum of 620 mg/cc.

(U) A statistical treatment of the uncertainty in these calculations has been performed utilizing comparison of analyses with measurements made in PAX-G reactor builds. A reactivity bias determined from this analysis is included in the graphite fuel loading predictions discussed above.

(U) A significant reduction in the maximum fuel loading can only be achieved by an increase in beryllium reflector dimension or reduction in the NbC coating.

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Absolute Reactivity Adjustment

A reactivity analysis was performed to evaluate the adequacy of absolute reactivity adjustment techniques expected to be available for in the NERVA engine and for R-1.

For the engine, the $\pm \$1.5$ shim capability of the In-Place Shim Adjustment (ISA) technique is adequate to limit the uncertainty in nuclear criticality to within $\pm \$0.15$. This conclusion is based upon the: 1) expected improvements in NDT methods, 2) production of identical engines, and 3) performance of an ISA for each reactor build.

A statistical treatment based upon NRX experience and present analytical uncertainties, indicates that the shim adjustment capability should be greater than $\pm \$1.5$. The PAX and NCX experimental program and improvements in NDT have a direct impact upon the final magnitude of the R-1 reactivity uncertainty.

2.2.2 Power Distribution

Radial Distribution

The NERVA radial power distribution for rated conditions was developed based upon the thirteen loading codes identified in Section 2.2.1 for the composite and graphite calculational models. The resultant power distribution satisfies the 1.1 power factor limit given in the Fuel Element Specification. The power factors range from a minimum of about 0.6 to a maximum of 1.1. A complete set of power factors for each fuel element channel which is based upon experimental PAX results is presented.

Analyses show that a reactivity loss of about $\$0.10$ results from fuel burnup after 10 hours of full power operation. This value, when added to the assumed $\$1.0$ reactivity loss from fuel element corrosion, gives a total reactivity loss of $\$1.1$ at the end of 10 hours. An insignificant power perturbation is calculated as the result of fuel corrosion, burnup and the increase in support stem flow necessary to compensate for the reactivity loss.

No significant reactivity loss is expected to result from fission product poisoning at the nominal hot critical angle. However, a peak reactivity loss of $\$0.12$ due to xenon buildup is expected after 1 hour of full power operation. This only becomes a consideration in a "dry" startup at end-of-life.

In going from full power to the throttled mode, a significant decrease of 7.6 percent in the peripheral power densities occurs due to the decreased peripheral hydrogen flow. From a core performance standpoint, the use of outward control drum movement to maintain a relatively flat radial power distribution is warranted. This outward control drum motion will result in a reactivity insertion of approximately \$0.75. This reactivity insertion would then require a corresponding decrease in support stem coolant flow.

Axial Distribution

The axial power distribution results in a peak-to-average power density ratio of about 1.4. The relative core inlet axial power density is 0.59 and 0.51 for the composite and graphite cores, respectively.

2.2.3 Control Drum Span

An examination of the reactivity control requirements was made to support the derivation and establishment of the overall drum span requirements. The results of this evaluation are consistent with the NSS minimum \$4.5 drum span requirement.

Based upon a statistical treatment of analytical versus experimental results, the predicted NERVA total control drum spans for the reference Reflector Assembly, Lateral Support, and Periphery designs are \$5.40 and \$5.35 for the composite and graphite fueled reactors, respectively. These predictions correspond to an 80 percent probability and a 95 percent confidence level. Both of these drum spans meet the \$4.5 minimum control drum requirement.

Final allocation of the available drum span for R-1 will require further evaluation of safety and operational considerations.

The above predictions do not include the following considerations:

- a) The analytical-experimental comparisons were primarily based upon the PAX experiments which did not include all significant R-1 design characteristics,
- b) NRX experience demonstrated a continuing decrease in drum span between successive reactors having the same basic design. The net reduction in measured drum span between NRX-A2 and XE-PRIME was \$0.80.

Inclusion of the latter consideration would result in drum spans of \$4.60 and \$4.55 for the composite and graphite fueled reactors in a flight engine.

Note that any reduction in reflector thickness in order to reduce drum span cannot be achieved without an increase in fuel loadings.

2.2.4 Cold-To-Hot Reactivity Feedback

Reactivity feedback models were developed for incorporation into the Transient Nuclear Thermal Code (TNT) for both composite and graphite cores. These detailed models include the effects of core, reflector, and peripheral hydrogen, core and reflector temperatures, and aluminum cylinder displacement. The resulting composite core reactivity feedback and drum worth reduction to rated conditions at start of life were calculated to be \$1.08 and 6 percent, respectively. The corresponding graphite model gave a full power feedback and drum worth reduction of \$1.23 and 10 percent, respectively.

The various components composing these feedback reactivities are contained in the following table:

FEEDBACK COMPONENTS (\$)		
	<u>Composite</u>	<u>Graphite</u>
Core Hydrogen	2.70	2.75
Core Temperature	-2.10	-2.00
Peripheral Hydrogen	+0.33	+0.34
Be Temperature	0	0
Barrel Contraction	+0.15	+0.14
	<u>+\$1.08</u>	<u>\$+1.23</u>

A statistical analysis of the expected operating reactivity feedback uncertainty for the NERVA engine yields a 0.999 probability to 95% confidence that the uncertainty will not exceed $\pm \$0.30$. Inclusion of the ambient uncertainty of $\pm 0.15\%$ gives a total operating reactivity uncertainty of $\pm \$0.35$. This operating reactivity uncertainty of $\pm \$0.35$ will be compensated by support stem coolant flow adjustment. The R-1 operating reactivity uncertainty, developed to be ± 80 cents, can be offset with a technique similar to that which will be used for the engine. However, in addition to the $\pm \$0.35$ that will be provided by support stem coolant flow variation, control drum motion equivalent to $\pm \$0.45$ may be necessary. A requirement for this effect has been factored into the control drum span analysis.

An alternate scheme for R-1 would be to increase the nominal support stem flow by 90 cents above the minimum position rather than the 45 cents as planned for the NERVA engine. In this scheme, the R-1 orifice drum angle would be reduced to compensate for the 45 cents additional positive reactivity feedback. The increased reactivity control available in the support stems would then be used to compensate for the entire R-1 cold-to-hot reactivity uncertainty.

2.2.5 Support Stem Reactivity Compensation at Full Power

Analysis of the support stem reactivity compensation at full power indicates a maximum reactivity swing of approximately \$4.0 for minimum to maximum support stem flow. This large reactivity swing has only a small effect on the radial power distribution. Any necessary adjustment of the poison shim content from the as-built \$1.5 value will effect the support stem reactivity swing by ± 9 percent per $\pm \$1.5$ of poison shim adjustment.

2.2.6 Nuclear Environmental Drawing

A nuclear environmental drawing was prepared which summarizes the key parameters including control drum angles and power factors for full power and throttled conditions, both for start and end of life. Also presented on this drawing are the control drum span, ambient critical control drum angle, cold-to-hot reactivity feedback and support stem reactivity swing.

2.3 REQUIREMENTS

2.3.1 Specification Identification

The governing specification for the NSS is Specification Number CP-677555A, "Part 1, Performance/Design and Qualification Requirements for NERVA NUCLEAR SUB-SYSTEM forming a part of CP 90290, Engine, NERVA 75K, Full Flow."

2.3.2 Applicable Requirements

The following three NUCLEAR SUBSYSTEM requirements are shown to be satisfied in Chapters 8 and 2, respectively, of this report.

3.3.1.3.2 Control Drum Reactivity Worth - For preliminary design, reactivity worth shall be based on an 80% probability of meeting the control drum reactivity span requirements specified in 3.1.1.1.5.1.2.1.2 (Nuclear Startup) with a confidence level of 95%.

3.3.1.3.3. Nuclear Criticality - The NSS shall achieve nuclear criticality within ± 0.15 of its specified design point at 540°R average core and reflector temperature without hydrogen flow. The reactivity shim content of the core shall not exceed -3.0 .

3.1.1.1.5.1.2.1.2. Nuclear Startup - The NSS shall provide the means to control reactivity to achieve criticality and control fission power within (tbd) and (tbd) Mw. This reactivity control shall be at least 4.50 at room temperature without propellant flow based on a minimum shutdown margin of 1.50 (at 540°R average core and reflector temperature).

The following NUCLEAR SUBSYSTEM Requirements require, as input, information from this chapter. Treatment of these requirements is discussed in the S-031, Thermal and Fluid Flow Analysis Report, and S-001, Systems Analysis Summary Report, NUCLEAR SUB-SYSTEM.

3.1.1.1.5.1.2 Startup Operations

3.1.1.1.5.1.2.1.1 Reactor Thermal Conditioning

3.1.1.1.5.1.3 Steady State Operation

3.1.1.1.5.1.3.1 Core Temperature Profile*

3.1.1.1.5.1.3.2 Temperature Controllability

3.1.1.1.5.1.4.1 Throttling

*From a nuclear analysis standpoint, this requirement is addressed by meeting a maximum 1.1 power factor as given in the Fuel Element Specification.

3.1.1.1.5.1.4.2 Throttle Hold
3.1.1.1.5.1.4.4 Pump Tailoff
3.1.1.1.5.3.2 Startup
3.1.1.1.5.3.4 Shutdown
3.1.1.1.16 Vehicle Checkout Operations
3.1.1.1.20 Space Station Operation
3.1.1.1.21 Propellant Depot Operation

2.4 METHOD OF ANALYSIS – DESIGN PRACTICE

This section describes the analytical models employed in the nuclear analysis of the overall NSS. Also included in this section is a discussion of the status/maturity of the data. The validity of the nuclear analysis is described and uncertainty factors applicable to the calculated nuclear data are developed.

2.4.1 Analytical Procedures

The purpose of this section is to describe: 1) the overall framework of nuclear, radiation, thermal, and structural analyses, 2) the relationship of the nuclear and radiation and shielding computations to this overall framework, and 3) the details of the nuclear procedures.

2.4.1.1 Overall Design Engineering Procedures

Figure 2-1 displays the flow chart of nuclear, thermal and structural analyses. In this figure, and in all subsequent figures describing analytical procedures, the following symbolism is employed:

- 1) An inverted trapezoid (shorter baseline on the bottom) denotes an input of information.
- 2) An upright trapezoid (shorter baseline on the top) denotes an output of information.
- 3) A rectangular block represents an operation.
- 4) All functions have a number in the upper left-hand corner which identifies the function on any subsequent flow chart.
- 5) When the operations performed for a given function are expanded into several subfunctions, the base identification number will remain the same and suffix numbers identifying the subfunctions will be added (i.e., the subfunctions for block 11 will be numbered 11-1, 11-2, etc.)
- 6) When a specific function is performed by an existing operational computer code, the name of the computer code will appear at the bottom of the function block.

The procedure for performing nuclear, radiation, thermal, and structural analyses has been broken down into four steps which are as follows:

- Step 1. The first step consists of developing or obtaining the data necessary to define the engineering model of the NSS.
- Step 2. The second step consists of translating the information in the engineering model of the NSS into the analytical models required to perform nuclear, thermal, and structural analyses.
- Step 3. The third step consists of employing the analytical model, developed in Step 2, to obtain nuclear, thermal and structural results for the NSS.
- Step 4. The fourth step consists of: 1) employing the detailed component models developed in Step 2 and the boundary conditions from the NSS calculational results developed in Step 3 to accomplish individual component thermal and structural analyses, and 2) reporting the results of nuclear, radiation, thermal, and structural analyses. Note that the nuclear analysis is performed only on the NSS as a whole and no component nuclear analysis is performed. The fourth step also briefly indicates the Design Engineering validation approach which includes: 1) comparing analytical and experimental results to obtain uncertainties in the calculational technique, and 2) employing experimental or other reference information to validate/improve the analytical model or technique.

Before a detailed description of the four steps followed in the analytic process, it is important to describe the incorporation of uncertainties and propagation of error methods throughout the analysis chain. These items, which constitute the essence of the reliability-related calculations, must begin with the input information of Step 1 being described probabilistically, namely, that drawings be a source not only of nominal dimensions but also of geometric uncertainties, that the materials properties be expressed as means, standard deviations, and (where available) test sample size, and that engine component simulation in the NSS performance models be described by nominal characteristics as well as uncertainties. The NSS level analysis, Step 3, is performed probabilistically in that the nuclear

analytic results include uncertainty estimates, while the thermal and structural analyses are performed in random trial, or Monte Carlo, fashion. The thermal and structural NSS computer representations are equipped with the necessary programming logic to select random values of all input parameters, perform a selected number of random trial (Monte Carlo runs), and reduce the resulting output data in terms of mean values and standard deviations. At present, 30 Monte Carlo cases for a given operating condition have been selected as a reasonable compromise between available resources and approach to an exact solution. In current practice, the Monte Carlo analyses are restricted to steady state conditions. The Step 4 component analyses must depend, of course, on the geometric and property information of Step 1 and boundary conditions as generated by the Step 4 gross NSS analysis. The probabilistic aspects of Step 1 and 4 as just described provide the raw material for probabilistic analysis at the Step 4 level. At this level of detailed component analysis, no fixed method of propagation of error analysis is followed.

Monte Carlo techniques were applied to every component finite difference or finite element model, required computer time would far exceed available program resources. The propagation of error analysis at the component level is performed by some scheme created by the ingenuity of the engineering analyst and best suited to the individual component analysis. Some suggested methods are available in NRPs 403, 411, and 605. The final result of probabilistic component analysis is, of course, a statistical description of component failure-governing "stress," stress in a broad sense which could actually be temperature, predicted reactivity loss, or structural load. The failure-governing stress is combined with the "strength," or the component ability to sustain the particular form of "stress," through the stress interference method as described in NRP 307. The interference analysis yields the probability of occurrence of a given component failure mechanism. The failure mechanism probabilities are indicated at the bottom of Figure 2-1 as a unique output of the probabilistic design process, distinct from the radiation environment, heating distribution, temperatures, pressures, stresses, and displacement which result from a conventional reactor analysis.

A brief description of each step follows.

From Figure 2-1, it is seen that Step 1 consists of the following inputs of information:

- 1) Block 1 is NSS and/or component drawings.
- 2) Block 2 is the bill of materials defining the type, quantity, and composition of individual parts.
- 3) Block 3 is material properties information which includes the nominal value and standard deviation for each property.
- 4) Block 4 is the run profile, state points, and engine component data information.
- 5) Block 5 consists of optional input data for malfunction and emergency mode which can be input into the analysis process in another reliability-related task, namely, prediction of the effects of engine component malfunctions.

The information from Blocks 1 through 5 is combined together to generate the engineering model of the NSS in Block 7. A Design Engineering goal is automation of this entire procedure. A computer description of each component and the bill of materials would be handled by the Weights And Moments of Intertia (WAMI) code to generate the weights, center of gravity, and moments of inertia for each component (Block 6). The WAMI code would also act as the file manager for the majority of the information in the engineering model and would provide the information needed to generate analytical models. The Nuclear, Thermal Stress (NUTS) code handles the material properties information and provides the information in a format which is directly useable by the design computation codes. Codes, such as the Design Performance Reliability (DPR) code, are being developed which will automatically generate probability of failure mechanism information via the Monte Carlo technique to evaluate propagation of errors (NRP 605) for the NSS as a whole. The DPR code is being developed as a specific tool to predict core assembly reliability.

A vast quantity of information is obtained from the drawings. The process of translating drawing information to computer input is being investigated. This process can be automated via the use of a digitizer. This machine takes the x and y coordinates of a point and/or any grouping of symbols and produces computer input from this information on punched cards (a magnetic tape is also available). Therefore, an operator using a digitizer, essentially

traces a drawing to produce a computer description of the component and any other information shown on the drawing. The computer description can then be used directly in calculations with WAMI to generate analytical models.

Step 2 in the nuclear, thermal, and structural analyses consists of developing the analytical models required by each discipline (Blocks 8, 9 and 10) from the information contained in the common engineering data bank (Block 7). This procedure may also be automated with the Nuclear Analytical Model (NAM), Thermal Analytical Model (TAM), and Stress Analytical Model (SAM) codes in order to make the optimum utilization of manpower and avoid duplication of effort. The NAM, TAM, and SAM codes are currently only names which may apply to several computer codes, some of which are already in existence, used to generate an analytical mode. The LOBO and MESHER codes that generate spatial mesh patterns for nuclear and thermal calculations have been programmed. The codes to translate data from WAMI into weights and atom densities for nuclear and thermal calculations have been formulated.

Step 3 in the nuclear, thermal, and structural analyses consists of performing the analysis related to the NSS as a whole. The nuclear analytical model developed in Block 8 is employed to perform the nuclear and radiation calculations in Block 11 (the details of the radiation portion of the nuclear computations will be described below). In addition, in order to perform the nuclear computations, the region average temperatures and hydrogen densities are required (Block 12) from the thermal analyses. However, note that the thermal analyses, both for the NSS and components (Blocks 13 and 19), require heating rates, fission distribution, and reactivity coefficients in order to be performed (Block 15). Therefore, as can be seen from Figure 2-2, the nuclear and thermal computations form a closed loop which require some iteration in order to obtain nuclear and thermal data.

The thermal calculations for the NSS (Block 13) employ the thermal analytical model (Block 9) and nuclear information (Block 15) to generate thermal data for the NSS and thermal-flow boundary conditions (Block 16) which are used in the thermal calculations for components (Block 19) and in the stress calculations for both the NSS (Block 14) and components (Block 21). The stress calculations for the NSS employ the stress analytical model

(Block 10) and the thermal-flow boundary conditions (Block 16) to generate the stress data for the NSS and the loads and displacement boundary conditions (Block 17) which are used in the stress calculations for components (Block 21).

Step 4 in the nuclear, radiation, thermal, and structural analyses consists of the following three phases:

- 1) Thermal and stress calculations are performed on a component level (Note that nuclear and radiation calculations are not performed on a component level).
- 2) Analytical and experimental results are compared to derive uncertainties or experimental results or other reference information are employed to improve the analytical model or technique (Block 18). This function provides feedback to the second step of this procedure.
- 3) Nuclear, radiation, thermal and stress computations are reported in the appropriate data item. In addition, the results of malfunction and emergency mode condition calculations for Monte Carlo analyses to evaluate failure mode probabilities with the aid of interference theory and/or malfunction analyses to evaluate FMECAs which have been obtained essentially the same calculational technique are reported.

The thermal calculations for component analysis (Block 19) employ the thermal analytical model (Block 19), the nuclear information (Block 15), and the thermal-flow boundary conditions from the NSS computations (Block 16) to generate thermal data for the components and the thermal boundary conditions (Block 20) for the components. The stress calculations for component analysis (Block 21) employ the stress analytical model (Block 10), the thermal-flow boundary conditions from the NSS and component analyses (Blocks 16 and 20), and the loads and displacement boundary conditions from the NSS computations (Block 17) to generate the stress data for the components. Radiation data are also provided to Thermal and Structural Analysis to modify radiation sensitive property data in the analyses.

2.4.1.2 Nuclear and Radiation and Shielding Design Procedure

Figure 2-2 displays the general outline of the procedure for performing nuclear

and radiation design analysis for the NSS (Block 11). The atom densities and problem spatial mesh description, which constitute the nuclear analytical model (Block 8), and the temperatures and hydrogen densities (Block 12) are employed to generate neutron spectra and neutron and photon cross sections (Block 11-1). These cross sections are employed to generate neutron and/or photon fluxes over the NSS (Block 11-2). This information is then processed to obtain:

- 1) the reactivity analysis for the NSS (Block 11-3) which includes determining the absolute criticality of the system and reactivity shimming,
- 2) the nuclear kinetic response for the NSS (Block 11-4) which consists of determining cold-to-hot reactivity changes,
- 3) the control drum worth (Block 11-5),
- 4) the power map and fuel zoning (Block 11-6),
- 5) the internal radiation environment (Block 11-7),
- 6) the radiation heating rates (Block 11-8),
- 7) the radiation sources (Block 11-9), and,
- 8) the external radiation environment (Block 11-10).

The information from Blocks 11-3, 11-4, and 11-5 and the results from Blocks 11-6, 11-7, and 11-8 which are processed by additional computer codes from the heating rate, fission distribution, and reactivity coefficient (Block 15) data required to perform thermal analyses. The data in Blocks 11-3 through 11-10 are reported in the appropriate data item (Block 22) and are compared with experimental results to obtain uncertainties in the analytical results and experimental results are used to improve the analytical technique (Block 17).

2.4.1.3 Nuclear Design Procedure

The WANL Integrated Nuclear, Radiation and Shielding Standard Design Method (WISDM)⁽¹⁾ was employed to obtain all the analytical results presented in this Data Item. The following sections describe the details of the nuclear analysis and the computer codes used. The description of the Nuclear Analyses has been separated into the two subsections for ease of presentation, understanding, and discussion. The two subsections will present, respectively, the analytical procedures for determining the following information:

- 1) NSS Absolute reactivity, fuel loadings, power distribution, and control drum span, and
- 2) cold-to-hot reactivity feedback.

Analytical Procedures for NSS Absolute Reactivity, Fuel Loadings, Power Distribution, and Control Drum Span Analysis

This section describes the analytical procedures for calculating the neutronic effects listed above. In this procedure, regionwise neutron energy spectra are generated for a reactor configuration for 52 broad energy groups. These spectra are generated through two iterations of the LINK code which simultaneously performs the GAMBIT, ANISN, and ANISIG calculations. Subsequent neutronic calculations are performed in 16 energy groups weighted by 52 broad energy spectra. ANISN is used for one-dimensional analyses, while EXTERMINATOR-2 is used for studies in two dimensions.

The procedure for the calculation of converged WISDM regionwise spectra is shown in Figure 2-3. Converged regionwise spectra are developed through iterative use of the LINK-2⁽²⁾ code system, an automated package of three computer programs. The first program, GAMBIT⁽³⁾ (Reference Number 1 of Figure 2-3) provides cross section averaging. In the first iteration, the WISDM procedure calls for the use of an infinite medium (P₁) flux and current spectrum for weighting of the fine group (68 fast and 93 thermal) data in the Nuclear Properties data file. The infinite medium flux and current spectrum is established (internal to GAMBIT) from the atom densities of the constituents of the region in question. For each region in the "Nuclear Model," a 52-group set of macroscopic cross sections is calculated and placed directly on magnetic tape or disc.

The second step in the LINK calculation is a multigroup solution of the neutron flux distributions in a reactor model using these macroscopic regionwise neutron cross section data in the ANISN⁽⁴⁾ code (Reference Number 2 of Figure 2-3). The purpose of this solution is the recalculation of the neutron flux and leakage (for current) spectra for each region in the system as it interacts with the rest of the system. The radial one-dimensional ANISN calculations are performed with a poison ring to simulate the control vanes in the reflector of the NERVA reactor. The scalar and angular neutron fluxes from the ANISN calculation are processed in ANISIG-1⁽⁵⁾, the third code in the LINK package, (Reference Number 3 of Figure 2-3) through use of a magnetic tape or disc.

ANISIG-1 serves three functions in the WISDM scheme. First, it computes regionwise neutron current in 52 groups from the ANISN angular flux data. Next, it interpolates the 52-group data to obtain flux and current data for 68 fast groups in the fast energy range (10 MeV to 0.414 eV) and 93 groups in the thermal energy range (1.86 eV to 0.0 eV). In addition, the ANISIG-1 code computes the conventional standard deviation of the groupwise percent differences between the neutron flux and current in the present and the previous iteration of the spectrum convergence. (This last computation is bypassed in the first "iteration.") The purpose of this last step in ANISIG-1 is to determine if the regionwise flux and current spectrum are converged and if the iterative process shown in Figure 2-3 may be terminated.

Experience to date in spectrum convergence indicates that two iterations are required for satisfactory convergence⁽¹⁾. Regionwise neutron cross sections computed by GAMBIT in the third LINK iteration using neutron flux and current data from the second iteration do not significantly differ from those cross sections developed in the previous iteration.

The nuclear analytical procedures for the calculation of the absolute reactivity, fuel loading, power distribution, and total drum span are summarized in Figures 2-4, 2-5, and 2-6. Figure 2-4 presents the procedural flow chart for the calculation of the system reactivity (maximum fuel loading). Figure 2-5 shows the flow chart for the calculation of the radial power factors. Figure 2-6 presents the procedural flow chart for the calculation of the total control drum span. The axial power distribution is calculated by a one-dimensional procedure described in Chapter 6. Using the above converged neutron spectra from the LINK

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program in the GAMBIT code, macroscopic region dependent 16-group neutron cross sections are generated in the EXTERMINATOR⁽⁶⁾ and ANISN formats (Reference Number 1 of Figures 2-4, 2-5, and 2-6). Absolute reactivity, control drum span, and the radial fission distribution are then calculated by the EXTERMINATOR code using the above mentioned cross sections (Reference Number 2 of Figure 2-4, 2-5, and 2-6) in conjunction with the Nuclear Model.

Absolute reactivity is provided as output from EXTERMINATOR with the control drum bank at the cold critical position. The total control drum span is calculated by using the k_{eff} s from EXTERMINATORS at the 0° and 180° drum positions. The differences in these k_{eff} s is divided by β to obtain the calculated control drum span in dollars. This calculated value is normalized by an experimental correction factor which includes 80% probability and 95% confidence. The radial fission distribution is calculated by EXTERMINATOR. As shown in Figure 2-5, the NERVA EXTERMINATOR fluxes are input to the PFENCER-1⁽⁷⁾ code (Reference Number 3 of Figure 2-5). These are used in conjunction with the experimentally determined PAX/R-1 correction factors (PFENCER-2, Reference Number 4 of Figure 2-5) to determine the radial power factor data in the individual coolant channels (PFENCER-3, Reference Number 5 of Figure 2-5). These power factors are used by Thermal Design as input to orifice the fuel channels. As stated previously, the complete analytical procedure was carried out for both the composite and graphite fuel element designs.

Analytical Procedures for Cold-To-Hot Reactivity Feedback and Power Distribution Change

This section describes the analytical procedures for calculating the cold-to-hot reactivity feedback and radial power changes due to reactor operation.

The analytical procedure for the analysis of cold-to-hot reactivity feedback and power distribution changes due to reactor operation involves the calculation of the separate and combined effects of the following mechanisms:

- 1) Core temperature via core thermal expansion, thermal base spectral shift and resonance absorption doppler broadening,
- 2) Core hydrogen via increased neutron moderation and reduced leakage,
- 3) Peripheral hydrogen via absorption and moderation,

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- 4) Reflector temperature via reduced upscatter and increased transparency at neutron energies below the beryllium Bragg Cutoff ($<.005$ eV),
- 5) Control drum worth as a function of reactor operating conditions.

(U) The overall flow chart defining the nuclear analytical procedure employed in the NUCLEAR SUBSYSTEM analysis is shown in Figure 2-1. Blocks are numbered where applicable for reference to the overall Nuclear and Shielding Design Procedure Flow Chart of Figure 2-2. Circled numbers are provided for reference in the discussion below.

(U) Previous cold-to-hot reactivity calculations have involved the generation of macroscopic, regionwise homogeneous cross section sets. However, the number of reactors analyzed in this report, the desirability of having consistent analyses and the possibility of future design changes all suggested a different approach was in order for this study. To this end, a master tape of microscopic, 52-group, P_1 scatter cross section sets, applicable to all reactors under consideration, was generated using the GAMBIT computer program (Reference Number 1 of Figure 2-7). The microscopic cross section sets of neutron resonance absorbing materials were generated for various absorber atom concentrations (various σ_p) covering the range of interest to this study. The resonance absorption cross sections are sensitive to the parameter σ_p defined as the total regionwise macroscopic scattering per absorber atom.

(CRD) This microscopic library is then used in conjunction with the reactor region atom densities and the mixing option of the SATURN computer program to generate regionwise dry (without hydrogen) macroscopic cross section set libraries for each reactor applicable to the calculation of the reactivity effects under consideration (Reference Number 2 of Figure 2-7). Using this technique, a considerable number of macroscopic core cross section sets were generated so as to separate core temperature feedback reactivity into its various components (thermal base shift, U, Nb, Ta, Zr, Mo doppler broadening) over a broad temperature range of 100° K to 1600° K. Hydrogen cross sections were retained in their microscopic form so as to take advantage of the flexibility possible with the use of the mixing option of the transport solution program (ANISN) discussed below (Reference Number 3 of Figure 2-7).

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(U) Finally, the macroscopic cross section libraries were combined to produce a master library tape to be used in conjunction with the ANISN computer program for the calculation of reactivity feedback effects and radial fission power distribution (Reference 4 of Figure 2-7). This constitutes completion of the flow diagram illustrating the analytical procedure to calculate the cold-to-hot reactivity feedback and power perturbation models.

(U) The analytical procedure described above was examined for consistency with a direct macroscopic cross section generation approach by comparison of certain PHOEBUS-2A reactivity results using both methods as shown in Table 2-1. The table compares the results of ambient base core and hot (1600° K) core calculations with control drums removed. As is noted, the discrepancy in absolute eigenvalue is essentially identical in both case and is less than 15¢. Thus, although there is a small discrepancy in absolute eigenvalue between the two methods, reactivity differences which are germane to the analysis of cold-to-hot feedback appear to be unaffected by choice of method.

(CRD) The basic design features of the various reactors analyzed are similar with the primary difference being size and structural support member design. In general, the radial geometry in order of sequence from the core periphery can be considered as consisting of a seal region (include all hydrogen flow between core periphery and support structure), a support structure and the beryllium reflector. The analytical approach involved the calculation of the separate and combined reactivity effects of core, seal, support structure, and reflector hydrogen contents and core and reflector temperatures all as a function of control drum position. Drum interaction effects are examined by performing identical calculations with and without the presence of a control poison sheath. Reactivity differences are then assumed to vary linearly with ambient control drum worth to removal. As an example, consider the PHOEBUS-2A reflector hydrogen results shown in Table 2-2. The reactivity feedback due to 4 pounds of reflector hydrogen was calculated at -12¢ and -45¢ at a 91.2° drum bank angle and without drum, respectively. The ambient worth between 91.2° and removal is \$4.56. As illustrated in Table 2-2, the feedback at the full power critical drum bank of 136.5°, which corresponds to \$2.00 to removal, would then be approximately -30¢. The increase in negative reflector hydrogen feedback with outward drum rotation indicated by this example is equivalent to a decrease in control drum worth (in this case, $(45-12)/456 = 7.2\%$).

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2.4.2 Analytical Model

Two models are required to perform the nuclear analysis of the NSS. Specifically, these models are subdivided for two-dimensional and one-dimensional analyses. Each model will be discussed in detail in the following sections.

2.4.2.1 Two-Dimensional Analytical Model

This section discusses the analytical models utilized to perform the nuclear evaluation of the NSS absolute reactivity, fuel loading distribution, power distribution, and drum span. As stated previously, detailed calculations were carried out for both the composite and graphite fuel element designs.

Composite Core Model

This calculational model⁽⁸⁾ utilizes the externally coated composite fuel which is the reference fuel design. Table 2-3 presents the number of elements, radii, and volumes for each region with the R-1 multizoned total core material inventory. Table 2-4 presents the corresponding ambient core number densities by region.

Graphite Core Model

This calculational model⁽⁹⁾ utilizes the externally coated graphite fuel design. Table 2-5 presents the number of elements, radii, and volumes for each core region with the R-1 multizoned total core material inventory. Table 2-6 presents the corresponding ambient core atom densities by region.

Reflector Model

The non-corrodible periphery, lateral support, and reflector designs of the reference model are documented in this section. The reflector model is compatible with both composite and graphite fueled cores described above. Figure 2-8 shows the geometry used for the non-core regions. Table 2-7 lists the corresponding non-core region material inventory, while Table 2-8 gives the corresponding number densities.

The outer region geometry shown in Figure 2-8 corresponds to a 10° sector mock-up of an R- θ EXTERMINATOR problem. In the EXTERMINATOR model, all radial regions are explicitly represented. In calculations for the 0° and 180° control drum positions, a 10° sector is simulated. The use of symmetric boundary conditions in these 10° calculations along the outer radial mesh lines allows the full 360° cylindrical system with appropriately positioned vanes to be prepared internally to EXTERMINATOR-2. Thus, by putting the drum centerline on an exterior boundary which represents one-half of the 0° or 180° position, the entire 18 drum system is obtained. For drum positions other than 0° or 180° , it is necessary to utilize a 20° sector representation of the entire drum area in conjunction with the periodic boundary condition. Thus the counterclockwise rotation of the vane relative to the core is maintained on each of the 18 drums.

In the EXTERMINATOR model, the control vane area is simulated to within ± 1 percent of the true physical area. This is done by defining an extremely fine mesh ($0.25^\circ \times 0.100$ inches) throughout the vane region. The spring and plunger regions are separated from the aluminum cylinder and beryllium reflector regions and represented explicitly in the R- θ geometry. In the reference model, the spring region which previously extended over the entire beryllium reflector thickness has been further subdivided in a spring and additional plunger region. This division provides a better representation of the actual location in the reflector of the A-286 spring material.

2.4.2.2 One-Dimensional Analytical Model

This section discusses the analytical model used to calculate the cold-to-hot reactivity feedback. This analytical model is based on a one-dimensional (radial) transport solution with 52 group P1 scatter cross section sets. Transverse (Axial) leakage is treated by the regionwise addition of fictitious DB^2 terms to the absorption cross sections. Control drums are represented by a poisoned 100 mil annular sheath located approximately at the drum centerline. Drum rotation is affected by a change in sheath radial position. The detailed atom densities and reactor geometry used in the analysis are discussed below for the composite and graphite cores.

Composite Core Model

The model described in this section utilized the externally-coated composite fuel and incorporates the reference non-corrodible periphery, lateral support, and reflector design. This 5-region core model is a collapsed version of the 15-region core model presented in Section 2.4.2.1. The reflector is also a one-dimensional model based on the two-dimensional reflector model also shown in Section 2.4.2.1. Table 2-9 presents the composite R-1 core material inventory by zone. Table 2-10 lists the ambient R-1 core number densities which correspond to the masses by region in Table 2-9. The radial region boundaries and volumes are included in Table 2-10. The radial non-core region materials for the aluminum structure reflector are given in Table 2-11 with the corresponding number densities in Table 2-12. The region definitions are given in Table 2-12.

Graphite Core Model

The graphite fueled core regions described in this section utilize the externally-coated graphite fuel design. This 5-region core model is a collapsed version of the 15-region core model presented in Section 2.4.2.1. The one-dimensional non-core regions which include the non-corrodible periphery, lateral support, and reflector are the same as those described in Tables 2-11 and 2-12 of this section. Table 2-13 lists the graphite R-1 core material inventory by zone, and Table 2-14 gives the ambient R-1 core atom densities corresponding to these masses by region. The radial boundaries and volumes are included in Table 2-14.

Core and Aluminum Barrel Radial Displacements

Table 2-15 lists composite core radial displacement data at both rated full power start of life (SOL) and at 40 seconds into the SOL startup (chamber temperature = 2520°R). Table 2-16 lists the corresponding composite support barrel displacement data. These data were obtained from the RATS Data run TAT-13 as generated per Stress/Thermal analyses in support of this PDR. The 40 seconds startup hold was chosen to represent the maximum barrel displacements experienced during startup.

Table 2-17 lists graphite core radial displacement data at both rated full power SOL and at 39.5 seconds into the SOL startup (chamber temperature = 2433°R). Table 2-18

lists the corresponding graphite support barrel displacement data. These data were obtained from RATS Data run TAT-14. Again the 39.5 seconds startup hold represents maximum barrel displacements.

2.4.2.3 Problem and Tape Log

A list of the various program tapes, the problem log, and corresponding output tapes pertinent to this analysis can be found in DRM-53365⁽¹⁰⁾ Source references to these problems are found in the lower right corner of all tables and figures where applicable. Basic input cross section data tapes are listed below under Section 2.4.3.

2.4.3 Status/Maturity of Input Data

This section presents the material property and nuclear cross section data which have particular application to the NUCLEAR SUBSYSTEM analysis. Basic material density data are referenced in detail elsewhere in this report and are not repeated here.

2.4.3.1 Composite Core

2.4.3.1.1 Materials Properties

This section presents the basic material properties and design assumptions utilized in the two- and one- dimensional nuclear analysis of the NSS. Table 2-19 summarizes the material properties, design parameters, and assumptions employed in the development of the composite core model used in this analysis. Table 2-20 shows the chemical composition of the stainless steels, A-286, and SS-304, utilized in the support stem assembly calculations.

2.4.3.2 Graphite Core

2.4.3.2.1 Material Properties

This section presents the basic material properties and design assumptions utilized in the two- and one- dimensional nuclear analysis of the NSS with the graphite fueled core. Table 2-21 summarizes the material properties, design parameters, and assumptions employed in the development of the graphite core model used in this analysis. For items which apply to both the composite and graphite fuel designs, see Table 2-19 of Section 2.4.3.1.1. As noted in Item 6, Figure 2-9 shows the variation in fueled element carbon weight as a function of the fuel element uranium loading. The chemical compositions of the stainless steels shown in Table 2-20 were also used in the graphite core calculations.

2.4.3.3 Neutron Cross Section Data

Table 2-22 lists the basic neutron cross section data tapes employed in the NSS Analysis for the two-dimensional analysis.

Table 2-23 lists the basic cross section data tapes employed in the NUCLEAR SUBSYSTEM one-dimensional analyses. The GAM library tape FSGAM29 includes an extended treatment of U-235 unresolved resonances to 167 eV as discussed in WANL-TME-1920.⁽¹¹⁾

2.4.4 Validity of Analysis

The validity of the basic analytical procedures described in section 2.4.1 and shown schematically as Block 18 in Figure 2-1 was verified by performing experiments and comparing analytical predictions with corresponding experimental results. The PAX/R-1 experimental results are compared in this section with absolute reactivity, power distributions and control drum span calculations. The PAX/R-1, PHOEBUS-2A and NRX experimental results are compared to calculations for the cold-to-hot reactivity feedback evaluation. In the second major subsection, "Uncertainty Analysis", a detailed treatment is presented of the sources of uncertainty and the techniques that have been devised to compensate for these uncertainties. Both the ambient reactor and the reactor at the design operating point are analyzed. In the concluding subsection an evaluation is given of the adequacy of the reactivity compensation techniques that have been devised to minimize the impact of these uncertainties upon NSS performance.

2.4.4.1 Comparison with Experimental Results

The comparisons presented in this section are related to calculations in the following areas:

- 1) absolute reactivity and fuel loading,
- 2) power distribution,
- 3) control drum span, and
- 4) cold-to-hot reactivity feedback.

Since many of the measurements used in these comparisons were made in the PAX/R-1 assembly (PAX-G series of mock-ups), these assemblies are briefly described below for orientation.

The PAX/R-1 assembly at WANEF is the critical assembly utilized for evaluating the neutronic effects of the R-1 design. Measurements are primarily performed in a 60 degree test sector which is modified periodically to mock up the particular experimental design. These modifications have been designated G0 through G3 and will be described

below. Basically, the PAX consists of a beryllium reflected reactor with a core consisting of NRX vintage fuel elements. The R-1 reactor core design contains 1878 fueled and 349 unfueled elements. The PAX/R-1 also contains 2227 elements, however, approximately half of the central elements are fueled with the highest NRX loading (460 mg/cc) in order to simulate R-1 loadings. The simulated PAX/R-1 maximum loading is 568 mg/cc versus the R-1 graphite loading of approximately 620 mg/cc. Reactor control is obtained through the rotation of beryllium control drums which contain B-10 neutron absorbing plates on one side.

PAX-G0, the original PAX/R-1 build, had a core configuration as described above. The reflector system for this reactor consisted of twelve 30° KIWI beryllium sectors, 4.5 inches thick with a non-beryllium fraction of 0.2. The control system had twelve NRX-type drums, each having a 3.85 inch vane outer diameter. PAX-G0 provided a base for subsequent reactivity worth measurements.

The PAX-G0 was modified to mock up in the test sector the R-1 18 drum reflector system and designated PAX-G1. While the core remained basically the same, the R-1 reflector was mocked up by replacing two 30 degree KIWI sectors of the 12 drum NRX configuration with a 60 degree R-1 reflector mock-up segment. This reflector segment was constructed of laminated beryllium slabs and contained three control drums with an outer diameter of 3.14 inches. Figure 2-10 shows the PAX-G1 reflector assembly.

The conversion from PAX-G1 to PAX-G2 included the substitution of Cu-B^{10} control vanes for the Al-B^{10} vanes in the 3 test sector mock-up drums. In the core, the TaC shims and pencils were redistributed uniformly to the core periphery. All "X" material mock-up wires were removed consistent with the R-1 fuel design and the core reshimmed with Nb wires.

The PAX-G2 was modified to update the mock-up of the R-1 reference reflector and designated PAX-G3. Graphite fillers were added in the test sector just outside the core periphery. Adjacent to these fillers, aluminum plates were placed to simulate the mass of the R-1 reflector barrel. As a result, the 60 degree beryllium segment was moved out so that the

control drums better approximated the R-1 drum centerline. Beryllium rods were added on both sides of the beryllium segment to fill the resulting azimuthal gaps. Figure 2-11 illustrates the R-1 reflector mock-up PAX-G3 configuration.

Several major differences exist between the PAX/R-1 configuration and the reference R-1 in addition to the overall fuel loading difference. The control drum diameter for the three test sector control drums is 3.14 inches compared to 4.0 inches in R-1. The Cu-B^{10} vane material is 24 v/o for PAX/R-1 yielding a surface density of 0.13 gm/cm^2 for 100 mils thickness compared to 18 v/o Cu-B^{10} and 0.100 gm/cm^2 respectively for R-1. Also the spring and spring holes are significantly larger in the R-1 compared to PAX/R-1 design. Another significant difference exists in the core periphery. The R-1 has a non-corrodible periphery with a 6 mil "X" wrapper while the PAX has NRX-type graphite filler strips.

2.4.4.1.1 Absolute Reactivity and Fuel Loading

The following experiments were performed with the PAX/R-1 mock-ups described above in support of R-1 absolute reactivity and fuel loading calculations:

- 1) PAX-G0 Approach to Critical
- 2) PAX-G0 Reactivity Coefficient Measurements
- 3) PAX-G1 Reflector Interchange
- 4) PAX-G1 Reactivity Effect Measurements
- 5) PAX-G2 Criticality
- 6) PAX-G3 Criticality

References 12 through 17, respectively, provide a detailed description of these experiments and the supporting nuclear analysis. A summary of the results of a comparison between measured and calculated absolute reactivity is presented in Table 2-24 for graphite fueled systems. The source of the various entries in Table 2-24 is discussed at length in Section 2.4.4.2.1.3 where a statistical treatment is given of the numerical values in the column labeled "Difference". The average difference, calculated to be $\pm 1.56\%$ corroborates a

previous WISDM PAX-E1 analysis where the difference in absolute reactivity between calculation and measurement was observed to be + 1.5\$⁽¹⁾. PAX-E1 was a WANEF mock-up that simulated the NRX-A6 reactor.

The observed error in calculated absolute reactivity is used to adjust the fuel loading required for criticality. An adjustment for the 1.56\$ bias was, thus, appropriate for the graphite core fuel loadings. No adjustment was made, however, in fuel loadings for the composite core pending identification of the magnitude of the analytical bias for this system.

2.4.4.1.2 Power Distribution

The following PAX/R-1 experiments were performed in support of the R-1 power distribution calculations:

- 1) PAX-G0 Power Distribution Measurements,
- 2) PAX-G1 Gross Power Measurements, and
- 3) PAX-G2 Fission Flux Distribution Measurements.

References 18, 19 and 16 respectively, provide a detailed description of these experiments and the corresponding nuclear analysis performed for comparison. A statistical analysis in Reference 1 indicated that the errors in calculation were very similar in these reactors and that PAX-G2 was typical. Table 2-25 gives a comparison of the PAX-G2 normalized predicted and measured fission per gram values at the position of the center channel of each fuel element. The ratio of the measured to predicted value (termed "NCi") is shown graphically in Figure 2-12 as it varies with distance from the core centerline.

2.4.4.1.3 Control Drum Span

The following PAX/R-1 experiments were performed in support of the R-1 control drum span calculations:

- 1) PAX-G0 Control Drum Span,
- 2) PAX-G1 Control Drum Span,

- 3) PAX-G2 Control Drum Span, and
- 4) PAX-G3 Control Drum Span.

References 13, 14, 20 and 17, respectively, provide a detailed description of the experiments and the corresponding nuclear analysis performed for comparison. Results are summarized in Table 2-26 along with a similar calculation⁽¹⁾ performed for the NRX-A6 reactor. With the exception of NRX-A6, the calculated drum spans are based on a model which simulates only the test sector. In the case of NRX-A6, an azimuthally uniform system, the reactor simulated in the analytical model was based on the average composition of the entire reactor. In section 8.4.4 a statistical treatment is presented of the ratio of the measured to the calculated drum span values which appear in the last column of Table 2-26. This statistical analysis provides a ratio value to be applied to calculated drum spans in order to assure that the calculated drum span will satisfy the requirement to an 80% probability at the 95% confidence level.

2.4.4.1.4 Cold-to-Hot Reactivity Feedback

The following subsections describe the results of analysis performed to validate techniques used in the calculation of cold-to-hot reactivity swing resulting from operating the NSS at rated conditions. Experimental results from the PAX/R-1, NRX/EST* and PHOEBUS-2A reactors are used for this validation.

2.4.4.1.4.1 PAX/R-1 Analysis

The PAX/R-1 assembly is essentially a zero power reactor operated at ambient conditions and as such can provide only limited data that may be used to validate cold-to-hot reactivity swing methods. Experiments have, however, been planned to evaluate the dependence of the material temperature coefficient on type of fuel.

The discussion that follows describes an ambient experiment⁽²¹⁾ and the corresponding analytical work related to the reactivity worth of hydrogen in the reactor core. The

experiment involved the placement of 85 polyethylene wires into the PAX/R-1 reactor in

* Analysis is presently incomplete.

roughly a uniform distribution. Reactivity worth was determined by period pickoff techniques. The results indicated a uniform hydrogen worth of 2.35\$/lb at approximately a 115° drum bank angle. This result was compared with a PAX/R-1 calculated uniform core hydrogen worth of 2.13\$/lb with control drums removed by assuming the calculated graphite R-1 interaction control drum effect discussed in Section 2.5. The analysis indicated a drum worth reduction of 4.8% per pound of core hydrogen. The PAX/R-1 drum worth from 115° to removal is \$3.61 (based on a total span of \$6.26 and a 180° to removal worth of \$1.80). The calculated hydrogen worth at removal of 2.13\$/lb can be extrapolated to the 115° drum bank position by the following equation:

$(H_2 \text{ worth at } 115^\circ) + (\text{actual drum worth to removal}) = (H_2 \text{ worth at removal}) + (\text{ambient drum worth to removal})$ or on a per pound basis with the 4.8% per pound drum worth reduction factor mentioned above,

$(H_2 \text{ worth at } 115^\circ) + (.952) (\$3.61) = \$2.13 + \3.61 giving 2.30 \$/lb for the calculated result at 115° which compares well with the measured result of 2.35 \$/lb.

2.4.4.1.4.2 NRX Analysis

The NRX/EST full power endurance test of March 1966 and the XE-PRIME reflector experiment of August 1969 were selected as part of the methods evaluation analysis. One objective is to compare the measured NRX/EST reactivity feedback to full power and the low temperature feedback and drum worth measurements of XE-PRIME with the corresponding results obtained by employing the analytical procedure outlined in Section 2.4.1. This comparison will help ascertain the validity of the method in the prediction of cold-to-hot effects. The analytical procedure will also be applied to the calculation of XE-PRIME xenon poisoning and the comparison with measurements. This comparison will provide a means of validating the R-1 xenon reactivity drum startup requirement as discussed in Section 2.5.

This analysis has not been completed. The master macroscopic NRX/EST cross

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section tape has been generated and completion of this analysis is pending the calculation of the appropriate transport eigenvalue solutions (ANISNs) and the subsequent comparison with the experimental results.

2.4.4.1.4.3 PHOEbus-2A Analysis

The PHOEbus-2A series of reactor tests conducted by LASL in 1968 provided data for methods evaluation as pertaining to cold-to-hot reactivity analysis. Of particular interest in this regard were the important and somewhat complicated neutronic effects associated with peripheral (lateral support, support structure, reflector) hydrogen flow exhibited by the PHOEbus-2A test results. The analytical procedure outlined in Section 2.4.1 was employed in an attempt to duplicate these effects at selected test holds. Core operating reactivity feedback contributions were not calculated but were investigated for possible interaction effects with the peripheral reactivity feedback. Reactivity feedback comparison with the test data was made using post-test predictions of core reactivity feedback contributions as published by LASL⁽²²⁾ and peripheral feedback contributions as calculated per this analysis. This analysis has been discussed in detail in reference 23. Major conclusions are summarized below.

The analysis indicates approximately a 32% reduction in drum worth at full power which is in good agreement with the measured results illustrated in Figure 2-13. The reactivity feedback results are in excellent agreement with the test results at high power holds but exhibit a substantial discrepancy at low power holds. This is illustrated graphically in Figure 2-14 which correlates the reactivity discrepancy (predicted - observed) with reflector hydrogen gas temperature. The discrepancy at low power is probably attributable to ambiguity in instrument performance, uncertainty in flow/temperature estimates and the possibility of para/ortho hydrogen species conversion.

In conclusion, the application of the analytical procedure, as outlined in Section 2.4.1, to the prediction of the PHOEbus-2A test results showed generally good agreement. In particular, measured control drum interaction effects and peripheral hydrogen feedback at high power holds were successfully predicted.

2.4.4.2 Uncertainty Analysis

(U) An uncertainty analysis is given in this section in the following areas:

- 1) absolute reactivity,
- 2) power distribution,
- 3) control drum span, and
- 4) cold-to-hot reactivity feedback.

(U) In the first subsection, the absolute reactivity discussion has been divided into the uncertainties expected in composite and graphite fueled cores. In each case there is a discussion of the ambient and operating reactivity uncertainty for both the R-1 and the engine system configuration. The discussion covers the reactivity effects of 1) the allowable manufacturing variations and 2) the uncertainty in nuclear calculation. A concluding section deals with the various reactivity uncertainty compensation techniques. These include 1) fuel element selection, 2) reactivity shimming and 3) support stem assembly flow variation.

2.4.4.2.1 Absolute Reactivity Uncertainty

2.4.4.2.1.1 Composite Core Ambient Reactivity Uncertainty

(U) The uncertainty in the ambient reactivity is derived in this section from consideration of 1) the allowed manufacturing variations and 2) the uncertainty in the WISDM absolute reactivity prediction. Note that, implicit in this latter area is the contribution of uncertainties in the true reactor component masses, i. e., errors as a result of NDT assay.

The R-1 reactor and the engine system are treated separately.

R-1 Reactor, Allowable Manufacturing Variations

(CRD) Table 2-27 summarizes the allowable manufacturing variations as defined by the fuel element drawings for the constituents in the fueled and unfueled support elements in the composite core. The values have been presented on a percent and on a gram per element

basis. Since the true masses of the constituents in each fueled and unfueled element will be known (to within the NDT accuracy) and will fall within the allowed variation band, the quoted core variation is taken to be the product of the square root of the number of elements and the variation per element. This value represents a statistically nominal condition and would only apply to a situation where there was no bias in manufacturing. The reactivity effect associated with mass variations of this magnitude is as follows:

Constituent	Mass Variation (+), kg	Reactivity Coefficient, ρ /kg	Reactivity Effect, ρ
Uranium	0.19	+ 11.5 ²⁴	2.2
Zirconium	1.84	+ 0.05 ¹⁵	0.1
Carbon	0.64	+ 4.3 ²⁴	2.7
Statistical Sum			<u>+ 3.5 ρ</u>

Thus, if the variations in constituents of the fueled and unfueled elements are distributed randomly in the allowable manufacturing variation band, the variability in the total core inventory is rather small with a reactivity worth of + 3.5 ρ .*

(CRD) However, this situation was not the experience of the NRX reactor test series. For these NRX graphite type fueled reactors, an average fuel reactivity difference (between as-specified and as-built fuel elements) of +14.3 ρ was observed with a standard deviation of +84.8 ρ . For details refer to Section 2.4.4.2.1.3. While this graphite fuel experience is judged typical of the non-zirconium composite fuel constituents, the presence of zirconium prevents the direct use of the graphite experience except as a general guide for what could be expected from allowable manufacturing variations.

* The input variations used in this section of the report are derived mainly from drawings and specifications. For purposes of propagation of error, it is assumed that these variations are consistent so that regardless of the choice of a standard deviation multiplier, K, it is used throughout and fits the following: $(\text{Variation})^2 = K^2 \sigma_1^2 + K^2 \sigma_2^2 + \dots + K^2 \sigma_n^2 = K^2 \sum_i \sigma_i^2$

$$\text{Variation} = K \sqrt{\sum_i \sigma_i^2}$$

Note that without assuming consistency, the variation could not be readily determined.

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R-1 Reactor, Uncertainty in Nuclear Calculations

(CRD) The uncertainty in nuclear calculations which includes NDT inventory for the composite fuel is undeveloped at this time since there is no experimental data available for comparison with calculations. The uncertainty for graphite type fuel (Refer to Section 2.4.4.2.1.3) has been identified. The substantial difference in material composition of the two cores does prevent, however, the direct use of the graphite fuel uncertainty for composite fuel. The major differences in core composition using these fuels are summarized below:

<u>Constituent</u>	<u>R-1 Mass, kg</u>	
	<u>Composite Fuel⁽⁸⁾</u>	<u>Graphite Fuel⁽⁹⁾</u>
Uranium	333	275
Carbon	869	1181
Niobium	0	80
Zirconium	1189	149

Of these, the niobium and zirconium content is of particular importance. In the case of niobium, comparisons with measurement have shown that the predicted worth was in substantial error.⁽¹³⁾ This was also found to be the case for zirconium.⁽²⁵⁾

(U) Experiments utilizing composite fuel in PAX-G4 are scheduled for performance. Appropriate analysis will be conducted to determine the uncertainty in composite fuel analysis.

NERVA Engine, Allowable Manufacturing Variations

(U) The reactivity effect of the allowable manufacturing variations is the same as that for the R-1 reactor.

NERVA Engine, Uncertainty in Nuclear Calculations

(U) As in the case of the R-1 Reactor, the reactivity effect of the uncertainty in the

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engine/reactor nuclear calculations and NDT inventory are undeveloped at this time. The uncertainty for graphite type fuel has been identified to be $-0.362 \pm 0.253\%$ and is considered to be typical of what might be expected in the composite fuel situation.

2.4.4.2.1.2 Composite Core Operating Reactivity Uncertainty

(U) The uncertainty in operating reactivity of both the R-1 reactor and the NERVA Engine with composite fuel is judged to be the same as that for these reactors with graphite fuel. The total uncertainty is developed to be $\pm 80\%$ for the R-1 reactor and $\pm 35\%$ for the NERVA Engine. For a detailed development of these values, refer to Section 2.4.4.2.1.4.

2.4.4.2.1.3 Graphite Core Ambient Reactivity Uncertainty

(U) The uncertainty in the ambient reactivity is derived in this section from consideration of 1) the allowed manufacturing variations and 2) the uncertainty in the WISDM⁽¹⁾ absolute reactivity prediction. Note that the uncertainty in the true reactor component masses is implicit in this latter area.

(U) The R-1 reactor and the Engine system are treated separately.

R-1 Reactor, Allowable Manufacturing Variations

(CRD) Table 2-28 summarizes the allowable manufacturing variations as defined by the fuel element drawings for the constituents in the fueled and unfueled support elements in the graphite core. The values have been presented on a percentage and on a gram per element basis. Since the true masses of the constituents in each fueled and unfueled element will be known (to within the NDT accuracy) and will fall within the allowed variation band, the quoted core variation is taken to be the product of the square root of number of elements and the variation per element. This value represents a statistically nominal condition and would only apply to a situation where there was no bias in manufacturing. The reactivity effect associated with mass variations of this magnitude is as follows:

Constituent	Mass Variation (+), kg	Reactivity Coefficient, ¢/kg	Reactivity Effect, (+), ¢
Uranium	0.16	+ 11.5 ⁽²⁴⁾	1.8
Carbon	0.62	+ 4.3 ⁽²⁴⁾	2.7
Zirconium	0.76	+ 0.05 ⁽¹⁵⁾	---
Niobium	0.39	- 4.0 ⁽¹³⁾	1.6
Statistical Sum $\pm 3.6¢$			

(U) Thus, if the variations in constituents of the fueled and unfueled elements are distributed randomly in the allowable manufacturing variation band, the variability in the total core inventory is rather small with a reactivity worth of $\pm 3.6¢$.*

(U) However, this situation was not the experience of the NRX reactor test series. Table 2-29 summarizes the difference in mass and reactivity between the specified and manufactured fuel elements used in the six reactors in the NRX series. The reactivity differences range from -108.7¢ to + 90.1¢ with an average of + 14¢. (The normality of the data is examined in Figure 2-15). While the numerical average is small, the important factor is the variability in the individual values. This is characterized by the standard deviation in the individual values of $\pm 84.8¢$ developed in Table 2-30. Also appearing in this table is the calculation of the 90 and 95% confidence envelope for the reactivity differences indicated by the sample. The values developed are plotted in Figure 2-16 versus probability. As an example of the use of this data; if it is desired to include 80 percent of the reactors which may be built to a 95% confidence level, then preparations should be made to use reactors which may deviate by as much as + 2.6\$, - 2.4\$ from nominal. The maximum value for the deviation from nominal can be computed from the data in Table 2-28 by not treating the allowable deviations statistically. Assuming this limiting situation, a value of $\pm 2.5\$$ has been computed for graphite type fuel (note that this is 2.0\$ for composite fuel). Thus, Figure 2-16 indicates that the maximum allowable situation corresponds to an (80, 95) percent condition.

*See footnote in corresponding composite fuel section.

R-1 Reactor, Uncertainty in Nuclear Calculations

In order to verify that a particular reactor design does nominally provide the required reactivity, Nuclear Design performs extensive neutronic calculations. Although the procedure followed in these calculations is rather detailed, experience has shown that there is a bias between the calculated reactivity values and that measured for the reactor simulated by the calculation. This bias is known to result from two major sources: 1) errors in nuclear calculations for the nominal reactor material inventory and 2) the extent to which this nominal inventory is unknown. This second area is related to errors in the NDT material assay and is implicit in an evaluation of the uncertainty in nuclear calculations. In the discussion that follows, the magnitude of the overall bias is identified from a WISDM analysis of four "graphite" fueled (of somewhat different composition than the R-1 graphite fuel) reactors in the PAX-G reactor series. These were PAX-G0, G1, G2 and G3 (refer to Section 2.4.4.1 for a description of these reactors.)

The WISDM procedure (see section 2.4.2.1 and 2.4.2.2 for a description of the overall framework of WANL analytical work) followed in the analysis of each of these four reactors is the same as that described in Section 2.4.1.3. The results of the analysis are summarized in Table 2-31 which lists the computed eigenvalues and reactivities for the 60 degree test sector and the 300 degree non-sector portion of the reactors. A correction to the computed reactivity to account for effects not included in the direct EXTERMINATOR-2 calculation is developed in Table 2-32. The correction, found to be 78¢ for PAX-G0, G1 and G2, primarily accounts for the pressure vessel (56¢) which is not simulated in nuclear design calculations. The 180 degree pressure vessel mock-up in PAX-G3 is estimated to be worth only 32¢ so that the PAX-G3 correction is 54¢. These corrections are added to the WISDM calculated reactivity in Table 2-24 where the final predicted absolute reactivity for the four reactors is compared to the measured reactivity. Note that, with the exception of PAX-G0, the test sector is the "Reactor" which is treated in each case. That is, the analysis assumes that the reactor is made up of six 60° test sectors. The "Measured Reactivity"

values in the table have been derived from measured reactivity changes between PAX-G0 and the other reactors in the PAX-G series.

The last column in Table 2-24 gives the error in the best estimate of absolute reactivity for the four reactors in this study. This is the information that will be used in the following discussion to develop 1) an average value for the analytical bias and 2) the safety margin that must be allowed in order to assure that the reactivity of the reactor design being evaluated will satisfy the design requirement without unnecessary penalty to the NSS design.

To allow use of a common statistical method for the calculation of the average analytical bias and its variability, a first step is the determination of the underlying population frequency distribution. Toward this end, Figure 2-17 has been prepared. This figure shows the four data points plotted on a probability scale of Cumulative Frequency⁽²⁶⁾. When data satisfying a "normal" distribution is plotted in such a fashion, it will fall on a straight line. The extent to which this is not the case will be an indicator of the sample's non-normality. For the four data points here, it appears that the non-normality is not great enough to be detected by the available sample. This allows, then, the statistical treatment of the data to proceed using the equations from "normal" distribution theory to calculate the parameters that characterize the population from which the four data points have been taken. These calculations are summarized in Table 2-33. The estimate of the population mean (M) and standard deviation of a single measurement (S) is found to be $+1.56\%$ and $\pm 0.76\%$, respectively. Note that the mean $+1.56\%$ agrees with previous WISDM evaluation analysis for the PAX-E1 reactor where a reactivity error of $+1.5\%$ was observed⁽¹⁾.

The lower portion of Table 2-33 shows the manner in which the parameters M and S are used to develop an uncertainty for calculated reactivity consistent with the desired probability and confidence levels. The reactivity necessary to assure that the reactor design being evaluated will satisfy the criticality requirement is shown in the last two columns in

this lower portion of the table. It has been calculated as $(M \pm KS)$ where K is a factor to account for 1) the size of the sample used to estimate the population parameters, 2) the desired confidence level and 3) the fraction of the population that is to be included. These values are plotted in Figure 2-18.

NERVA Engine, Allowable Manufacturing Variations

The reactivity effect of the allowable manufacturing variations are the same as that for the R-1 reactor.

NERVA Engine, Uncertainty in Nuclear Calculations

As engine/reactor tests proceed on duplicate systems, the ambient reactivity uncertainty of these reactors will be reduced from that of a graphite fueled R-1 situation. These uncertainties are known to result from two major sources: 1) errors in nuclear calculations for the nominal reactor material inventory and 2) the extent to which this nominal inventory is unknown. This second area is related to errors in the NDT material assay and is implicit in an evaluation of the uncertainty in nuclear calculations. To define typical uncertainties for "duplicate" graphite fueled reactors, the experience of the NRX testing program which had no in-place shim adjustment has been examined in detail. Table 2-34 summarizes a comparison between prediction and measurement for the NRX-A2, A3, A4, A5, A6 and XE-Prime reactors. These reactors are all structurally very similar to one another with the exception of NRX-A6 and it could be logically argued that, for the purpose of developing characteristics of "duplicate" reactors, NRX-A6 should be excluded. It has, however, been included since its nuclear characteristics do not differ significantly from the others and, therefore, prediction of these characteristics logically represents another measure of the analytical bias in nuclear calculations. The last column in Table 2-34 gives the value of this bias for each of the six reactors. The values are seen to range from + 6.9% to -64.8% and will be used in the following discussion to develop 1) an average value for the analytical bias and 2) the safety margin that will be necessary to account for the variability in this analytical bias.

To allow use of a common statistical method for the calculation of the average analytical bias and its variability, a first step is the determination of the underlying population frequency distribution. Toward this end, Figure 2-19 has been prepared. This figure shows the six data points plotted on a probability scale of Cumulative Frequency⁽²⁶⁾. When data satisfying a "normal" distribution is plotted in such a fashion, it will fall on a straight line. The extent to which this is not the case will be an indicator of the non-normality of the sample. For the six data points here, it appears that the non-normality is not great enough to be detected by the available sample. This allows, then, the statistical treatment of the data to proceed using the equations from "normal" distribution theory to calculate the parameters that characterize the population from which the six data points have been taken. These calculations are summarized in Table 2-35. The estimate of the population mean (M) and standard deviation of a single measurement (S) is found to be $-0.362\$$ and $\pm 0.253\$$, respectively.

The lower portion of Table 2-35 shows the manner in which the parameters M and S are used to develop an uncertainty for calculated reactivity consistent with the desired probability and confidence levels. The uncertainty in calculated reactivity is shown in the last two columns in this lower portion of the table. It has been calculated as $(M \pm KS)$ where K is a factor to account for 1) the size of the sample used to estimate the population parameters, 2) the desired confidence level and 3) the fraction of the population that is to be included. These values are plotted in Figure 2-20.

2.4.4.2.1.4 Graphite Core Operating Reactivity Uncertainty

In this section, the uncertainty in the reactivity at design operation is presented for the R-1 reactor and for the NERVA Engine system.

R-1 Reactor

The uncertainty in the reactivity at design operation ("hot reactivity") of the R-1 reactor is developed from the square root of the sum of the squares of the uncertainties in the components from which this hot reactivity is calculated.

Thus,

$$S(H) = (S(A)^2 + S(V)^2 + S(N)^2)^{1/2}$$

where $S(H)$ is the uncertainty in the hot reactivity,

$S(A)$ is the uncertainty in the ambient reactivity,

$S(V)$ is the uncertainty in the cold-to-hot reactivity swing resulting from variations in thermal conditions and material inventory, and

$S(N)$ is the uncertainty in the nuclear calculation of the cold-to-hot reactivity swing.

In the discussion that follows, a value for each of the component factors is given on the basis of engineering judgement:

a) The uncertainty in the ambient reactivity with in-place shimming (ISA), $S(A)$, is taken to $\pm 15\%$.

b) The effect of variations in thermal conditions and material inventory, $S(V)$, has been included in a single factor so that the NERVA Engine analysis would be applicable. To adapt the value derived in that section, $\pm 30\%$, to an R type reactor, an additional effect must be incorporated to account for the uncertainty in the reactivity worth of the support stem coolant. This latter uncertainty has been judged to be $\pm 40\%$. Thus, $S(V)^2 = 30^2 + 40^2$ or $S(V) = \pm 50\%$.

c) The last factor $S(N)$ is developed below for graphite type fuel. The composite fuel effects are also shown to indicate their similarity.

Reactivity Feedback Effect	Uncertainty, %	Calculated Feedback, \$ Composite/Graphite	Uncertainty, \$
Core Hydrogen	10	+2.70/+2.75	0.28
Core Temperature	25	-2.10/-2.00	0.50
Peripheral	50	+0.48/0.48	0.05

The total uncertainty in the calculated cold-to-hot swing $S(N)$ is obtained by statistical addition of these values to be $\pm 58\%$.

Combining the three factors statistically as shown in the first equation in this section, the uncertainty in the hot reactivity of the R-1 reactor, $S(H)$, is found to be $(15^2 + 50^2 + 58^2)^{1/2} \cong \pm 80\%$.

NERVA Engine

In this section, the uncertainty in the reactivity at design operation ("hot reactivity") is developed from a statistical analysis of NRX test data. An uncertainty derived in this manner will account for both typical variations in 1) thermal conditions and 2) material inventory. Note that uncertainties resulting from support stem coolant have not been isolated as a special effect as was the case for the R-1 reactor discussed above. It is judged that this effect will not be a special factor in engine performance.

A summary of cold-to-hot reactivity swing observations in the NRX test series is presented in Table 2-36. Only those conditions corresponding to design full power have been included. The degree to which these deviate from the "design point" (last entry in the table) can be seen from the chamber conditions appearing in the rightmost three columns of the table. In Table 2-37, the observed cold-to-hot swing values from Table 2-36 are corrected by the technique shown for these deviations from "design point" conditions. The values shown in the last column in the Table 2-37 are examined for "normality" in Figure 2-21. As in the case of other statistical studies in this chapter, the non-normality is not great enough to be detected by the available sample. Using then, the equations from "normal" distribution theory, the parameters have been calculated that characterize the population from which the nine data points have been taken. As shown in Table 2-37, these are 1) a mean value of -78.9% and 2) a standard deviation (S) in the individual value of $\pm 4.7\%$.

The uncertainty in the cold-to-hot reactivity swing is developed statistically in Table 2-38. The uncertainty analysis has been based only on the $\pm 4.7\%$ variability experienced in the NRX reactor test series and is not specifically concerned with the mean cold-to-hot swing. The values from Table 2-38 are plotted versus probability in Figure 2-22 where it can be seen that the $\pm 30\%$ value, which has been utilized in previous engine analysis, represents a condition which includes at least 99.9 percent of the population to a 95 percent confidence level.

The uncertainty in the hot reactivity is then calculated from 1) the $\pm 30\%$ cold-to-hot uncertainty and 2) the cold uncertainty itself, which has been judged to be $\pm 15\%$. Summing these statistically, the uncertainty in the engine reactivity at design operation is seen to be approximately $\pm 35\%$.

2.4.4.2.1.5 Uncertainty Compensation Techniques

(U) The methods that have been devised to compensate for the reactivity uncertainties developed above are discussed in this section. These techniques will allow reactor/engine operation at design conditions and will eliminate undesirable deviations from design operation that would result from:

- 1) allowable manufacturing variations, and
- 2) errors in nuclear design predictions (which include errors in NDT assay).

The discussion is divided into techniques expected to be available for the NERVA engine and those presently planned to be available for the R-1 reactor. The methods that will be used are the same for composite and graphite fueled systems, so that a single, general treatment is appropriate in this regard.

NERVA Engine

(U) This section presents a discussion of the reactivity compensation techniques expected to be available for the NERVA engine and quotes the magnitude of the reactivity effect to be compensated as developed in the preceding uncertainty analysis. The methods discussed are:

1) fuel element selection, 2) in-place shim adjustment, and 3) support stem assembly flow variation. In the concluding section, the logic for engine reactor assembly is presented.

a) Fuel Element Selection

(U) Fuel element selection and in-core placement, as it is defined below, will be used to compensate for reactivity bias that is a result of non-random occurrence of allowable manufacturing variations. Random variations are shown above to have an insignificant $\pm 4\%$ effect. NRX experience has shown, however, that a non-random situation is more likely. A statistical analysis of the NRX data indicated that a 14% bias could be expected on the average with a variability of $\pm 85\%$. Figure 2-16 presents this information statistically as a function of probability of occurrence and confidence level.

(CRD) Fuel element selection, as it is conceived for the engine situation, allows free choice of fuel elements of average uranium, carbon, niobium, and zirconium loading such that reactivity deficiencies that result from known manufacturing bias can be eliminated. Thus, a group of fueled and unfueled elements may be selected to construct an engine's reactor core such

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that their average material inventory will be adequate for design operation and will compensate for known manufacturing bias. In addition to correct reactivity, these elements will also be placed in the core configuration so that variations in outlet propellant temperature are minimized.

b) In-Place Shim Adjustment

A technique whereby neutron poison material can be added or removed from the reactor core has been devised in order to allow a final adjustment of the ambient reactivity. This procedure is termed In-Place Shim Adjustment (ISA). The procedure will be used to compensate for uncertainties in the ambient reactivity from two sources as has been discussed previously in this section. One uncertainty originates from the nuclear analysis by which the nominal fuel loadings are specified. The other uncertainty is a result of the inherent inaccuracy of the nondestructive tests (NDT) which are used to assay the content of the fuel elements employed in the core loading. The NSS specification (CP-677555A, Paragraph 3.3.1.3.3) states that the total shim content of the flight reactor shall not exceed $-\$3.00$.

The adequacy of the ISA techniques for the NERVA engine can be evaluated from the uncertainty analysis presented above. The experience in the prediction of NRX reactor criticality is judged to be typical of what might be expected in the engine system. A probabilistic analysis of the NRX data indicated an average error of -36% with a variability of $\pm 25\%$. Figure 2-20 presents this information statistically as a function of probability of occurrence and confidence level. It is seen that (for example) if it is desired to cover 80 percent of the population to a 95 percent confidence level, an uncertainty of -110% , $+40\%$ must be prepared for. Note that the NRX experience does include NDT effects as discussed earlier.

As can be seen, the planned engine ISA capability is adequate for such a situation. The discussion that follows presents a summary of material appearing in Reference 27, a report giving test data and corresponding analysis for the shim materials.

Two alternate procedures are being considered for the In-Place Shim Adjustment method of reactivity adjustment. Both of these would provide an initial shim content of $-\$1.50$ in the core using TaC coated tubes. In the first method, shim tubes are placed in

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one-half of all support stems in the initial construction. This procedure is termed the (1/2) (1/2) scheme and provides a shim swing of $\pm \$1.50$. In the second method, shim tubes are placed in only one-third of all support stems in the initial build. This procedure is termed the (1/3) (2/3) scheme and provides a shim swing of $+\$1.50$ to a maximum of $-\$3.00$. The latter scheme provides greater latitude for shimming of ground test reactors.

(CRD) The mass of TaC coating and the corresponding coating thickness were determined for each shim procedure for the two candidate fuels based on measurements performed at WANEF⁽²⁷⁾. The required masses (and nominal thickness) of the TaC coating per tube in the composite core are 31.4 gms (3.2 mils) for the (1/2)(1/2) scheme and 53.9 gms (5.8 mils) for the (1/3)(2/3) procedure. For the graphite core, the TaC requirements are 27.6 gms (2.8 mils) and 46.7 gms (5.5 mils) for the respective shim procedures.

(U) The supporting ISA experiments also provided data on the interaction of shim material with support stem hydrogen and propellant hydrogen. With the $-\$1.50$ worth of TaC in the initial build of either core, the average reactivity worth of support stem hydrogen is reduced by approximately 9 percent relative to the hydrogen worth in an unshimmed core. For the same shim content, the propellant worth is reduced by approximately 3 percent. These reductions in hydrogen worth and consequent change of the cold-to-hot reactivity swing will be factored into the final number of shim tubes used to adjust ambient reactivity, that is, the ambient reactivity will be adjusted for this effect so that the operating reactivity at start-of-life will be as required.

c) Support Stem Assembly Flow Variation

(U) Variation in the flow of support stem assembly coolant has been adopted as the technique that will be used to compensate for errors in the reactivity at design operation. The uncertainty developed above for this condition is $\pm 35\%$. The dominant component of this uncertainty, $\pm 30\%$, is the result of a statistical analysis of NRX cold-to-hot reactivity swing data. The probabilistic analysis of this information identified the variability to be $\pm 4.7\%$ and indicated as shown in Figure 2-22 that an uncertainty of $\pm 30\%$ for the cold-to-hot reactivity swing was not likely to be exceeded. A 35% bias in the minimum start-of-life stem flow rate has been designated to account for this uncertainty.

~~CONFIDENTIAL~~

d) Summary

The reactivity compensation techniques presently planned for the NERVA engine are seen to be adequate and will allow the reactivity of the system to be adjusted so that full power can be attained at design conditions. Figure 2-23 illustrates schematically the logic of the reactivity compensation operation during reactor assembly. This relates, of course, only to the first two reactivity compensation techniques: fuel element selection and in-place shim adjustment. Support stem flow variation does not appear on the diagram in that it occurs during powered operation.

R-1 Reactor

This section presents a discussion of the reactivity compensation techniques expected to be available for the R-1 reactor and quotes the magnitude of the required compensation as developed in the preceding uncertainty analysis. Methods discussed are in-place shim adjustment for ambient uncertainties and SSA flow variation and drum motion for uncertainties at design conditions. The adequacy of the various methods is evaluated in each case.

a) Fuel Element Selection

Since fuel element selection will not be available for the R-1 reactor, the large reactivity variations expected (see Figure 2-16) from allowed variations in fuel manufacturing must be compensated in some other manner. It is expected that element selection will be used on later reactors as additional fuel elements become available.

b) In-Place Shim Adjustment (ISA)

The ISA technique has been presented in the NERVA engine discussion. In this section, only the magnitude of reactivity adjustment required of the ISA system for the R-1 reactor is treated. The situation is represented by the reactivity error envelopes shown in Figure 2-24. The curves shown in this graph come from two uncertainty analyses discussed earlier. The outer limit envelope is derived from $\pm$ KS values shown in Table 2-33 for the variability in the WISDM absolute reactivity prediction accuracy. This is considered the error variability limit of an uncorrected WISDM analysis. The inner limit envelope has been

derived from the $\pm$ KS values shown in Table 2-35 for the NRX absolute reactivity prediction experience. This is judged to be the minimum error variability envelope. In both cases it has been assumed that the bias has been taken into account and essentially set to zero so that only the variability remains. Note that since the bands shown are based on comparisons between calculations and experiments the effect of NDT inaccuracies is implicitly included.

Thus, the shaded area between the minimum and maximum envelopes represents the range of possible reactivity variations of the R-1 reactor* excluding the influence of subsequent WANEF experimentation. The effect of these experimental results will be to move the R-1 situation from position on the outer limits of the envelope to some point in the shaded area. The limit of this improvement is taken to be the NRX experience curves and corresponds to NCX with a full core of R-1 fuel. Clearly, a $\pm$ \$1.50 ISA capacity does not represent a situation that is likely to be shimmable. For example, if it is desired to include 80 percent of the possible variations in prediction error, then preparations to compensate for a maximum reactivity variation of $\pm$ \$3.20 are necessary. As introduced above in the NERVA engine discussion, use of the proposed (1/3)(2/3) ISA procedure (+\$1.50, -\$3.00) will provide additional one sided capacity relative to the (1/2) (1/2) scheme providing $\pm$ \$1.50 noted above.

c) SSA Flow Variation and Control Drum Motion

Capability to compensate for a total error in operating reactivity of $\pm 80\%$ has been planned for the R-1 reactor as indicated by the uncertainty analysis of Section 2.4.4.2.1.4. Of this $\pm 80\%$, $\pm 35\%$ can be compensated by SSA flow variation as discussed for the NERVA engine. The control drum span has been designed to compensate for the additional $\pm 45\%$. Thus, control drum motion will provide the remaining reactivity. This is factored into the analysis for the composite and graphite fueled cores in Section 2.5.1.1.4.1.

d) Summary

The R-1 reactivity compensation technique planned for operating reactivity uncertainties is seen to be adequate while that planned for ambient reactivity uncertainties is not likely to be sufficient. Several alternative compensation techniques for ambient uncertainties are as follows:

*Note that this refers to the reactor prior to reactivity adjustments by the ISA procedure.

- 1) Provide additional ISA capacity (by the (1/3) (2/3) scheme, for example).
- 2) Manufacture extra fuel elements for either a) element selection which accounts for manufacturing variations and/or b) reactivity prediction errors.
- 3) Limit initial fuel specification and manufacture to some large (50 percent, for example) core fraction so that an early NCX can be performed. This would allow a revised fuel specification if necessary for the elements yet to be manufactured.

These alternative schemes imply problems of a budgetary and/or programmatic nature and require further evaluation.

2.4.4.2.2 Power Distribution Uncertainty

In the R-1 power distribution analysis, the ratio of measured to calculated fission density values from the PAX-G2 reactor has been used to correct the calculated R-1 fission density. Section 2.4.4.1.2 presents the PAX-G2 results in detail.

2.4.4.2.3 Control Drum Span Uncertainty

The uncertainty in the control drum span is developed in Reflector Assembly Section 8.4.4.

2.4.4.2.4 Cold-To-Hot Swing Uncertainty

The uncertainty in the cold-to-hot swing is developed in the discussion for absolute reactivity as a factor in the operating reactivity uncertainty. In Section 2.4.4.2.1.4, the cold-to-hot swing uncertainty analysis is presented for the R-1 reactor and the NERVA engine and is combined statistically with the ambient reactivity uncertainties to provide the uncertainty in reactivity at design operation.

2.5 RESULTS, ANALYSIS, AND INTERPRETATION

This section presents the results of the nuclear analysis of the overall NSS and interpretation of the significance of these results for both a composite and graphite fueled core.

2.5.1 Composite Core

This section presents the detailed nuclear results for the NSS containing a composite fueled core.

2.5.1.1 Steady State

2.5.1.1.1 NSS Absolute Reactivity-Fuel Loading Distribution

The predicted NSS absolute reactivity and corresponding fuel loading distribution was developed using the Method of Analysis described in Section 2.4. Utilizing the composite core calculational model in conjunction with the 16-group regionwise spectrally weighted cross sections (See Section 2.4.1) R-θ EXTERMINATOR problems were set up for the 50°, 0°, and 180° drum bank positions. These problems incorporated a negative shim worth of \$1.5 distributed uniformly throughout the core. The calculation resulted in an absolute reactivity value of -\$2.63 for the 50° drum position. The prescribed initial cold critical drum position is approximately 80° degrees. Utilizing the results of the three composite EXTERMINATOR problems (0°, 50°, 180° drum positions) to determine the change in reactivity as a function of drum angle, the calculated value of reactivity at the 80° cold critical position is -\$0.88. Correcting this EXTERMINATOR reactivity for a pressure vessel, instrumentation, external structure and nozzle results in a predicted reactivity at the 80° cold critical position of -\$0.03. The maximum uranium loading utilized in this composite analysis is 765 mg/cc, which was developed on the assumption that the fueled element carbon density is 1.3 gm/cc for all uranium loadings.

In this analysis, it has been assumed that the R-1 EXTERMINATOR problems correctly calculate the absolute reactivity. The calculated/experimental reactivity

correction developed for the graphite fuel in Section 2.4.4.1.1 was not utilized for the composite fuel. Therefore, based on the above EXTERMINATOR results, a composite core maximum fuel loading of 765 mg/cc is correct to achieve the desired 80° cold critical drum angle.

In support of the development of the absolute reactivity prediction for an R-1 composite core, the following PAX/R-1 experimental program is planned. In the PAX-G4 critical facility, an approximate 60 degree fuel sector of the graphite core will be replaced by LASL composite fuel. Reactivity analysis will be performed on this composite sector (PAX-G4) to obtain a degree of understanding on the composite absolute reactivity level. In addition, spatial material reactivity coefficient worths will be measured and used with the R-1 composite fuel composition to perform a mass/reactivity balance.

2.5.1.1.2 Radial Power Distribution

The preliminary NERVA radial power distribution was also developed using the Method of Analysis described in Section 2.4. The R-θ EXTERMINATOR problem at the 50° drum bank position was used for this analysis. As noted previously, this problem incorporated a negative shim worth of \$1.5 uniformly distributed over the entire core. The orifice drum angle was initially selected at 50°; the final TNT angle was 47°. The output of this EXTERMINATOR results in a calculated pointwise radial and azimuthal fission flux distribution for the preliminary R-1 composite loadings. Table 2-39 summarizes the multizoned fuel distribution used in this analysis. As noted, this distribution is composed of thirteen separate fuel element loadings. Figure 2-25 presents the elemental loading distribution which results in the fifteen radial core region breakup. As noted, regions 11 and 14 have the same loadings as do regions 6 and 15. Figure 2-26 shows the desired radial fission distribution at the 50° orifice drum bank angle. This shape is restricted by the requirement that the maximum power factor must not exceed 1.1.

As shown in the analytical flow charts of Section 2.4.1, the detailed radial power factor calculations are performed using the PFENCER series of codes in conjunction with the EXTERMINATOR output fission fluxes and the PAX/R-1 experimental power measurements and normalization coefficients. A comparison of the PAX/R-1 power measurements and predictions have shown that the calculation underpredicts the measured values by as much as 20 to 30% at the core periphery. The ratio of the PAX/R-1 measured to calculated fissions/gm for each fuel channel normalized to unity at the core centerline is defined as the experimental correction factor, NC_i . Figure 2-27 shows the distribution of the PAX-G2 NC_i 's which were used in the development of the preliminary loading distribution. Consequently, the power distribution calculated by EXTERMINATOR at the correct drum position will result in a significant error in the peripheral power when experimentally measured. Therefore, the desired fission distribution (Figure 2-26) was divided by the PAX-G2 NC_i distribution to obtain the initial aim EXTERMINATOR power, see Figure 2-28. Figure 2-29 shows a comparison of the aim EXTERMINATOR distribution (Figure 2-28) and the output of the 50⁰ EXTERMINATOR problem. As noted, there is good agreement between the power distribution based on the preliminary loading distribution utilized in the composite model and the aim EXTERMINATOR distribution.

Using the 16 group flux output tape from this EXTERMINATOR as input, a PFENCER 1 calculation was performed to obtain the calculated fission/gm distribution for each of the 35,682 fuel element channels in the core. Using the NC_i 's obtained from the PAX-G2 PFENCER 2 calculation, a PFENCER 3 calculation was performed which resulted in the experimentally corrected R-1 power distribution. Table 2-25 shows the PAX-G2 experimental and predicted channel 10 fission/gm data from which the NC_i 's were developed. In previous NRX reactors, the radial fission distribution change in going from the ambient conditions to the hot start of life condition (with hydrogen) has been on the

order of a few percent. For this power distribution analysis, the cold-to-hot power perturbation has been assumed to be zero. The following section develops the composite core cold-to-hot power change. Because those calculations were performed simultaneously with the analysis described here, the effect was not included in the operating power distribution. However, as NRX experience demonstrated, the power perturbations are calculated to be small. Figure 2-30 shows the central element power perturbation which was input to the PFENCER-3 code. This local power perturbation was assumed on the basis of NRX experience as was the cold-to-hot power effect. The corresponding calculated local fuel element power perturbation is also developed below. Figure 2-31 shows the relative radial distribution of the photon and neutron kinetic heating also used in this analysis. Table 2-40 shows the channel power factors resulting from the PFENCER-3 calculation.

In developing this composite loading scheme, all of the experimental data was obtained from the PAX/R-1 reactor which has a graphite core. As discussed above, to aid in the composite fuel fission distribution determination, an approximate 60° sector of LASL composite fuel will be assembled in the PAX-G4 critical configuration. Radial fission distribution data will be obtained for this sector to update and substantiate the R-1 composite radial fission distribution predictions.

2.5.1.1.2.1 Cold - To - Hot Power Distribution Change

The radial fission power profile change for the composite core reactor has been calculated in going from ambient conditions to nominal rated full power at start of life. In the composite core reactor, cold critical is at a control drum bank position of 80° and nominal hot critical is at an angle of 47° . Reactivity changes and fission power profile perturbations due to drum rotation, seal, cylinder and reflector hydrogen flow, reflector and core temperature change, core expansion, cylinder contraction and core hydrogen flow were calculated as a function of control drum bank position.

Figure 2-32 presents the fractional fission power perturbation per dollar of reactivity as a function of radial position in the composite core reactor for outward control drum bank rotation. Figures 2-33 through 2-40 present respectively without and with a control drum bank representation these power coefficients for seal, cylinder and reflector hydrogen flow and for peripheral flow which is the combined effect of these three components. Figures 2-41 through 2-44 show respectively both without and with a control drum bank mockup the power coefficients for reflector temperature change and for the net effect of peripheral hydrogen flow and reflector temperature change. Figures 2-45 through 2-52 present without and with a control drum bank mockup the power coefficients for core temperature change, core expansion, cylinder contraction and core hydrogen flow, respectively.

Table 2-41 presents the fractional fission power perturbation per dollar of reactivity at the core edge, calculated both without and with a control drum bank representation for outward control drum rotation, seal, cylinder and reflector hydrogen flow, reflector and core temperature change, core expansion, cylinder contraction and core hydrogen flow. Table 2-41 also shows the power coefficients for each of these components at the core edge as extrapolated to nominal hot critical, 47 degrees, when the control drum bank has a worth to removal of \$5.59.

Based on these power coefficients extrapolated to nominal hot critical and the reactivity changes in each of these components determined in Section 2.5.1.1.5, a net increase of 2.5% was calculated in the fission power at the R-1 composite core edge in

[REDACTED]

going from ambient conditions to nominal rated full power at start of life. Figure 2-53 presents the total fractional fission power change as a function of radial position in the composite core in going from ambient conditions to nominal rated full power at start of life, using the power coefficients as calculated with a control drum bank representation and the reactivity changes as defined in Section 2.5.1.1.5.

2.5.1.1.2.2 Local Fuel Element Power Perturbations

(U) As noted above, local fission power gradient for fuel elements was assumed for preliminary design purposes. This distribution, shown in the first column of Table 2-42, was derived from previous measurements in dry, ambient fuel clusters.⁽¹⁸⁾

(U) Recent measurements of the local fission density were made in the PAX-G2 reactor for dry, ambient clusters with polyethylene in the support stems. The power gradients obtained from these measurements are labeled "dry" and "with poly" in the Table 2-42. The power gradients which correspond to the support stem minimum and maximum hydrogen flows were derived from the PAX-G2 measurements. These are labeled "Min. Flow" and "Max. Flow" in the table. Although the gradient with hydrogen is greater than in the dry measurements, it is seen that the gradient with minimum flow is in good agreement with the distribution assumed in the above power distribution analysis.

(CRD) The effect of shim tubes on the power gradient was not included in the PAX-G2 measurements. However, the perturbation due to 3-mil TaC shim tubes was calculated for the PAX-G2 fuel clusters. The calculated power gradients for minimum and maximum stem flow are shown in the right hand portion of Table 2-42. It is seen that the calculated power gradient is essentially the same with and without a shim tube. From these results, it is concluded that the measured power gradients (which did not include shim effects) can be employed for both shimmed and unshimmed clusters in future design calculations.

(CRD) However, the PAX-G2 calculations showed that the average power in an element adjacent to a central containing a 3-mil TaC tube (approximately 31 gms of TaC) is depressed by approximately one-half percent relative to the average power of an element in an unshimmed cluster. These calculations are qualitatively verified by comparison with measurements

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performed in the NRX reactor series.⁽²⁸⁾ Those measurements showed that the local power in an element adjacent to an unfueled element containing about 75 gms of TaC was depressed by one to two percent relative to the power in other elements.

2.5.1.1.2.3 Operating Power Distribution Change

The radial fission power profile change for the composite core reactor after a 10 hour reactor operating lifetime and for throttle mode operation has been calculated and that combination of SSA flow and drum reactivity compensation determined which results in a "flat" end of life profile. Reactivity effects and fission power profile perturbations due to drum rotation, SSA flow, corrosion, fuel burnup and fission product poisoning were calculated as a function of control drum bank position.

The calculations correspond to a total reactivity loss of \$1.11 which was found to occur at the end of 10 hours reactor operation at the nominal hot critical bank position, based on an assumed \$1.00 loss from corrosion and an 11 ¢ loss due to fuel burnup. The one-dimensional transport calculations performed with and without a control drum bank representation indicate that at the nominal hot critical angle of 47 degrees no significant reactivity loss is associated with fission product poisoning.

Figures 2-54 through 2-62 present the fractional fission power perturbation per dollar of reactivity as a function of radial position in the composite core reactor for drum rotation and without and with a control drum bank representation for SSA flow, corrosion, fuel burnup and fission product poisoning, respectively. Table 2-43 presents the fractional fission power perturbation per dollar of reactivity both without and with a control drum bank representation at the core edge. In this analysis the fractional fission power change per dollar of reactivity as calculated with the control drum bank representation was used, since as can be seen from Table 2-43 these power coefficient are not strongly dependent on drum bank position and as noted above at the nominal hot critical angle of 47° there is only an 11 ¢ reactivity loss due to fuel burnup and no significant reactivity loss associated with fission product poisoning.

Calculations show that the net effect of compensatory $\beta_{1.11}$ SSA flow reactivity change will result in no significant power change at the edge of the core and therefore, no compensatory drum rotation is required. Figure 2-63 presents these radial fission power profile changes due to corrosion, fuel burnup and SSA flow at the end of 10 hours reactor operation.

In the throttle mode of reactor operation the same net hydrogen worth is maintained in the core, in that the reactivity change due to the decreased propellant flow is compensated for by an increase in SSA flow to produce the same net core reactivity effect. Therefore, on a gross basis due to hydrogen and core temperature there will be no change in the radial fission power profile in going from the operating to the throttle condition. However, there is a change in the radial fission power profile due to the decreased peripheral hydrogen flow in the throttle mode of operation. TNT calculations indicate a reactivity decrease of about 15% due to the decreased seal, cylinder and reflector hydrogen flow and no significant reactivity change due to the increased reflector temperature, as determined from the reactivity feedback plots of Section 2.5.1.1.5. Based on these reactivity changes and the power coefficients for seal, cylinder and reflector hydrogen flow at the core edge at nominal hot critical as specified in Table 2-41, a net decrease of 7.6% was determined in the fission power at the core edge in going from the operating to throttle mode of operation. Table 2-44 summarizes this calculation. The required outward control drum motion to maintain a relatively flat power distribution is approximately $\beta_{0.75}$.

2.5.1.1.3 Axial Power Distribution

The axial power distribution results are presented and discussed in Chapters 6 and 7.

2.5.1.1.4 Total Control Drum Span

For completeness, the control drum span analysis is included in this chapter. Further information on the reflector design and calculations are included in Section 8.5. The reference R-1 control system consists of 18 drums with a control drum vane outer diameter of 4.0 inches. The control vane is made of $B^{10}Cu$ with a B^{10} surface density of 0.100 grams/cm².

The optimized NSS outer dimension of 54.0 inches is the result of the Radial Dimension Optimization Study No. 776. Based on the results of that study, the reference reflector design was developed.

Using the Method of Analysis presented in Section 2.4, the total control drum span was calculated for this reference design. The difference in reactivity between the two exterminator problems ($\rho_{180^\circ} - \rho_{0^\circ}$) corresponds to the R-1 calculated control drum span for the reference composite fuel and the reference reflector assembly. The results of these two EXTERMINATOR calculations indicate the calculated control drum span to be \$6.90.

In Section 8.4 the drum span experimental normalization factors are developed. These were generated utilizing the following reactors: NRX-A6, PAX-GO, PAX-G1, PAX-G2, and PAX-G3. The experimental normalization factor developed statistically from these individual values for an 80% probability and a 95% confidence level, to insure a \$4.5 drum span, is 0.782. Using this 0.782 coefficient and the EXTERMINATOR drum span calculation, the final total control drum span for the composite fuel is \$5.40 ($0.782 \times \6.90).

The specified requirement for preliminary design is that the drum span be at least \$4.5 based on an 80 percent probability and 95 percent confidence level. Based on this requirement, the drum span value of \$5.40*meets the specified value of a \$4.50 minimum.

2.5.1.1.4.1 Control Drum Utilization

This section develops the NERVA total control drum utilization for both the composite and graphite fuel element designs. The following individual items are included in its development: 1) corrosion compensation, 2) fuel burnup and fission product poisoning, 3) cold-to-hot reactivity feedback, 4) ambient and operating uncertainties, 5) safety shutdown requirements, 6) operational mode requirements, 7) space storage, 8) cold flow testing, and 9) an actuator malfunction compensation.

Utilizing the analytical models of Section 2.4.2 calculations were performed to determine the NERVA fission product poisoning and fuel burnup effects. The fission product calculations were performed for the "worst case" conditions; that is, after a 60-minute

*Derivation of this result does not include provision for design changes between successive reactors. NRX experience showed a net \$0.80 decrease in drum span from NRX-A2 to XE-PRIME.

burn and assuming a startup at 10.5 hours following shutdown. This particular restart condition corresponds to the peak xenon concentration. The fuel burnup analyses assumed a 10 hour total lifetime. The results of these calculations are summarized in Table 2-45. As noted, the total fission product reactivity effect is approximately equal to -12% , -13% in the composite and graphite fuel element designs respectively. A combination of these values and the fuel burnup reactivity losses results in the drum span requirement for this item (fission products and fuel burnup) of -23% for the composite fuel and -25% for the graphite fuel. In addition, a corrosion compensation of $\$1.0$ was used for both the graphite and composite fuel designs.

The cold-to-hot reactivity feedback effects which were included in the drum span utilization are consistent with the results of the TNT model used in the S-001 (Systems Analysis Summary Report) and the S-031 (Thermal and Fluid Flow Analysis Report) Data Items. Where current cold-to-hot analyses differs from these TNT calculations, the impact of these differences are noted. Values are included for both fuel element designs. For the composite fuel, the total cold-to-hot reactivity feedback reported in the TNT model equals $\$1.08$. It should be noted that $\$0.45$ of this value is due to the SSCV being partially open from the minimum position. However, there is a corresponding reduction in the control drum span in going from ambient to the hot operating conditions of 6%. To arrive at the desired hot orifice angle, the total required cold-to-hot drum movement, including the reduced drum worth, is equal to $\$1.15$, ($\$1.08 + \0.07). Thus, the value which must be included in the development of the total control drum requirements for the cold-to-hot reactivity feedback of the composite fuel design is $\$1.15$. A similar situation exists for the NERVA reactor utilizing the graphite fuel design. For this fuel, the total cold-to-hot reactivity feedback reported in the TNT model equals $\$0.75$. The 6% drum span reduction is again incurred from the cold critical drum bank angle to the hot orifice angle. Therefore, the value that should be included in the development of the total control drum requirements for the cold-to-hot reactivity feedback of the graphite fuel design is $\$0.80$. However, as noted in Section 2.5.2.1.6 current cold-to-hot analyses has indicated a reactivity feedback in the graphite fuel design of $\$1.35$ including a 10% drum span reduction.

An analysis of the ambient and operating uncertainties is presented in Section 2.4.4.2 of the NUCLEAR SUBSYSTEM. Values are developed for both the composite and graphite fuel designs. As noted in the development of the cold critical errors, the uncertainty in the absolute reactivity prediction of the NERVA reactor is dependent on the experimental program used to predict the core neutronics and the procedure used to shim the reactor. In this drum span requirement development, it is assumed that the engine experimental program will include an ISA (In-place Shim Adjustment) experiment utilizing shim tubes with a ± 1.5 shim capability. As demonstrated in the NSS Subsection 2.4.4.2, this procedure, in conjunction with an element selection scheme to minimize manufacturing variation, should result in a relatively small uncertainty in the engine cold critical of ± 0.15 . This same uncertainty has been assumed for the R-1 reactor. Since the R-1 errors will be considerably larger, and additional elements may not be available to account for the R-1 manufacturing variation, various schemes have been proposed to minimize the R-1 cold critical uncertainties. Statistically combining the cold critical uncertainty (± 0.15) and the cold-to-hot uncertainties developed in Section 2.4 results in the hot critical uncertainties of ± 0.35 for the engine and ± 0.80 for R-1.

The NERVA safety shutdown requirements are 1.5 for non-operating conditions and 0.25 during operation. These values apply to R-1 as well as the engine.

A 0.70 outward drum movement from the cold critical position is required for a dry startup. This 0.70 reactivity insertion is also sufficient for bootstrap operation included in the normal operating mode procedure (See Sections 2.5.1.2 and 2.5.2.2).

A value of 0.20 is added to the overall drum span to allow for the condition that the reactor may not be at room temperature, between the various operating runs. This 0.20 value assumes a 170°R temperature increase* above the ambient temperature. The nominal SSCV position at start of life has been selected 0.45 above the minimum. The major portion of this reactivity, 0.35, is utilized for compensating for uncertainties in the hot operating angle. The remaining 0.10 reactivity is designated as a controllability margin. This 0.10 value compensates for the ± 5 psi variability encountered during control on the chamber pressure.

*The effect of the difference between this value and that inferred in the S-001 Data Item is minor (less than 10%).

The neutronic reactivity variations due to environmental changes during space storage were estimated to determine their influence on the overall drum span. A range of space temperatures were obtained for various NERVA missions. These temperatures varied from 10.0°R (for NERVA near the planet Pluto) to slightly above the ambient (540°R). The extreme temperature variation relative to ambient occurs if a hydrogen leak develops and extends for over a long duration. It has been estimated that this condition could result in an average reactor temperature of approximately 50°R . Using the reactivity feedback coefficients developed in Section 2.5.1.1.5 of the NUCLEAR SUBSYSTEM for core contraction, core hydrogen, cylinder contraction and reflector temperature, it was estimated that the NERVA shutdown is reduced to approximately $\$0.70$ at the 50°R temperature. A value for the space soak shutdown requirement has not been specified. Consequently, it has been assumed that the $\$0.70$ shutdown reactivity prediction for NERVA is sufficient from safety considerations.

Cold flow testing is discussed in this section to present the limiting reactivity insertion permissible for the minimum drum control scheme. The cold critical drum position is $\$1.65$ from the 0° drum position, $\$1.5$ non-operating shutdown + $\$0.15$ cold critical uncertainty. The $\$0.15$ uncertainty could result in the actual cold critical position being only $\$1.50$ from 0° . It is assumed that the reactor will be critical during the cold flow testing such that the safety aspects of the liquid insertion can be monitored. As stated previously, the operating safety shutdown is $\$0.25$ for all planned operations. Therefore, the maximum amount of reactivity which can be inserted during the NERVA cold flow testing program is $\$1.25$ without violating the safety shutdown requirements.

The final item discussed in this section which has an input to the overall drum span utilization scheme is an actuator malfunction allowance of $\$0.25$. When a NERVA control drum actuator fails, the AMOD override device returns the control drum to its orifice angle. Thus, the positive reactivity insertion due to the individual drum movement from the orifice angle to the 180° drum position is lost for the remainder of the reactor's operating lifetime. The $\$0.25$ allowance in the drum bank movement is sufficient to override this control reduction.

Table 2-46 presents a summary tabulation of the total NERVA control utilization as discussed above. Figure 2-64 depicts graphically the stack up of these requirements for the composite fuel, with the resulting overall reactivity summation being the total drum span. Figure 2-65 presents the corresponding drum stack up for the graphite fuel design. Utilization of the current cold-to-hot reactivity feedback of +\$1.35 developed in Section 2.5.2.1.6, rather than the +\$0.80 value from TNT, results in a total required drum span utilization for the graphite fuel of \$4.60 rather than the \$4.20 value shown in Figure 2-65*.

However, additional analysis has indicated another possible increase may be necessary in the total drum span over those shown in Figures 2-64 and 2-65. In reviewing the operating profiles (Figure 2-83 and 2-132 of Sections 2.5.1.2 and 2.5.2.2) a problem may exist in the reactivity introduced by the support stems preceding the resumption of full power operation. In this preliminary analysis which assumed extremes in temperature transients, the control drums were required to roll into the \$0.25 operating shutdown limit position. Though this does not violate the operating safety requirements, a problem exists in that this programmed profile is for the nominal reactivity prediction. For the NERVA reactor, a possible \$0.45 hot critical uncertainty exists in the operating angle.

To allow for the possibility of the operating angle coming out on the low side (excess reactivity over the nominal prediction), it would be necessary to add an additional \$0.45 to the overall drum span to maintain the same drum reactivity insertion that was utilized in the TNT runs. An additional problem was noted in the shutdown procedure of the TNT runs described in Sections 2.5.1.2 and 2.5.2.2. The normal mode shutdown is initiated with Throttling and a Throttle Hold. The analyses described herein employed a 40 second Throttle Hold followed by a Temperature Retreat from 4250°R to 1800°R at a rate of -175°R/sec. Simultaneously with the initiation of Temperature Retreat, the drums are rolled in at the maximum rate and the SSCV is ramped to its minimum flow position. During this time, P_c and core flow are increased to provide sufficient cooling to achieve the temperature ramp. Consequently, the core propellant flow and feedback reactivity peak during the Temperature Retreat. The feedback reactivity increases very rapidly at this time. There is the possibility that the corresponding peak in NSS reactivity could exceed zero, prohibiting

*Final allocation of these values will be consistent with the requirements of Table 2-46 and any up-date in safety or operational requirements.

shutdown. This situation is seen to be quite possible in Figure 2-83 for the SOL but is less probable at EOL with the effect of an additional -1.00 . However, the uncertainties in the NSS may be such that the drum reactivity insertion from hot critical to the full-in drum position is less than that noted in the reference shutdown (-70%). Thus, it appears that the drum span allocation must be increased in order to insure a successful shutdown. Based on the transient analyses discussed in Section 2.5.1.2, at least 45% additional drum span may be required to account for NSS uncertainties. However, this requirement must be reevaluated when an optimum operating profile is defined.

2.5.1.1.5 Cold-To-Hot Reactivity Feedback Analysis

(U) This section develops the formulation of the cold-to-hot feedback effects expected during operation of the composite reactor. The analytical method employed in this analysis is described in Section 2.4.1.3. The first subsection is a separate treatment of the various terms contributing to the total feedback without regard to interaction effects. The second subsection discusses the corresponding control drum interaction effects indicated from the analysis. The third subsection considers interdependencies between the separate feedback effects. The next two subsections discuss the analyses of the reactivity effects of para/ortho hydrogen conversion and shim tube interaction respectively. Finally, these results are factored into the formulation of a reactivity feedback model. This feedback model is compared with that now incorporated in the thermal TNT program.

Separable Feedback Effects

- (U) Each of the following effects represent perturbations to an ambient reactor system without flow and with a cold critical control drum bank angle of 80° .
- (U) 1. Core hydrogen worth - Figure 2-66 shows the reactivity feedback variation with uniform core hydrogen mass content. At nominal rated full power start-of-life, the "effective" stored hydrogen mass content is 1.13 pounds which corresponds to a \$2.65 reactivity feedback as noted in the figure. An effective mass content is required due to the axial variations in propellant and SSA hydrogen densities and hydrogen worth.
- (CRD) 2. Core temperature Worth - Figure 2-67 shows the reactivity feedback variation with core temperature excluding core expansion effects. At nominal rated full power start-of-life, the average core temperature is 3173°R which corresponds to a reactivity feedback of $-\$0.34$ as noted

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from the figure. These results do not treat differences in fueled and support element temperatures. It should be pointed out that sufficient cause exists for questioning these calculated core temperature neutronic feedback results. In particular, the calculations appear to significantly underestimate this feedback. Further discussion of this seeming discrepancy is included below.

Core temperature neutronic feedback in the composite R-1 reactor is effected by uranium and zirconium resonance doppler broadening and thermal neutron energy base shift with carbon temperature. Table 2-47 shows the relative contributions of these effects as calculated for the case of control drums removed. As is noted, zirconium resonance doppler broadening appears to be the major contributor to the core temperature neutronic feedback.

3. Filler Strip and Seal Hydrogen Worth - Figure 2-68 shows the reactivity feedback variation with combined hydrogen mass content in the filler strip and seal regions. Filler strip flow contributes less than 3% to the combined mass content. At nominal rated full power start-of-life the combined filler strip and seal region hydrogen mass content is 0.277 pounds which correspond to a \$0.152 reactivity feedback as noted in the figure. These results and those discussed below for cylinder and reflector hydrogen worths are based on a 37°R Para hydrogen scatter kernel. The effects of species conversion, Para $\rightarrow$ Ortho, and gas temperature are considered below.
4. Support Cylinder Hydrogen Worth - Figure 2-69 shows the reactivity feedback variation with hydrogen mass content in the aluminum support cylinder. At nominal rated full power start-of-life the support cylinder

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mass content is 0.416 pounds which corresponds to a \$0.154 reactivity feedback as noted from the figure.

5. Reflector Hydrogen Worth - Figure 2-70 shows the reactivity feedback variation with reflector hydrogen mass content. At nominal rated full power start-of-life the reflector hydrogen mass content is 1.237 pounds which corresponds to a \$0.027 reactivity feedback as noted from the figure.
6. Beryllium Temperature Worth - Figure 2-71 shows the reactivity feedback variation with beryllium temperature. At nominal rated full power start-of-life the average beryllium temperature is 245°R and has an essentially neutral effect on reactivity feedback as noted from the figure.
7. Support Cylinder Displacement Worth - Figure 2-72 shows the reactivity feedback variation with support cylinder radial displacement upon cooldown. These results assumed a corresponding reflector movement due to attachment via the aluminum lateral support spring capsules. Temperature contraction of these members is neglected as a cursory examination of the thermal coefficients of expansion indicate this would have a small secondary effect to that of the overall displacement. Cylinder displacement data from RAT's data runs were given earlier in the composite materials properties Section 2.4.2. Effective displacements are developed in Table 2-48 by reactivity sine weighting these data. At nominal rated full power start-of-life the effective cylinder radial displacement is -0.080 in. which corresponds to a \$0.148 reactivity feedback as noted in Figure 2-72. A maximum effective cylinder displacement of -0.082 in. occurs at 40 seconds into the SOL startup. At this point the feedback effect would be \$0.152 as is also noted in the figure.

8. Core Expansion Worth - Figure 2-73 shows the reactivity feedback variation with percent core expansion. Isotropic core thermal expansion properties are assumed. Core expansion displacement data from RAT's data runs were given earlier in the composite materials properties (Section 2.4.2). Effective displacements are developed in Table 2-49 by reactivity sine weighting these data. At nominal rated full power start-of-life the effective core radial displacement is 1.03% which corresponds to a -1.50 reactivity feedback as noted in Figure 2-73. At 40 seconds into the SOL startup the effective radial core displacement is 0.54% which corresponds to a -0.77 reactivity feedback as is also noted in the figure.

Control Drum Interactions

Each of the effects contributing to reactivity feedback examined in the previous section were analyzed as a function of control drum bank position. Variations in feedback with drum bank position are interpreted as control drum worth variations using the method discussed under the analytical procedures Section 2.4. of this report. The results are as follows:

1. Drum Worth Dependence on Core Hydrogen Mass - Figure 2-74 shows the control drum worth variation with uniform core hydrogen mass content. As is noted, control drum is reduced by 5.4% due to core hydrogen at nominal rated full power start-of-life.
2. Drum Worth Dependence on Core Temperature - Figure 2-75 shows the control drum worth variation with core temperature excluding core expansion effects. Control drum worth is reduced by 1.7% due to core temperature at nominal rated full power start-of-life. Again, as was discussed above, calculated core temperature neutronic effects are suspect and requires further elaboration as will be given below.

3. Drum Worth Dependence on Filler Strip and Seal Hydrogen Mass - The calculations have indicated less than a 1% drum worth variation over a range of mass content of twice that at nominal rated full power start-of-life.
4. Drum Worth Dependence on Support Cylinder Hydrogen Mass Content - Figure 2-76 shows the control drum worth variation with aluminum support cylinder hydrogen mass content. Control drum worth is reduced by 0.8% due to support cylinder mass at nominal rated full power start-of-life.
5. Drum Worth Dependence on Reflector Hydrogen Mass Content - The calculations have indicated less than a 1% drum worth variation over a range of mass content of twice that at nominal rated full power start-of-life.
6. Drum Worth Dependence on Beryllium Temperature - Figure 2-77 shows the control drum worth variation with beryllium temperature. Control drum worth is reduced by 2.4% due to beryllium temperature change at nominal rated full power start-of-life.
7. Drum Worth Dependence on Support Cylinder Displacement - Figure 2-78 shows the control drum worth variation with support cylinder radial displacement. Control drum worth is enhanced by 1.65% due to support cylinder inward displacement at nominal rated full power start-of-life.
8. Drum Worth Dependence on Core Expansion - Figure 2-79 shows the control drum worth variation with percent core expansion. Control drum worth is enhanced by 5.5% due to core expansion at nominal rated full power start-of-life.

Separability of Effects

The above effects were investigated for additivity and the results are tabulated in Tables 2-50 and 2-51. The tables compare certain combined reactivity feedbacks without control drums and drum worth effects with the corresponding results assuming additivity is valid. Parentheses are used to define manners of addition. For example, the case denoted as $H_c + (H_p + T_R)$ compares the combined calculated effect of core hydrogen, peripheral hydrogen and reflector temperature with the sum of the calculated core hydrogen effect and the calculated combined effect of peripheral hydrogen and reflector temperature. These results correspond to hydrogen masses and material temperatures which are in general typical of rated full power start-of-life conditions. An exception is reflector temperature which corresponds to an extreme variation of this parameter. In general, the results indicate that additivity is valid for both reactivity feedback and drum interaction effects. The largest discrepancy occurs in the addition of core hydrogen and core temperature effects which gives a result 13% greater than the calculated combined result.

The effects of core and cylinder displacements have not been considered in regard to this separability analysis. Further analyses will be required to ascertain whether these effects are properly additive.

Para/Orho Conversion

The above results consider peripheral hydrogen feedback effects based on 36.7°R para hydrogen scatter properties. The effects of species conversion (para → ortho) and gas temperature are shown in Figures 2-80 and 2-81. The curves assumed linearity between results calculated at the end points. In general, the results indicate that:

1. Both feedback and drum interaction effects are species independent at ambient (540°R) gas temperature. This is true regardless of beryllium temperature.

2. The ortho and para neutronic effects diverge with decreasing gas temperature. In general, para $\rightarrow$ ortho species conversion would reduce the positive peripheral hydrogen reactivity feedback and enhance drum worth.
3. The maximum reactivity feedback variation over the range of variables considered is only about 10%. The peripheral hydrogen properties used in the formulation of feedback in the earlier sections of this chapter correspond to the upper limit (maximum feedback of this range).
4. Drum worth interaction effects are more sensitive to these parameters and range from a 1.5% drum worth reduction with para hydrogen and cold (139.3°R) beryllium to a 5% drum worth enhancement with ortho hydrogen properties. The properties used in the formulation of drum worth variations in the earlier sections of this chapter correspond to the lower limit (a small drum worth reduction) of this range so that species conversion may increase the predicted operating drum worth.

Separate analyses have shown that core hydrogen worth is independent of both hydrogen species and gas temperature. Carbon scattering overshadows any differences in hydrogen scattering properties.

Shim Tube Interaction

The preceding development has suggested a \$2.65 feedback effect attributable to core hydrogen content at nominal rated full power start of life. To anticipate the direction of this discussion this result has been apportioned into propellant and support stem flow components of \$1.08 and \$1.57 respectively by an examination of the TNT thermal program results.

Recent analysis in conjunction with PAX/R-1 measurements have indicated that core hydrogen reactivity effects can be significantly influenced by core shim poison content. This

relationship is investigated in detail in the test report.⁽²⁷⁾ Conclusions of this study which have particular application to this analysis are:

1. Calculated core hydrogen worth based on a homogeneous core model representation is reduced by 4% per \$1.50 of TaC shim content. The core hydrogen feedback results discussed above are based on a homogeneous representation with \$3.00 TaC shim content (maximum engine concentration). The shim interaction effects discussed above suggest that these results should be increased by the factor $.96/.92 = 1.043$ so as to correspond to the nominal shim content requirement of \$1.50.
2. The average support stem hydrogen worth is 5% less than the homogeneous calculated results with \$1.50 TaC shim poison content.
3. The average propellant hydrogen worth is 1% greater than the homogeneous calculated results with \$1.50 TaC shim content.

Table 2-52 illustrates the development of the core hydrogen feedback effects to nominal rated full power start of life incorporating the above shim tube interaction effects. The first column restates the feedback effects as calculated using a homogeneous model representation with a \$3.0 TaC shim content. These results are then modified to reflect first the corresponding homogeneous result with \$1.50 shim content and finally the heterogeneous effects discussed under items 2 and 3 above. It should be noted that support stem feedback has been separated into a contribution to minimum SSA flow (closed SSCV) and a 45 ¢ contribution required for hot critical uncertainty adjustments. The latter contribution represents a requirement on reactivity adjustment and is therefore unaffected by shim interaction effects.

Formulation

The purpose of this section is to formulate an overall Composite R-1 reactivity feedback model for eventual inclusion into the TNT thermal computer program. This model is then compared with the model currently employed in this regard and documented in DRM 52764.

The results and conclusions of the preceding sections can be summarized as follows:

1. Reactivity feedback and control drum interaction effects can, in general, be examined on a per component basis and assumed separable (additive). A notable exception to this rule occurs upon examination of the combined reactivity feedback effects of core hydrogen and core temperature as indicated in Table 2-51. These results indicate a core temperature neutronic feedback enhancement effect of approximately 10% per dollar of core hydrogen worth. For example, the separate core hydrogen and core neutronic temperature feedbacks at nominal rated full power start of life were calculated to be +\$2.98 and -\$0.41, respectively. The above interaction relationship would modify the temperature feedback term by $(10\%/ \$)(2.98\$) = 29.8\%$ giving a revised result of $(-\$0.41) \times (1.298) = \0.53 . The remaining feedback terms attributable to seal, cylinder and reflector flow and beryllium temperature are additive to within a 15% discrepancy.
2. The core hydrogen reactivity feedback derived above should be modified so as to correspond to a \$1.50 TaC shim control and to account for heterogeneous shim interaction effects as discussed in the preceding section. These effects suggest a separate treatment of propellant and SSA hydrogen worth due to evidence of shim tube flux shielding. To this end, it is recommended that the hydrogen worth results of Figure

2-66 should be enhanced by 5.4% for application to propellant flow feedback and the same data should be reduced by 0.9% for application to SSA flow feedback.

3. The PAX/R-1 core hydrogen worth analysis of Section 2.4.4.1.4.1 has indicated reasonably good agreement can be expected between calculated and measured uniform core hydrogen worths.

As mentioned above, a weakness of this analysis appears to be in the treatment of core temperature reactivity feedback effects. This has been indicated from an examination of certain PHOEBUS-2A and NRX/EST calculations in which core temperature feedback appears to be grossly underestimated. This area needs considerably more effort both by more detailed analyses of available experimental data (NRX/EST and XE-PRIME) and the basic cross section data evaluation (particularly as regards resonance absorption parameters). Further definition will be provided by PAX/R-1 core temperature worth measurements now planned in 1971. These measurements will provide data over a temperature range of ambient to approximately 700°R and will attempt to differentiate the contributing effects of core expansion and core neutronic feedback. However, based on available measurements and calculations, the R-1 core temperature feedback per this analysis was corrected to reflect apparent discrepancies noted in previous NRX/EST analyses.

The reactivity feedback accompanying a core temperature increase is effected by a neutron thermal energy base shift due to reduced carbon upscatter, resonance absorber doppler broadening and core expansion. The thermal base shift and doppler broadening contributions are conveniently referred to collectively as the core temperature neutronic feedback. Core expansion feedback results from increased neutron leakage which, unlike the more complicated effects associated with neutronic feedback, should be amenable to analytical treatment. It is assumed, for the purpose of this derivation, that the core expansion feedback contribution is properly calculated.

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(CRD) NRX/EST core temperature feedback has been previously calculated using the FSGAM8 cross section library tape. As discussed in the Analytical Procedure Section 2.4.1 of this report, the present analysis is based on the FSGAM29 library tape which is an update of the FSGAM8 tape. However, separate studies (based on the PHOEBUS-2A reactor) have indicated consistency in regard to the calculation of core temperature reactivity feedback using either library tape. This derivation assumes that this consistency is also realized in the case of the NRX/EST results. The calculated NRX/EST results indicate feedback contributions of $-\$0.46$ and $-\$0.72$ to 2880°R for core expansion and core temperature neutronic effects respectively. NRX/EST mapping⁽²⁹⁾ and XE-PRIME low power measurements⁽²⁵⁾ suggest the total core temperature reactivity feedback profile of Figure 2-82. The figure indicates a feedback of $\$1.98$ to 2880°R which is greater than the aforementioned calculated results. Assuming the core expansion contribution is correctly calculated (as discussed previously) the comparison suggests that a normalization factor of 2.1 be applied to calculated NRX/EST neutronic feedback. As discussed above, zirconium doppler broadening is a major contributor to the calculated composite core neutronic feedback. The absence of zirconium in the NRX/EST suggest that the normalization factor derived above be applied only to that portion of the calculated composite neutronic feedback which excludes the zirconium doppler contribution.

(U) Tables 2-53 and 2-54 summarize the composite R-1 reactor reactivity feedback effects to nominal rated full power start of life. As discussed previously, reactivity feedback is dependent on drum angle and these results correspond to the ambient critical drum bank angle of 80° . The tables compare both the unmodified results presented above and the corresponding results after application of the normalization procedures outlined in this section. The reactivity feedback is $+\$1.08$ (at the ambient critical drum bank) and the drum worth reduction is 4.7%. The modifications of the calculated drum worth change were made by assuming direct proportionality with reactivity feedback. The hot orifice drum position is determined from the equation:

$$(1-0.047)(0.40-2.25 \cos \theta) = -1.08$$

which is based on a $\$4.50$ ambient drum span and a cold critical position of 80° . Solution of the equation gives a hot orifice bank angle of $47^{\circ}02'$.

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The corresponding results as determined by the current version of TNT are a reactivity feedback of +\$1.08 and drum worth reduction of 6% to nominal rated full power start of life. The corresponding hot orifice bank angle is 46° 50'.

The following is a formulation of the composite R-1 reactivity feedback results derived and discussed in this section. This formulation is intended as an update of the reactivity feedback model for CAM and TNT documented in DRM 52764. The reactivity feedback is formulated to correspond to the ambient critical drum bank angle of 80° (\$1.85 shutdown).

1. Core Hydrogen

Effective hydrogen mass contents are obtained by weighting the core hydrogen density distributions with the following normalized worth profiles:

AXIAL		RADIAL		
<u>Core Station (In.)</u>	<u>W_i</u>	<u>Channel</u>	<u>O.R. (In.)</u>	<u>W_i</u>
Core Inlet 0	.0357	1	11.6	1.327
4	.378	2	17.23	.857
8	.786	3	18.66	.592
12	1.10			
16	1.41		Average =	1
20	1.62			
24	1.71			
28	1.66			
32	1.50			
36	1.17			
40	.898			
44	.488			
48	.249			
52	.0254			
Average =	1			

The effective propellant and support stem mass contents are then,

$$M_P^{EFF} = \sum_{i=1}^{13} (W_i + W_{i+1}) \sum_{j=1}^3 W_j A_j^P (\rho_{i,j}^P + \rho_{i+1,j}^P) \text{ LBS}$$

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$$M_{SS}^{EFF} = \sum_{i=1}^{13} (W_i + W_{i+1}) \sum_{j=1}^3 W_j A_i^{SS} (\rho_{i,j}^{SS} + \rho_{i+1,j}^{SS}) \text{ LBS}$$

where $\rho_{i,j}^P$ and $\rho_{i,j}^{SS}$ are the propellant and support stem gas densities respectively in lbs/in^3 at axial station i and radial channel j and A_i^P and A_i^{SS} are the propellant and support stem flow areas respectively in in^2 for radial channel j .

The propellant and support stem hydrogen reactivity feedbacks are then,

$$\$_{PROP} = (2.53) M_P^{EFF} [1 - (.0212) (M_P^{EFF} + M_{SS}^{EFF})] \text{ and}$$

$$\$_{SS} = (2.38) M_{SS}^{EFF} [1 - (.0212) (M_P^{EFF} + M_{SS}^{EFF})]$$

The corresponding fractional drum worth changes are,

$$F_{\theta}^{PROP} = -(.0556) M_P^{EFF} [1 - (.0867) (M_P^{EFF} + M_{SS}^{EFF})] \text{ and}$$

$$F_{\theta}^{SS} = -(.0523) M_{SS}^{EFF} [1 - (.0867) (M_P^{EFF} + M_{SS}^{EFF})]$$

2. Core Neutronic Temperature

The reactivity feedback is,

$$\$_{CORE TEMP} = \begin{cases} -(.145) \ln(T/532.8) [1 + (.106) (\$_{PROP} + \$_{SS})] & \text{for } T \leq 532.8^{\circ}\text{R} \\ -(.295 \times 10^{-3}) \Delta T [1 - (.151 \times 10^{-3}) \Delta T] \times \\ [1 + (.106) (\$_{PROP} + \$_{SS})] & \text{for } T \geq 532.8^{\circ}\text{R} \end{cases}$$

where T is the average fuel temperature in $^{\circ}\text{R}$ and $\Delta T = T - 532.8$.

The corresponding fractional drum worth change is,

$$F_{\theta}^{CORE TEMP} = \begin{cases} 0, & \text{for } T \leq 532.8^{\circ}\text{R} \\ -(2.02 \times 10^{-5}) \Delta T [1 - (.227 \times 10^{-3}) \Delta T] \times \\ [1 + (.106) (\$_{PROP} + \$_{SS})] & \text{for } T \geq 532.8^{\circ}\text{R} \end{cases}$$

3. Core Expansion

An effective filler strip radial displacement is obtained by weighting the stationwise displacements with the following normalized sine worth profile:

<u>Core Station (In.)</u>	<u>W_i</u>
Core Inlet 0	0
4	.378
8	.734
12	1.05
16	1.30
20	1.48
24	1.57
28	1.57
32	1.48
36	1.30
40	1.05
44	.734
48	.378
52	0
Average =	1

The effective radial displacement is then,

$$\Delta R_{CORE}^{EFF} = \frac{1}{52} \sum_{i=1}^{13} (W_i + W_{i+1}) (\Delta R_i^{CORE} + \Delta R_{i+1}^{CORE}) \text{ IN.}$$

The reactivity feedback is then,

$$\rho_{CORE EXP} = - (7.44) \Delta R_{CORE}^{EFF}$$

and the fractional drum worth change is,

$$F_{\theta}^{CORE EXP} = (.277) \Delta R_{CORE}^{EFF}$$

4. Cylinder Contraction

An effective support cylinder radial displacement is obtained by weighting the stationwise displacements with the normalized sine worth profile defined above.

The effective displacement is then,

$$\Delta R_{CYL}^{EFF} = \frac{1}{52} \sum_{i=1}^{13} (W_i + W_{i+1}) (\Delta R_i^{CYL} + \Delta R_{i+1}^{CYL}) \text{ IN.}$$

and the reactivity feedback is,

$$\$_{\text{CYL CONT.}} = - (1.87) \Delta R_{\text{CYL}}^{\text{EFF}}$$

The corresponding fractional drum worth change is,

$$F_{\theta}^{\text{CYL CONT.}} = - (.23) \Delta R_{\text{CYL}}^{\text{EFF}}$$

5. Seal and Filler Strip Hydrogen

The reactivity feedback is,

$$\$_{\text{SEAL}} = (.577) M_s [1 - (.132) M_s]$$

where M_s is the stored hydrogen mass content in the filler strip and seal regions.

There is no fractional drum worth change associated with seal and filler strip flow, i.e.,

$$F_{\theta}^{\text{SEAL}} = 0$$

6. Cylinder Hydrogen

The reactivity feedback is,

$$\$_{\text{CYL}} = (.319) M_c [1 + (.366) M_c]$$

where M_c is the cylinder stored hydrogen mass content.

The corresponding fractional drums worth change is,

$$F_{\theta}^{\text{CYL}} = - (.0257) M_c [1 - (.582) M_c]$$

7. Reflector Hydrogen

The reactivity feedback is,

$$\$_{\text{REFL}} = - (.0126) M_R [1 - (2.22) M_R]$$

where M_R is the reflector stored hydrogen mass content.

There is no fractional drum worth change associated with reflector hydrogen flow, i.e.,

$$F_{\theta}^{\text{REFL}} = 0.$$

8. Beryllium Temperature

The reactivity feedback is,

$$\$_{BE\ TEMP} = \begin{cases} 0 & , \text{ for } T \leq 532.8^{\circ}R \\ - (.0579 \times 10^{-3}) \Delta T [1 + (2.21 \times 10^{-3}) \Delta T] & , \text{ for } T \geq 532.8^{\circ}R \end{cases}$$

where ΔT is the change in average beryllium temperature from $532.8^{\circ}R$.

The corresponding fractional drum worth change is,

$$F_{\theta}^{BE\ TEMP} = (7.30 \times 10^{-5}) \Delta T [1 - (.997 \times 10^{-3}) \Delta T]$$

In general, the system excess reactivity is,

$$\$_{EX} = \$_{DRUMS} (1 + F_{\theta}) + \$_{FEEDBACK}$$

where,

$\$_{DRUMS}$ is the ambient drum reactivity from ambient critical determined from the equation, $\$_{DRUMS} = 0.40 - (2.25) \cos \theta$.

F_{θ} is operating fractional drum worth change determined by,

$$F_{\theta} = \frac{(1 + F_{\theta}^{PROP}) (1 + F_{\theta}^{SS}) (1 + F_{\theta}^{CORE\ TEMP}) (1 + F_{\theta}^{CORE\ EXP})}{(1 + F_{\theta}^{CYL\ CONT.}) (1 + F_{\theta}^{CYL}) (1 + F_{\theta}^{BE\ TEMP})} - 1$$

$\$_{FEEDBACK}$ is the net cold to hot reactivity feedback determined by,

$$\$_{FEEDBACK} = \begin{cases} \$_{PROP} + \$_{SS} + \$_{CORE\ TEMP} + \$_{CORE\ EXP} + \\ \$_{CYL\ CONT.} + \$_{SEAL} + \$_{CYL} + \$_{REFL} + \\ \$_{BE\ TEMP} \end{cases}$$

2.5.1.1.6 Nuclear Environmental Drawing

Table 2-55 summarizes the salient nuclear characteristics of the composite core design. These results are consistent with present TNT (Transient Nuclear Thermal) program analyses as discussed in Data Items S-031 and S-001.

Table 2-55 shows a \$5.40*ambient control drum span based on an 80% probability with 95% confidence level which compares with a required span of \$4.50 on the same basis. The support stem reactivity swing of \$2.69 (minimum SSCV setting at full power to maximum SSCV setting throttled) is consistent with throttle mode requirements after 10 hours full power operating lifetime at fixed operating drum bank angle, a \$0.10 temperature control margin and a $\pm$ \$0.35 nominal start of life hot critical reactivity adjustment requirement. The cold start of life critical drum bank angle of 80° is based on a \$4.50 total ambient drum span with a nominal cold critical shutdown margin of \$1.85 (\$1.70 minimum). The predicted span of \$5.40 could accommodate an additional shutdown margin of \$0.90. The operating critical drum bank angle of 47° results from a \$1.08 reactivity feedback to nominal rated start of life with a 6% operating drum worth reduction. These results are corroborated by current analyses. The operating angle is shown as fixed throughout a one hour operating lifetime for both the rated and throttled modes. The cold end of life critical bank angle of 82.5° is based on a \$0.10 operating reactivity loss via corrosion and fuel burnup and negligible xenon buildup. The space storage cold critical drum bank angle of 55° is based on a uniform reactor temperature of 50°R (\$1.00 reactivity insertion from ambient) with \$0.10 operating reactivity loss. The power distributions (max and min radial core power factors) shown are based on a fixed hot critical drum bank position throughout the operating lifetime in both the rated and throttled operating modes. The power distribution is unaffected by rated full power operation but an 8% peripheral power factor reduction is associated with the throttled mode. This reduction suggest the desirability of compensating drum movement in the throttled mode. Such movement (approximately \$0.75) would relax the support stem reactivity requirement without imposing an additional requirement on control drum span.

*Derivation of this result does not include provision for design changes between successive reactors. NRX experience showed a net \$0.80 decrease in drum span from NRX-A2 to XE-PRIME.

2.5.1.2 Transient Analysis

The analysis of the transient behavior of the NERVA reactor is a mutual undertaking of various WANL disciplines. Nuclear Design provides the following neutronic data: effective neutron lifetime, effective delayed neutron fraction, control drum worth, and temperature/propellant reactivity worth coefficients. Systems Analysis employs the PAM model using these coefficients to perform transient analyses over a wide spectrum of parameters to select an operating profile. The profile thus selected from the PAM model is employed by Thermal Design in a more detailed TNT-II model to determine the detailed transient behavior of the NSS. The results of recent PAM and TNT-II studies are presented below to examine if the required control drum maneuvering of the NUCLEAR SUBSYSTEM is consistent with that available in the drum span. For details of the transient analyses, see Data Items S-001 (Systems Analysis Summary Report) and S-031 (Thermal and Fluid Flow Analysis Report).

2.5.1.2.1 Kinetic Parameters

Nuclear Design determined the neutron kinetic parameters β_{eff} (effective delayed neutron fraction) and λ^* (the effective prompt neutron lifetime) for the flight graphite reactor.⁽³⁰⁾ The values were determined with the PERT-1D perturbation code in a 52-energy group calculation. Inputs to the PERT-1D code included forward and adjoint neutron fluxes generated with 52-group radial ANISN calculations. The parameters thus determined are:

$$\begin{array}{l} \text{and} \quad \beta_{\text{eff}} = 0.0070 \\ \lambda^* = 10.6 \times 10^{-6} \text{ sec.} \end{array}$$

The above values were transmitted to the Systems Analysis and Thermal Design Department for incorporation into the PAM and TNT-II models for both the graphite and composite designs.

2.5.1.2.2 PAM and TNT-II Studies

As noted above, the PAM model was employed to select operating profiles for NERVA startup and shutdown. The profiles described here are representative of the extreme transients. The assumed operational procedure which was used during startup and shutdown is described in detail S-031. The power profile is illustrated in the top portion of Figure 2-83. The major

characteristics of the operating profile are discussed below.

The nuclear startup phase was not treated explicitly in the PAM and TNT-II studies. The studies assumed that a dry nuclear startup had been achieved and a power of 4 MW was attained at the initiation of the thermal conditioning (chilldown) phase.

The bootstrap phase was tailored to achieve a chamber pressure (P_c) of 60 psia, at a chamber temperature (T_c) of 1130°R. During bootstrap the control drums were positioned by an open loop control to obtain a 1.6 second period (e-folding time) on the power increase.

During Thrust Buildup, Full Power Operations, and Throttling, drum control was set on T_c demand with an open loop profile on the support stem control valve (SSCV) position. The open loop profile on SSCV position was varied to minimize drum motion. The reader should consult Data Item S-001 for a detailed discussion of the operating profile selection.

For the Temperature Retreat, the drums were ramped in at a maximum rate of 10 degrees/sec to 0°, the chamber temperature was ramped at the specification maximum (-175°R/sec), and the SSCV was stepped to its minimum flow position.

Portions of the TNT-II startup and shutdown studies of interest to Nuclear Design are discussed below to illustrate the control drum, and support stem flow requirements for reactivity control. In Figure 2-83 the reactivity plots show the Impressed Reactivity, which is due to drum motion; Feedback Reactivity, which includes the thermal, hydrogen flow, and corrosion effects in the core and reflector; and the NSS Reactivity, which is the sum of the Impressed and Feedback components. It should be noted that the NSS Reactivity employed here is titled Excess Reactivity in S-001 and S-031.

2.5.1.2.2.1 Startup Operations

The Startup Operations as defined by Specification CP-677555A consists of three major subdivisions: Temperature Conditioning and Nuclear Startup, Bootstrap, and Thrust Buildup. The Temperature Conditioning phase is subdivided further into Reactor Thermal Conditioning, Nuclear Startup, and Complete Engine Temperature Conditioning. The Reactor Thermal Conditioning phase is not considered in the transient analyses described here. The Nuclear Startup is the act of bringing the reactor power from the subcritical source level to a power where sensible heat is produced. The Complete Engine Temperature Conditioning is replaced by the chilldown phase which is included in the present discussion. The Bootstrap and Thrust Buildup are sequences which bring the entire NERVA engine up to a rated power.

The Nuclear Startup may be accomplished before hydrogen flow is initiated during the chilldown; this is termed the dry start. Alternatively, the Nuclear Startup may be initiated simultaneously with the hydrogen flow for chilldown; this is termed the wet start. The method by which the NERVA Nuclear Startup will be achieved has not been selected. Previous NSS specifications have stated that the nuclear power be brought up from subcritical to 1 Mw in 45 seconds or less prior to initiation of a 25-second chilldown. The startup was to be accomplished by ramping the control drum to the predicted cold critical angle (drum control phase) and then continue ramping out until a period of approximately two seconds is achieved (period demand phase). This dry start procedure would require a maximum drum roll-out approximately 98° with SOL nominal reactivity feedback and out to approximately 125° with EOL nominal reactivity feedback to achieve 70¢ excess reactivity. This is based on a cold critical drum position of 80° . It is noted that presently 70¢ is provided in the control drum reactivity span for this type of startup.

For illustration purposes, a dry nuclear start was assumed in which a conservative two-step program was employed for rolling out the control drums. The assumed startup

procedure is to ramp the drums out at the maximum rate (nominally 10 degrees/sec) to some arbitrary point, say 85° at SOL and then ramping out more slowly at some convenient rate, say $0.2^{\circ}/\text{sec}$, until the desired period or power level is attained. The drums are then ramped back to the critical angle at the maximum rate. At EOL, the corresponding drum program would include Corrosion Compensation of $\$1.00$. This procedure is illustrated in the Drum Position versus Time plot in Figure 2-83 as a representative startup prior to the chilldown phase.

The presence of fission product buildup and fuel burnup must be considered in determining the drum motion which is required for the worst case startup. Analyses show that the upper bound of fission product buildup at 10.5 hours after shutdown from one hour of full power operation plus fuel burnup at the end of life amounts to approximately -23% in the composite core. This reactivity is an upper bound which includes the enhancement of the fission product worth when control drums are near 180° .

During the chilldown phase, drum movement is employed to maintain a constant power of 4 Mw during the chilldown. This phase is illustrated in Figure 2-83 by dashed lines showing constant drum positions from 0 to 25 seconds.

During the Bootstrap and Thrust Buildup phases, the drum motion is correlated to feedback reactivities and flow conditions in the core and support stems as well as reactivity demand.

Bootstrap phase is initiated (at 25 seconds) as a 1.6 second period demand for 7.6 seconds duration by drum motion. The required NSS reactivity is controlling during this phase. Initially, all NSS reactivity is provided by outward drum motion until feedback reactivity increases (from 0 at SOL and from $-\$1.00$ at EOL) to contribute to the total NSS reactivity. The maximum outward motion of the drums is to 94.5° with SOL nominal reactivity feedback and to 121.5° with EOL nominal reactivity feedback. The 70% provided for the dry nuclear autostart in the control drum reactivity span is also sufficient for drum requirements in the Bootstrap phase. As in the Nuclear Startup discussion, when the maximum negative reactivity of fission and product buildup and fuel burnup of -23% must also be overridden, the drums must move outward by an additional amount.

Following the Bootstrap phase, the demand profile for Thrust Buildup is $175^{\circ}\text{R}/\text{sec}$ in T_c (to simulate the extreme rate of temperature buildup) and $13\text{ psi}/\text{sec}$ in P_c until the rated temperature of 4250° is reached. The NSS reactivity is the same for SOL and EOL. During this time period, the SSCV position is changed by open loop profile so that control drums can be rolled in to the orifice angle by the time the throttle point is reached while maintaining the demand profile.

After the temperature 4250°R is achieved at a chamber pressure of 293 psia (the Throttle Point), P_c is increased to the rated value of 450 psia at a ramp of $50\text{ psia}/\text{sec}$. The SSCV is ramped inward causing the support stem flow to be reduced at this time, while the total reactivity feedback increases. Hence, a compensating inward motion of the drum is required in this operating profile to limit the NSS reactivity. In this region, the drums attain their minimum inward position during Startup Operations. In this operating profile at SOL, the minimum angle is 34° and at the EOL, the minimum angle is 24.6° . The drum worth which is available for shutdown at the point of minimum drum position is approximately -37% at SOL and 23% at EOL. These conditions are attained when the peak support stem flow is realized during startup. Without feedback uncertainties, there seems to be an adequate shutdown margin during SOL startup but a flow adjustment may be necessary in the EOL profile to assure the required -25% shutdown margin. With a limitation on available flow control, it may be necessary to provide an additional 45% in the drum span to account for the uncertainties in the cold-to-hot feedback predictions in order to assure the 25% shutdown requirement during all phases of the startup procedure. It is emphasized that the interpretations above are based on a profile which simulates extreme thermal transients and must be reexamined when optimal operating profiles are defined.

Finally, as the Full Power phase is approached, the NSS reactivity is reduced to zero as the drums are moved toward the nominal hot critical angle of 47° (Orifice Angle). The Orifice Angle is reached when hot equilibrium is established. For hot equilibrium with a drum angle of 47° , the support stem flow at EOL is greater than at SOL in order to compensate for corrosion. For precise numerical values of flow rates and drum angle, the reader should consult Data Item S-031.

2.5.1.2.2.2 Throttling

The transient analyses assumed that throttling action was initiated at 600 seconds following a 10-minute steady state operation at Rated Conditions. The throttling demand is a -50 psia/sec rate of decrease in P_c with T_c held at 4250°R . The Support System flow is increased during throttling and the drums are rolled inward from the Orifice Angle at the initiation of Throttling. For the profile considered here, the drums move in to approximately 38° for EOL throttle initiation but only to about 42° for SOL throttle initiation. During a Throttle Hold, equilibrium is again established with the drums at the Orifice Angle of 47° .

2.5.1.2.2.3 Shutdown

The normal mode shutdown is initiated with Throttling and a Throttle Hold. The analyses described herein employed a 40-second Throttle Hold. The discussion of Section 2.5.1.2.2.2 is applicable to the initial phases of Shutdown. The corresponding reactivities, drum position and support system flow rate during shutdown are shown in Figure 2-83.

The Temperature Retreat phase begins at the conclusion of the Throttle hold. In the study described here, the Temperature Retreat is taken at $-175^\circ\text{R}/\text{sec}$ from 4250°R to 1800°R . Simultaneously with the initiation of Temperature Retreat, the drums are rolled in at the rate of 10 degrees/sec and the SSCV is ramped to its minimum flow position. During this time, P_c and core flow are increased to provide sufficient cooling to achieve the temperature ramp. Consequently, the core propellant flow and feedback reactivity reach a peak after the initiation of Temperature Retreat. The feedback reactivity increases very rapidly during this time. There is the possibility that the corresponding peak in NSS reactivity could exceed zero, prohibiting shutdown. This situation is seen to be quite possible in Figure 2-83 for the SOL but is less probable at EOL due to $-\$1.00$ loss in reactivity due to corrosion. However, the uncertainties in the NUCLEAR SUBSYSTEM may be such that the drum reactivity insertion from hot critical to the full-in drum position is less than 70% noted in the shutdown

profile illustrated here. Thus, it appears that the drum span utilization may have to be increased in order to insure a successful shutdown. For the operating profile discussed here, at least 45% additional drum span is required to account for NSS uncertainties, but this requirement must be reexamined when an optimal shutdown profile is defined.

2.5.1.2.3 Summary of Drum Requirements During Transients

As discussed in the previous sections, the drum span requirement during startup either for a dry Nuclear Startup or for rapid power buildup during Bootstrap is less than or equal to the 70% presently provided in the drum utilization scheme (See Section 2.5.1.1.4.1.). However, the results of the transient analysis indicated that an additional 45% in the drum span allotment may be necessary for some operating profiles to account for uncertainties so that adequate shutdown reactivity is available at all times during startup and shutdown procedures. However, the reader should consult Data Items S-001 and S-0031 for conclusions and recommendations on system performance. In particular, S-001 considers the system performance for other profiles as well as the profile described here.

2.5.1.2.4 Feedback Reactivity Map($T_c - P_c$)

The total core and reflector reactivity feedback for the composite core shown in Figure 2-84 as contours on a $T_c - P_c$ diagram for the conditions of minimum support stem flow and maximum support stem flow. These data were obtained from the PAM model but are in agreement with the results of the TNT-II model at those points where data are available. The break point at 4250°R and 293 psia is the Throttle Point. It is seen that the reactivity feedback at the Throttle Point with maximum SSCV opening is +\$2.43. Similarly, the reactivity at full power (4250°R and 450 psia) with minimum stem flow is seen to be \$0.63. (These values were obtained with the TNT-II model.)

The results of the transient analyses show that throttled operation is possible with virtually no compensating drum motion for nominal parameters. The reactivity feedback utilization which is necessary to maintain the control drums at the nominal hot critical angle

(orifice angle) is shown in Table 2-56. The total reactivity feedback requirement for full power operation through the end of life with an allowance for uncertainties is \$2.43. Since this amount of feedback is also available at the Throttle Point with the maximum SSCV opening, the drums can be maintained at the orifice angle in the throttled mode if the system performs as specified by the nominal NSS parameters. It should be noted that at full power, the support stem swing is considerably greater than the 1.8\$ required to compensate for corrosion, fuel burnup, nuclear uncertainties and controllability.

However, it may be desirable to move the drums away from the orifice angle in throttled operation in order to shape the edge power. The edge power profile is influenced by reduced hydrogen flow in the reflector in the throttled mode. This effect is discussed in Section 2.5.1.1.2.3.

2.5.2 Graphite Core

This section presents the detailed nuclear results for the NSS containing an HCTE graphite fueled core.

2.5.2.1 Steady State

2.5.2.1.1 NSS Absolute Reactivity - Fuel Loading Distribution

The predicted NSS absolute reactivity and corresponding fuel loading distribution was developed using the Method of Analysis described in Section 2.4.1.3. Utilizing the graphite core calculational model in conjunction with the 16-group regionwise spectrally weighted cross sections, R-θ EXTERMINATOR problems were set up for the 0°, 50°, and 180° drum bank positions. These problems incorporated a negative shim worth of \$1.5 distributed uniformly throughout the core. The calculation resulted in an absolute reactivity value of -\$0.42 for the 50° drum position. The prescribed initial cold critical drum position is approximately 80°. Utilizing the results of the three graphite EXTERMINATOR problems (0°, 50°, and 180° drum positions) to determine the change in reactivity as a function of drum angle, the calculated value of reactivity at the 80° cold critical position is +\$1.17. Correcting the EXTERMINATOR reactivity for a pressure vessel, instrumentation, external structure and nozzle results in a calculated reactivity at the 80° cold critical position of +\$2.02. The maximum uranium loading utilized in this graphite analysis is 630 mg/cc.

In this reactivity analyses the bias in predicted absolute reactivity of -\$1.56 (developed in Section 2.4.4.1.1.) was utilized. Therefore based on the above EXTERMINATOR results and this calculated reactivity, the graphite core loading may be decreased by \$0.46 (\$2.02-\$1.56) to achieve the desired cold critical drum bank angle. This \$0.46 reactivity reduction would result in a decrease in the uranium inventory of approximately 1.6%, resulting in a revised maximum uranium loading of 620 mg/cc. In the requirement section, (Section 2.3) it is stated that the RA shall be compatible with a maximum graphite loading of 630 mg/cc and an initial \$1.5 positive reactivity at assembly prior to shimming. Both

calculational models incorporate on initial \$1.5 shim content, and as noted above, the graphite maximum loading of 620 mg/cc meets the 630 mg/cc loading requirement.

As an additional aid in the prediction of the R-1 graphite absolute reactivity, spatial material reactivity coefficients will be measured in the subsequent PAX-G3 critical configuration. These will be used in conjunction with the graphite fuel element composition to perform a mass/reactivity balance.

2.5.2.1.2 Radial Power Distribution

The NERVA radial fission distribution was also developed using the Method of Analysis described in Section 2.4.1.3. The R-θ EXTERMINATOR problem at the 50° drum bank position was used for this analysis; this problem incorporated a negative shim worth of \$1.5 uniformly distributed over the entire core. The orifice drum angle was initially assumed at 50°, the final TNT angle was 58°. The output of this EXTERMINATOR results in a calculated pointwise radial and azimuthal fission flux distribution for the preliminary fuel loadings. Table 2-57 summarizes this multizoned fuel distribution. As noted, this distribution is composed of thirteen separate fuel elements loadings. Figure 2-25 presents the elemental loading distribution which results in the fifteen radial core region breakup. As noted, regions 11 and 14 have the same loadings, as do regions 6 and 15. Figure 2-26 shows the desired radial fission distribution at the 50° orifice drum bank angle. This shape is restricted by the requirement that the maximum power factor must not exceed 1.1.

As shown in the analytical flow charts of Section 2.4.1, the detailed radial power factor calculations are performed using the PFENCER series of codes in conjunction with the EXTERMINATOR output fission fluxes and the PAX/R-1 experimental power measurements and normalization coefficients. A comparison of the PAX/R-1 power measurements and predictions have shown that the calculation underpredicts the measured values by as much as 20 to 30% at the core periphery. The ratio of the PAX/R-1 measured to calculated fissions/gm for each channel normalized to unity at the core centerline is defined as the experimental correction

factor, NC_i . Figure 2-27 shows the distribution of the PAX-G2 NC_i s which were used in the development of the preliminary R-1 loading distribution. Consequently, the power distribution calculated by EXTERMINATOR at the correct drum position will result in a significant error in the peripheral power when experimentally measured. Therefore, the desired fission distribution (Figure 2-26) was divided by the PAX-G2 NC_i distribution to obtain the initial aim EXTERMINATOR power, (See Figure 2-28.). Figure 2-85 shows a comparison of the aim EXTERMINATOR distribution (Figure 2-28) and the output of the 50° EXTERMINATOR problem. As noted, there is fair agreement between the power distribution based on the preliminary loading distribution utilized in the graphite model and the aim EXTERMINATOR distribution.

2.5.2.1.2.1 Cold-To-Hot Power Distribution Change

The radial fission power profile change for the graphite core reactor has been calculated in going from ambient conditions to nominal rated full power at start of life. In the graphite core reactor, cold critical is at a control drum bank position of 80° and nominal hot critical is at an angle of 58°. Reactivity changes and fission power profile perturbations due to drum rotation, seal, cylinder and reflector hydrogen flow, reflector and core temperature change, core expansion, cylinder contraction and core hydrogen flow were calculated as a function of control drum bank position.

Figure 2-86 presents the fractional fission power perturbation per dollar of reactivity as function of radial position in the graphite core reactor for outward control drum bank rotation. Figures 2-87 through 2-94 present respectively without and with a control drum bank representation these power coefficients for seal, cylinder and reflector hydrogen flow and for peripheral flow which is the combined effect of these three components. Figures 2-95 through 2-98 show respectively both without and with a control drum bank mockup, the power coefficients for reflector temperature change and for the net effect of peripheral hydrogen flow, and reflector temperature change. Figures 2-99 through 2-106 present without

and with a control drum bank mockup the power coefficients for core temperature change, core expansion, cylinder contraction, and core hydrogen flow, respectively. It has been assumed that the power coefficients for peripheral hydrogen and reflector temperature effects are the same for the graphite core reactor as they were for the composite core reactor.

Table 2-58 presents the fractional fission power perturbation per dollar of reactivity at the core edge, as calculated both without and with a control drum bank representation for outward control drum rotation, seal, cylinder and reflector hydrogen flow, reflector and core temperature change, core expansion, cylinder contraction and core hydrogen flow. Table 2-58 also shows the power coefficients for each of these components at the core edge as extrapolated to nominal hot critical, 60° , where the control drum bank has a worth to removal of \$5.16.

Based on these power coefficients extrapolated to nominal hot critical and the reactivity changes in each of these components as determined in Section 2.5.2.1.5, a net decrease of 3.9% was calculated in the fission power at the graphite core edge in going from ambient conditions to nominal rated full power at start of life. Figure 2-107 presents the total fractional fission power change as a function of radial position in the graphite core in going from ambient conditions to nominal rated full power at start of life, using the power coefficients as calculated with a control drum bank representation and the reactivity changes as defined in Section 2.5.2.1.5.

2.5.2.1.2.2 Local Fuel Element Power Perturbations

The local fuel element power perturbation is treated in Section 2.5.1.1.2.2.

2.5.2.1.2.3 Operating Power Distribution Change

The radial fission power profile change for the graphite core reactor after a 10 hour reactor operating lifetime and for throttled mode operation has been calculated and that combination of SSA flow and drum reactivity compensation determined which results in a "flat" end of life profile. Reactivity effects and fission power profile perturbations due to drum rotation, SSA flow, corrosion, fuel burnup and fission product poisoning were calculated as a function of control drum bank position.

The calculations correspond to a total reactivity loss of \$1.12 which was found to occur at the end of 10 hours reactor operation at the nominal hot critical bank position, based on an assumed \$1.00 loss from corrosion and a 12¢ loss due to fuel burnup. The one-dimensional transport calculations performed with and without a control drum bank representation indicate at the nominal hot critical angle 58° a negligible reactivity loss associated with fission product poisoning. The ANISN calculations with no control drum representation, for the graphite core reactor, show approximately a factor of four reduction in Xe-135 worth when compared to studies based on a different reflector design*. This change in Xe-135 reactivity worth is mainly due to substantial changes in the thermal flux between the two designs. EXTERMINATOR-2 calculations for the two designs show a factor of 3.5 reduction in the thermal flux for the present design relative to the previous one. In addition, the use of updated Xenon-135 thermal cross section data (available on BIT library tape CGBIT15 rather than CGBITII) accounts for a further factor of 1.2 reduction in Xe-135 worth in the present graphite core design.

Figures 2-108 through 2-116 present the fractional fission power perturbation per dollar of reactivity as a function of radial position in the graphite core reactor for drum rotation, and without and with a control drum bank representation for SSA flow, corrosion, fuel burnup and fission product poisoning, respectively. Table 2-59 presents the fractional fission power perturbation per dollar of reactivity both without and with a control drum bank representation at the core edge. In this analysis the fractional fission power change per dollar of reactivity as calculated with the control drum bank representation was used, since as can be seen from Table 2-59 these power coefficients are not strongly dependent on drum bank position and as noted above at the nominal hot critical angle of 58° there is only a 12¢ reactivity loss due to fuel burnup and no significant reactivity loss associated with fission product poisoning.

* Drawing Number 940J143.

Calculations show that a compensatory \$1.12 SSA flow reactivity change will result in no significant power change at the edge of the core and therefore, no compensatory drum rotation is required. Figure 2-117 presents these radial fission power profile changes due to corrosion, fuel burnup and SSA flow at the end of 10 hours reactor operation.

In the throttle mode of reactor operation the source net hydrogen worth is maintained in the core, in that the reactivity change due to the decreased propellant flow is compensated for by an increase in SSA flow to produce the source net core reactivity effect. Therefore on a gross basis due to hydrogen and core temperature there will be no change in the radial fission power profile in going from the operating to the throttle condition. However, there is a change in the radial fission power profile in the graphite core reactor due to the decreased peripheral hydrogen flow in the throttle mode of operation which is assumed to be comparable to that determined for the composite core reactor, as discussed in Section 2.5.1.2.3. The analysis for the composite core reactor determined a net decrease of 7.6% in the fission power at the core edge in going from the operating to the throttle mode of operation. The required outward control drum motion to maintain a relatively flat power distribution is approximately \$0.75.

2.5.2.1.3 Axial Power Distribution

The axial power distribution results are presented and discussed in Chapters 6 and 7.

2.5.2.1.4 Total Control Drum Span

For completeness, the total control drum span analysis is included under NSS. Further detailed information on the reflector design and calculations are included in Section 8.5.

The R-1 total control drum span was developed using the Method of Analysis described in Section 2.4. The difference in reactivity between the two EXTERMINATOR problems ($\rho_{180^\circ} - \rho_{0^\circ}$) corresponds to the R-1 calculated control drum span for the graphite fuel and the reference reflector assembly. The results of these two EXTERMINATOR calculations indicate the calculated control drum span to be \$6.84.

In Section 8.4.4, the drum span experimental normalization factors are developed. These were generated utilizing the following reactors: NRX-A6, PAX-G0, PAX-G1, PAX-G2, and PAX-G3. The experimental normalization factor developed statistically from these individual values for an 80% probability and 95% confidence level, to insure a \$4.5 drum span, is 0.782. Using this 0.782 coefficient and the EXTERMINATOR drum span calculation, the final total control drum span for the graphite fuel is \$5.35 to $(0.782 \times \$6.84)$.

The specified requirement for preliminary design is that the drum span be at least \$4.5 based on an 80% probability and a 95% confidence level. Based on this requirement, the drum span value of \$5.35 meets the specified value of a \$4.50 minimum.

2.4.2.1.4.1 Control Drum Utilization

See Section 2.4.1.1.4.1.

2.5.2.1.5 Cold-To-Hot Reactivity Feedback Analysis

(U) This section develops the formulation of the cold-to-hot feedback effects expected during operation of the graphite reactor. The analytical method employed in the analyses is discussed in Section 2.4.1.3. The first subsection is a separate treatment of the various terms contributing to the total feedback without regard to interaction effects. The second subsection discusses the corresponding control drum interaction effects indicated from the analysis. The third subsection considers interdependencies between the separate feedback effects. The next two subsections discuss the analyses of the reactivity effects of para/ortho hydrogen conversion and shim tube interaction respectively. Finally, these results are factored into the formulation of a reactivity feedback model. This feedback model is compared with that now incorporated in the thermal TNT program.

Separable Feedback Effects

(U) Each of the following effects represent perturbations to an ambient reactor system without flow and with a cold critical control drum bank angle of 80° .

(U) 1. Core Hydrogen Worth - Figure 2-118 shows the reactivity feedback variation with uniform core hydrogen mass content. At nominal rated full power start-of-life, the "effective" stored hydrogen mass content is 1.14 pounds which corresponds to a \$2.70 reactivity feedback as noted in the figure. An effective mass content is required due to the axial variations in propellant and SSA hydrogen densities and hydrogen worth.

(CRD) 2. Core Temperature Worth - Figure 2-119 shows the reactivity feedback variation with core temperature excluding core expansion effects. At nominal rated full power start-of-life, the average core temperature is 3150°R which corresponds to a reactivity feedback of \$0.30 as noted from the figure. These results do not treat differences in fueled and central support element temperatures. It should be pointed out that sufficient cause exists for questioning these calculated core temperature neutronic feedback results. In particular, the calculations appear to significantly underestimate this feedback. Further discussion of this seeming discrepancy is included below.

(U) Core temperature neutronic feedback in the composite R-1 reactor is effected by uranium and niobium resonance doppler broadening and thermal neutron energy base shift with

[REDACTED]
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carbon temperature. Table 2-60 shows the relative contributions of these effects as calculated for the case of control drums removed. As is noted, uranium and niobium resonance doppler broadening appear to be the major contributors to the core temperature neutronic feedback.

3. Filler Strip and Seal Hydrogen Worth - Figure 2-120 shows the reactivity feedback variation with combined hydrogen mass content in the filler strip and seal regions. Filler strip flow contributes less than 3% to the combined mass content. At nominal rated full power start-of-life the combined filler strip and seal region hydrogen mass content is 0.292 pounds which corresponds to a \$0.162 reactivity feedback as noted in the figure. These results and those discussed below for cylinder and reflector hydrogen worths are based on a 37°R Para hydrogen scatter kernel. The effects of species conversion, Para → Ortho, and gas temperature are considered below.

4. Support Cylinder Hydrogen Worth - Figure 2-121 shows the reactivity feedback variation with hydrogen mass content in the aluminum support cylinder. At nominal rated full power start-of-life the support cylinder mass content is 0.405 pounds which corresponds to a \$0.150 reactivity feedback as noted from the figure.

5. Reflector Hydrogen Worth - Figure 2-122 shows the reactivity feedback variation with reflector hydrogen mass content. At nominal rated full power start-of-life the reflector hydrogen mass content is 1.208 pounds which corresponds to a \$0.027 reactivity feedback as noted from the figure.

6. Beryllium Temperature Worth - Figure 2-123 shows the reactivity feedback variation with beryllium temperature. At nominal rated full power start-of-life, the average beryllium temperature is 252°R and has an essentially neutron effect on reactivity feedback as noted from the figure.

7. Support Cylinder Displacement Worth - Figure 2-124 shows the reactivity feedback variation with support cylinder radial displacement upon cooldown. These results assumed a corresponding reflector movement due to attachment via the aluminum lateral support spring capsules. Temperature contraction of these members is neglected as a cursory

examination of the thermal coefficients of expansion indicate this would have a small secondary effect to that of the overall displacement. Cylinder displacement data from RAT's data runs were given earlier in the graphite materials properties (Section 2.4.2.). Effective displacements are developed in Table 2-61 by reactivity sine weighting these data. At nominal rated full power start-of-life the effective cylinder radial displacement is -0.077 in. which corresponds to a \$0.143 reactivity feedback as noted in Figure 2-124. A maximum effective cylinder displacement of -0.080 in. occurs at 39.5 seconds into the SOL startup. At this point the feedback effect would be \$0.148 as is also noted in the figure.

8. Core Expansion Worth - Figure 2-125 shows the reactivity feedback variation with percent core expansion. Isotropic core thermal expansion properties are assumed. Core expansion displacement data from RAT's data runs were given earlier in the graphite materials properties Section 2.4.2. Effective displacements are developed in Table 2-62 by reactivity sine weighting these data. At nominal rated full power start-of-life the effective core radial displacement is 0.85% which corresponds to a -\$1.20 reactivity feedback as noted in Figure 2-125. At 39.5 seconds into the SOL startup the effective radial core displacement is 0.41% which corresponds to a -\$0.58 reactivity feedback as is also noted in the figure.

Control Drum Interactions

Each of the effects contributing to reactivity feedback examined in the previous section were analyzed as a function of control drum bank position. Variations in feedback with drum bank position are interpreted as control drum worth variations using the method discussed under the analytical procedures Section 2.4.1 of this report. The results are as follows:

1. Drum Worth Dependence on Core Hydrogen Mass - Figure 2-126 shows the control drum worth variation with uniform core hydrogen mass content. As is noted, control drum worth is reduced by 5.5% due to core hydrogen at nominal rated full power start-of-life.
2. Drum Worth Dependence on Core Temperature - Figure 2-127 shows the control drum worth variation with core temperature excluding core expansion effects. Control drum worth is reduced by 2.95% due to core temperature at nominal rated full power start-of-life.

Again, as was discussed above, calculated core temperature neutronic effects are suspect and requires further elaboration as will be given below.

3. Drum Worth Dependence on Filler Strip and Seal Hydrogen Mass - The calculations have indicated less than a 1% drum worth variation over a range of mass content of twice that at nominal rated full power start-of-life.

4. Drum Worth Dependence on Support Cylinder Hydrogen Mass Content - Figure 2-128 shows the control drum worth variation with aluminum support cylinder hydrogen mass content. Control drum worth is reduced by 0.8% due to support cylinder mass at nominal rated full power start-of-life.

5. Drum Worth Dependence on Reflector Hydrogen Mass Content - The calculations have indicated less than 1% drum worth variation over a range of mass content of twice that at nominal rated full power start-of-life.

6. Drum Worth Dependence on Beryllium Temperature - Figure 2-129 shows the control drum worth variation with beryllium temperature. Control drum worth is reduced by 2.4% due to beryllium temperature change at nominal rated full power start-of-life.

7. Drum Worth Dependence on Support Cylinder Displacement - Figure 2-130 shows the control drum worth variation with support cylinder radial displacement. Control drum worth is enhanced by 1.52% due to support cylinder inward displacement at nominal rated full power start-of-life.

8. Drum Worth Dependence on Core Expansion - Figure 2-131 shows the control drum worth variation with percent core expansion. Control drum worth is enhanced by 4.95% due to core expansion at nominal rated full power start-of-life.

Separability of Effects

The above effects were investigated for additivity of reactivity feedback and the results are tabulated in Table 2-63. The table compares certain combined reactivity feedbacks without control drums with the corresponding results assuming additivity is valid. Parentheses are used to define manners of addition. For example, the case denoted as $H_c + (H_p + T_R)$ compares the combined calculated effect of core hydrogen, peripheral

hydrogen and reflector temperature with the sum of the calculated core hydrogen effect and the calculated combined effect of peripheral hydrogen and reflector temperature. These results correspond to hydrogen masses and material temperatures which are in general typical of rated full power start-of-life conditions. An exception is reflector temperature which corresponds to an extreme variation of this parameter. In general, the results indicate that additivity of the separated reactivity feedback effects are additive. The largest discrepancy occurs in the addition of core hydrogen and core temperature effects which gives a result

13 % greater than the calculated combined result. Additivity of drum interaction effects was not repeated for the graphite reactor and the general conclusion that these effects are additive is assumed from comparison with the composite results.

The effects of core and cylinder displacements have not been considered in regard to this separability analysis. Further analyses will be required to ascertain whether these effects are properly additive.

Para/Ortho Conversion

Upon comparison of the graphite and composite results presented above and Section 2.5.1.1.5, respectively, it is noted that peripheral feedback effects are the same for either reactor. This observation suggests that the composite para $\rightarrow$ ortho conversion effects discussed in Section 2.5.1.1.5 should be applicable to the graphite reactor. As noted in that section the maximum peripheral hydrogen reactivity feedback variation is only about 10% over a range of 100% para to 100% ortho and a gas temperature variation from 139°R to 530°R.

Shim Tube Interaction

This analysis assumes that the shim tube interaction effects discussed in the PAX-G2 ISA test report ⁽²⁷⁾ are applicable to both the graphite and composite reactors. Section 2.5.1.1.5 discusses the repercussions of these effects to the calculated composite core hydrogen worth and the same conclusions apply to the graphite reactor. Namely, propellant hydrogen worth is 5.4% greater and support stem hydrogen worth is 0.9% less than that given in Figure 2-118.

Formulation

The purpose of this section is to apply approximate normalization factors to the calculated feedback effects presented above to obtain a final estimate of reactivity feedback and drum worth change to nominal rated full power start of life. These results are then compared with the current TNT results.

In general, the conclusion outlined in the Composite Formulation Section 2.5.1.1.5 are applicable. Namely,

1. The reactivity feedback terms are generally additive with the exception of an interaction effect noted between core hydrogen and core neutronic temperature whereby feedback is enhanced by 10% per dollar of core hydrogen worth.
2. The calculated hydrogen worth results of Figure 2-118 should be increased by 5.4% and reduced by 0.9% for application to propellant and support item feedback to minimum stem flow respectively. These factors reflect both an adjustment of the calculated worth to correspond to a nominal \$1.50 TaC shim content and the heterogeneous shim tube interaction effects discussed above.
3. A normalization factor of 2.1. should be applied to the calculated core neutronic temperature feedback to reflect discrepancies noted in earlier NRX/EST analyses.

Tables 2-64 and 2-65 summarize the graphite R-1 reactivity feedback effects to nominal rated full power start of life. The results correspond to the ambient critical drum angle of 80°. The tables compare both the unmodified results presented above and the corresponding results after application of the normalization procedures outlined above. The results indicate a reactivity feedback of \$1.23 (at the ambient critical drum bank) and a drum worth reduction of 10.2%. The corresponding results as determined by the current version of TNT are a reactivity feedback of +\$0.75 and a drum worth reduction of 5.9%.

The following is a formulation of the graphite R-1 reactivity feedback results derived and discussed in this section. This formulation is intended as an update of the reactivity feedback model for CAM and TNT documented in DRM 51519. The reactivity feedback is formulated to correspond to the ambient critical drum bank angle of 80° (\$1.85 shutdown).

1. Core Hydrogen

Effective hydrogen mass contents are obtained by weighting the core hydrogen density distributions with the following normalized worth profiles:

AXIAL		RADIAL		
<u>Core Station (In.)</u>	<u>W_i</u>	<u>Channel</u>	<u>O.R. (In.)</u>	<u>W_i</u>
Core Inlet 0	.0357	1	11.6	1.327
4	.378	2	17.23	.857
8	.786	3	18.66	.592
12	1.10	Average =		
16	1.41			
20	1.62			
24	1.71			
28	1.66			
32	1.50			
36	1.17			
40	.898			
44	.488			
48	.249			
52	.0254			
Average =		1		

The effective propellant and support stem mass contents are then,

$$M_P^{EFF} = \sum_{i=1}^{13} (W_i + W_{i+1}) \sum_{j=1}^3 W_j A_{ij}^P (\rho_{i,j}^P + \rho_{i+1,j}^P) \text{ LBS}$$

$$M_{SS}^{EFF} = \sum_{i=1}^{13} (W_i + W_{i+1}) \sum_{j=1}^3 W_j A_j^{SS} (\rho_{i,j}^{SS} + \rho_{i+1,j}^{SS}) \text{ LBS}$$

where $\rho_{i,j}^P$ and $\rho_{i,j}^{SS}$ are the propellant and support stem gas densities respectively in lbs/in³ at axial station i and radial channel j and A_j^P and A_j^{SS} are the propellant and support stem flow areas respectively in in.² for radial channel j .

The propellant and support stem hydrogen reactivity feedbacks are then,

$$\$_{PROP} = (2.58) M_P^{EFF} [1 - (.0212) (M_P^{EFF} + M_{SS}^{EFF})] \text{ and}$$

$$\$_{SS} = (2.42) M_{SS}^{EFF} [1 - (.0212) (M_P^{EFF} + M_{SS}^{EFF})]$$

The corresponding fractional drum worth changes are,

$$F_{\theta}^{PROP} = -(.0565) M_P^{EFF} [1 - (.0867) (M_P^{EFF} + M_{SS}^{EFF})] \text{ and}$$

$$F_{\theta}^{SS} = -(.0531) M_{SS}^{EFF} [1 - (.0867) (M_P^{EFF} + M_{SS}^{EFF})]$$

2. Core Neutronic Temperature

The reactivity feedback is,

$$\$_{CORE TEMP} = \begin{cases} -(.141) \ln (T/532.8) [1 + (.0891) (\$_{PROP} + \$_{SS})] & \text{for } T \leq 532.8^{\circ}R \\ -(.394 \times 10^{-3}) \Delta T [1 - (.151 \times 10^{-3}) \Delta T] \times \\ [1 + (.0891) (\$_{PROP} + \$_{SS})] & \text{for } T \geq 532.8^{\circ}R \end{cases}$$

where T is the average fuel temperature in $^{\circ}R$ and $\Delta T \equiv T - 532.8$.

The corresponding fractional drum worth change is,

$$F_{\theta}^{CORE TEMP} = \begin{cases} 0, & \text{for } T \leq 532.8^{\circ}R \\ -(5.37 \times 10^{-5}) \Delta T [1 - (.227 \times 10^{-3}) \Delta T] \times \\ [1 + (.0891) (\$_{PROP} + \$_{SS})] & \text{for } T \geq 532.8^{\circ}R \end{cases}$$

3. Core Expansion

An effective filler strip radial displacement is obtained by weighting the stationwise displacements with the following normalized sine worth profile:

<u>Core Station (In.)</u>	<u>W_i</u>
Core Inlet	0
4	.378
8	.734
12	1.05
16	1.30
20	1.48
24	1.57
28	1.57
32	1.48
36	1.30
40	1.05
44	.734
48	.378
52	0
Average =	1

The effective radial displacement is then,

$$\Delta R_{\text{CORE}}^{\text{EFF}} = \frac{1}{52} \sum_{i=1}^{13} (W_i + W_{i+1}) (\Delta R_i^{\text{CORE}} + \Delta R_{i+1}^{\text{CORE}}) \text{ IN.}$$

The reactivity feedback is then,

$$\beta_{\text{CORE EXP}} = - (7.29) \Delta R_{\text{CORE}}^{\text{EFF}}$$

and the fractional drum worth change is,

$$F_{\theta}^{\text{CORE EXP}} = (.298) \Delta R_{\text{CORE}}^{\text{EFF}}$$

4. Cylinder Contraction

An effective support cylinder radial displacement is obtained by weighting the stationwise displacements with the normalized sine worth profile defined above.

The effective displacement is then,

$$\Delta R_{\text{CYL}}^{\text{EFF}} = \frac{1}{52} \sum_{i=1}^{13} (W_i + W_{i+1}) (\Delta R_i^{\text{CYL}} + \Delta R_{i+1}^{\text{CYL}}) \text{ IN.}$$

and the reactivity feedback is,

$$\$_{\text{CYL CONT.}} = - (1.87) \Delta R_{\text{CYL}}^{\text{EFF}}$$

The corresponding fractional drum worth change is,

$$F_{\theta}^{\text{CYL CONT.}} = - (.23) \Delta R_{\text{CYL}}^{\text{EFF}}$$

5. Seal and Filler Strip Hydrogen

The reactivity feedback is,

$$\$_{\text{SEAL}} = (.577) M_s [1 - (.132) M_s]$$

where M_s is the stored hydrogen mass content in the filler strip and seal regions.

There is no fractional drum worth change associated with seal and filler strip flow,

i.e.,

$$F_{\theta}^{\text{SEAL}} = 0$$

6. Cylinder Hydrogen

The reactivity feedback is,

$$\$_{\text{CYL}} = (.319) M_c [1 + (.366) M_c]$$

where M_c is the cylinder stored hydrogen mass content.

The corresponding fractional drums worth change is,

$$F_{\theta}^{\text{CYL}} = - (.0257) M_c [1 - (.582) M_c]$$

7. Reflector Hydrogen

The reactivity feedback is,

$$\$_{\text{REFL}} = - (.0126) M_R [1 - (2.22) M_R]$$

where M_R is the reflector stored hydrogen mass content.

There is no fractional drum worth change associated with reflector hydrogen

flow, i.e.,

$$F_{\theta}^{\text{REFL}} = 0.$$

8. Beryllium Temperature

The reactivity feedback is,

$$\beta_{\text{BE TEMP}} = \begin{cases} 0 & , \text{ for } T \leq 532.8^\circ\text{R} \\ - (.0579 \times 10^{-3}) \Delta T [1 + (2.21 \times 10^{-3}) \Delta T] & , \text{ for } T \geq 532.8^\circ\text{R} \end{cases}$$

where ΔT is the change in average beryllium temperature from 532.8°R .

The corresponding fractional drum worth change is,

$$F_{\theta}^{\text{BE TEMP}} = (7.30 \times 10^{-5}) \Delta T [1 - (.997 \times 10^{-3}) \Delta T]$$

In general, the system excess reactivity is,

$$\beta_{\text{EX}} = \beta_{\text{DRUMS}} (1 + F_{\theta}) + \beta_{\text{FEEDBACK}}$$

where,

β_{DRUMS} is the ambient drum reactivity from ambient critical determined from the equation, $\beta_{\text{DRUMS}} = 0.40 - (2.25) \cos \theta$.

F_{θ} is operating fractional drum worth change determined by,

$$F_{\theta} = \frac{(1 + F_{\theta}^{\text{PROP}}) (1 + F_{\theta}^{\text{SS}}) (1 + F_{\theta}^{\text{CORE TEMP}}) (1 + F_{\theta}^{\text{CORE EXP}}) \times (1 + F_{\theta}^{\text{CYL CONT.}}) (1 + F_{\theta}^{\text{CYL}}) (1 + F_{\theta}^{\text{BE TEMP}}) - 1}{1}$$

β_{FEEDBACK} is the net cold to hot reactivity feedback determined by,

$$\beta_{\text{FEEDBACK}} = \begin{cases} \beta_{\text{PROP}} + \beta_{\text{SS}} + \beta_{\text{CORE TEMP}} + \beta_{\text{CORE EXP}} + \\ \beta_{\text{CYL CONT.}} + \beta_{\text{SEAL}} + \beta_{\text{CYL}} + \beta_{\text{REFL}} + \\ \beta_{\text{BE TEMP}} \end{cases}$$

2.5.2.1.6 Nuclear Environmental Drawing

Table 2-66 summarizes the salient nuclear characteristics of the graphite design. These results are consistent with present TNT (Transient Nuclear Thermal) program analyses as discussed in Data Items S-031 and S-001.

Table 2-66 shows a \$5.35 ambient control drum span based on an 80% probability with 95% confidence level which compares with a required span of \$4.50 on the same basis. The support stem reactivity swing of \$2.23 (minimum SSCV setting at full power to maximum SSCV setting throttled) is consistent with throttled mode requirements after 10 hours full power operating lifetime at fixed operating drum bank angle, a \$0.10 temperature control margin and a -\$0.35 nominal start of life hot critical reactivity adjustment requirement. This swing does not accommodate a + \$0.35 hot critical adjustment requirement. The cold start of life critical drum bank angle of 80° is based on a \$4.50 total ambient drum span with a nominal cold critical shutdown margin of \$1.85 (\$1.70 minimum). The predicted span of \$5.35 could accommodate an additional shutdown margin of \$0.85. The operating critical drum bank angle of 58° results from a \$0.75 reactivity feedback to nominal rated start of life with a 6% operating drum worth reduction. Current analyses have indicated a different result of \$1.23 cold-to-hot feedback with a 10% operating drum worth reduction. The difference is primarily due to a smaller estimate of the core temperature reactivity coefficient to full power. The corresponding hot orificed drum bank angle would be 44° based on start of life cold critical at 83.5° (\$2.00 nominal cold critical shutdown). The increased cold critical shutdown margin is required to insure a minimum \$0.25 operating shutdown margin. The operating angle is shown as fixed throughout at a one hour operating lifetime for both the rated and throttled modes. The cold end of life critical bank angle of 82.5° is based on a \$0.10 operating reactivity loss via corrosion and fuel burnup and negligible xenon buildup. The space storage cold critical drum bank angle of 55° is based on a uniform reactor temperature of 50°R (\$1.00 reactivity insertion from ambient) with \$0.10 operating reactivity loss. The power distributions (max and min radial core power factors) shown are based on a fixed hot critical drum bank position throughout the operating lifetime in both the rated and throttled operating modes. The power distribution is unaffected

by rated full power operating but an 8% peripheral power factor reduction is associated with the throttled mode. This reduction suggests the desirability of compensating drum movement in the throttled mode. Such movement (approximately \$0.75) would relax the support stem reactivity requirement without imposing an additional requirement on control drum span.

2.5.2.2

Transient Analysis

A transient analysis was performed for the graphite core in the same manner as discussed in Section 2.5.1.2 for the composite core. The various phases of Startup and Shutdown are described there. The results of the TNT-11 analysis for the graphite core are shown in Figure 2-132 for an operating profile which assumes extreme thermal transients. For a step-by-step discussion of the various plots, the reader should consult the analogous plot in the aforementioned section. It suffices here to present the pertinent summary for the graphite core.

2.5.2.2.1

Summary of Drum Requirements During Transients

The maximum outward movement of the control drums during the Bootstrap phase with the graphite core is 96° at SOL and 123° at EOL with compensation for ~ 1.00 corrosion. The 70% provided in the drum span utilization for startup is sufficient for the Nuclear Startup and the Bootstrap phases. For the worst case startup, the drums must be able to compensate for a -25% due to maximum fission production buildup (at 10.5 hrs. after shutdown from one-hour of full power operation) and fuel burnup at the end of life. This reactivity includes the enhancement fission product worth when control drums are near 180° .

For the operating profile illustrated here, the minimum drum position during the Thrust Buildup phase is 49.5° at SOL and 40° at EOL. The drum reactivity which remains available for shutdown at these positions are approximately -79% and -52% , respectively. Thus, without consideration of uncertainties there is sufficient reactivity to assure at least -25% shutdown even with the peak support stem flow during Startup. At SOL, the required 25% shutdown margin is assured within the reference drum span utilization with an allowance for uncertainties in the cold-to-hot predictions. However, if flow control cannot compensate for uncertainties, an additional 18% may be required in the drum span to assure a 25% shutdown margin at EOL with the graphite core. However, this requirement must be reexamined when an optimal operation profile is defined.

During the Shutdown phase, the control drums are moved inward at both SOL and EOL to initiate Throttling. However, the drums return to the approximate Orifice Angle when equilibrium is established in the Throttle Hold. The drum angle plots shown in Figure 2-132 indicate that the drum position at the end of the Throttle Hold is approximately 58° , the hot critical or orifice angle. As discussed in Section 2.5.1.2, there is a peaking in power and feedback reactivity during the Temperature Retreat. Although the profile shown in Figure 2-132 shows a successful shutdown, no allowance was made for uncertainties in the cold-to-hot predictions in the model. In the profile illustrated here the control drums insert a reactivity of approximately $- \$1.00$ when rolled completely in from the hot critical angle. However, after compensating for uncertainties, the amount of drum which is available for shutdown may be less than $\$1.00$. Thus, to assure sufficient impressed reactivity for shutdown with uncertainties, an additional $15 \text{ } \mu$ may be necessary in the drum span utilization. However, this requirement must be reexamined when an optimal shutdown profile is defined.

In summary, the drum span utilization for the graphite core may require an addition of 15 to 18 μ to provide the required shutdown margin during maneuvering operations for some operating profiles. However, the reader should consult Data Items S-001 and S-031 for conclusions and recommendations on system performance. In particular, S-001 considers the system performance for other profiles as well as the profile described here.

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TABLE 2-1

COMPARISON OF FINAL EIGENVALUES USING THE DIRECT MACROSCOPIC
AND MICROSCOPIC MIXED METHODS OF CROSS SECTION GENERATION*

	Direct Macroscopic Method	Eigenvalue		Reactivity Difference
		Microscopic Mixed Method		
Ambient Base Case	1.0128069	1.0136787		+12.5¢
Hot (1600° K) Core Case	1.0028120	1.0036519		+12.0¢

* These are results for the PHOEBUS-2A reactor with control drums removed.

[REDACTED]
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TABLE 2-2
ILLUSTRATION OF PHOEBUS-2A REACTIVITY
FEEDBACK/DRUM INTERACTION EFFECTS

<u>Drum Bank Angle</u>	<u>Ambient Drum Bank Worth to Removal</u>	<u>Reactivity Effect of 4 lbs. of Reflector Hydrogen</u>
∞ (removed)	0	- \$0.45 (calculated)
91.2°	\$4.56	- \$0.12 (calculated)
136.5°	\$2.00	- \$0.30 (interpolated)*

$$* - \$0.30 = - \$0.45 + (- \$0.12 + \$0.45) \times \left(\frac{\$2.00}{\$4.56} \right)$$

$$\% \text{ Reduction in Drum Worth} = \frac{- \$0.12 + \$0.45}{\$4.56} = 7.2\%$$

[REDACTED]

TABLE 2-3
 R-1 MULTIZONED CORE MATERIAL INVENTORY - COMPOSITE, 15 REGIONS

REGION	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	TOTAL
1. Radil, cm	8.033	14.058	17.850	20.084	22.633	25.344	27.519	29.431	32.368	36.400	39.903	41.318	42.757	45.032	47.388	---
2. Volume, cc x 10 ³	26.776	55.228	50.206	35.164	45.181	53.970	47.708	45.183	75.313	115.051	110.910	47.688	50.201	82.872	90.350	931.801
3. Number of Fueled Elements	54	114	102	72	96	108	96	96	150	235.3	226.7	96	102	162	168	1878
4. Number of Unfueled Elements	10	18	18	12	12	21	18	12	30	39.7	38.3	18	18	36	48	349
5. Fuel Loading Code	11	12	13	14	15	16	17	18	19	20	21	23	22	21	16	---
6. Uranium wt, Kg	7.058	15.322	14.453	10.469	14.333	17.593	16.070	16.474	28.245	45.695	45.385	20.736	21.022	32.432	27.367	332.654
7. Carbon wt Fueled Matrix, Kg	19.818	41.838	37.434	26.424	35.232	39.636	35.232	35.232	55.050	86.355	83.199	35.232	37.434	59.454	61.656	689.226
8. Carbon wt Fueled Coatings, Kg	0.810	1.710	1.530	1.080	1.440	1.620	1.440	1.440	2.250	3.530	3.400	1.440	1.530	2.430	2.520	28.170
9. Carbon wt Unfueled Matrix, Kg	3.148	5.666	5.666	3.778	3.778	6.611	5.666	3.778	9.444	12.498	12.057	5.666	5.666	11.333	15.110	109.865
10. Carbon wt Unfueled Coatings, Kg	0.045	0.081	0.081	0.054	0.054	0.095	0.081	0.054	0.135	0.179	0.172	0.081	0.081	0.162	0.216	1.571
11. Carbon wt Pyro-Sleeve, Kg	1.140	2.052	2.052	1.368	1.368	2.394	2.052	1.368	3.421	4.527	4.367	2.052	2.052	4.105	5.473	39.791
12. Total Carbon wt, Kg	24.961	51.347	46.763	32.704	41.872	50.356	44.471	41.872	70.300	107.089	103.195	44.471	46.763	77.484	84.975	868.623
13. Zirconium wt Fueled Matrix, Kg	23.630	49.886	44.635	31.507	42.010	47.261	42.010	42.010	65.640	102.967	99.204	42.010	44.635	70.891	73.517	821.813
14. Zirconium wt Fueled Coating, Kg	6.210	13.110	11.730	8.280	11.040	12.420	11.040	11.040	17.250	27.060	26.070	11.040	11.730	18.630	19.320	215.970
15. Zirconium wt Unfueled Matrix, Kg	3.935	7.083	7.083	4.722	4.722	8.264	7.083	4.722	11.805	15.622	15.071	7.083	7.083	14.166	18.888	137.332
16. Zirconium wt Unfueled Coatings, Kg	0.330	0.594	0.594	0.396	0.396	0.693	0.594	0.396	0.990	1.310	1.264	0.594	0.594	1.188	1.584	11.517
17. Total Zirconium wt, Kg	34.105	70.673	64.042	44.905	58.168	68.638	60.727	58.168	95.685	146.959	141.609	60.727	64.042	104.875	113.309	1186.632

TABLE 2-3 (CONT'D)

REGION	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	TOTAL
18. Mn wt, Kg*	0.039	0.070	0.070	0.046	0.046	0.081	0.070	0.046	0.116	0.154	0.148	0.070	0.070	0.138	0.184	1.348
19. Cr wt, Kg	0.365	0.656	0.656	0.438	0.438	0.766	0.656	0.438	1.094	1.448	1.397	0.656	0.656	1.314	1.752	12.730
20. Ni wt, Kg	0.579	1.041	1.041	0.694	0.694	1.215	1.041	0.694	1.735	2.297	2.216	1.041	1.041	2.082	2.776	20.187
21. Mo wt, Kg	0.029	0.051	0.051	0.034	0.034	0.060	0.051	0.034	0.085	0.113	0.109	0.051	0.051	0.102	0.136	0.991
22. Ti wt, Kg	0.047	0.084	0.084	0.056	0.056	0.098	0.084	0.056	0.140	0.185	0.178	0.084	0.084	0.168	0.224	1.628
23. Fe wt, Kg	1.321	2.377	2.377	1.585	1.585	2.773	2.377	1.585	3.962	5.243	5.057	2.377	2.377	4.755	6.340	46.091
24. Ta wt, Kg**	0.157	0.283	0.283	0.188	0.188	0.330	0.283	0.188	0.472	0.624	0.602	0.283	0.283	0.566	0.754	5.484

* Items 18-24 represent the shim tube, support stem and liner tube in the central elements.

**Tantalum weight corresponds to 5.5 mils of TaC on 1/3 of the total shim tubes distributed uniformly throughout the core.

TABLE 2-4

R-1 AMBIENT TEMPERATURE CORE ATOM DENSITIES - COMPOSITE, 15 REGIONS

REGION	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
<u>Material</u>															
Nu235x10 ⁻³ *	0.6292	0.6622	0.6872	0.7107	0.7572	0.7781	0.8040	0.8703	0.8952	0.9480	0.9768	1.0379	0.9996	0.9342	0.7230
Nu238x10 ⁻⁴	0.457	0.481	0.499	0.516	0.550	0.565	0.584	0.632	0.650	0.688	0.709	0.754	0.726	0.678	0.525
Ncx10 ⁻²	4.6761	4.6636	4.6722	4.6652	4.6488	4.6802	4.6758	4.6486	4.6823	4.6690	4.6672	4.6778	4.6726	4.6900	4.7177
NZr90x10 ⁻³	4.332	4.353	4.339	4.344	4.379	4.326	4.330	4.379	4.321	4.345	4.343	4.331	4.339	4.304	4.266
NZr91x10 ⁻³	0.942	0.947	0.944	0.945	0.952	0.941	0.942	0.952	0.940	0.945	0.944	0.942	0.944	0.936	0.928
NZr92x10 ⁻³	1.438	1.445	1.441	1.442	1.454	1.436	1.438	1.454	1.435	1.443	1.442	1.438	1.441	1.429	1.416
NZr94x10 ⁻³	1.464	1.471	1.466	1.468	1.480	1.462	1.463	1.480	1.460	1.468	1.467	1.463	1.466	1.454	1.441
NZr96x10 ⁻³	0.236	0.237	0.236	0.236	0.238	0.235	0.235	0.238	0.235	0.236	0.236	0.235	0.236	0.234	0.232
NMnx10 ⁻⁴	0.15	0.13	0.15	0.14	0.11	0.16	0.16	0.11	0.16	0.14	0.14	0.16	0.15	0.18	0.22
NCrx10 ⁻³	0.157	0.137	0.151	0.144	0.112	0.164	0.159	0.112	0.168	0.145	0.145	0.159	0.151	0.183	0.224
NNix10 ⁻³	0.221	0.193	0.212	0.202	0.157	0.231	0.223	0.157	0.236	0.204	0.205	0.224	0.212	0.257	0.315
NM095x10 ⁻⁵	0.1														
NM096x10 ⁻⁵	0.1														
NM097x10 ⁻⁵	0.1														
NM098x10 ⁻⁵	0.3														
NM0100x10 ⁻⁵	0.1														
NTix10 ⁻⁴	0.22	0.19	0.21	0.20	0.16	0.23	0.22	0.16	0.23	0.20	0.20	0.22	0.21	0.25	0.31
NFex10 ⁻³	0.532	0.464	0.510	0.486	0.378	0.554	0.537	0.378	0.567	0.491	0.491	0.537	0.510	0.618	0.757
NTox10 ⁻⁴	0.100														

* For example, Nu235 in region 1 = 0.0006292.

TABLE 2-5

R-1 MULTIZONED CORE MATERIAL INVENTORY - GRAPHITE, 15 REGIONS

Region	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	TOTAL
1. Radli, cm	8.033	14.058	17.850	20.084	22.633	25.344	27.519	29.431	32.368	36.400	39.903	41.318	42.757	45.032	47.388	---
2. Volume, ccx10 ³	26.585	54.835	49.849	34.914	44.860	53.586	47.369	44.861	74.777	114.232	110.121	47.349	49.844	82.282	89.707	925.171
3. Number of Fueled Elements	54	114	102	72	96	108	96	96	150	235.3	226.7	96	102	162	168	1878
4. Number of Unfueled Elements	10	18	18	12	12	21	18	12	30	39.7	38.3	18	18	36	48	349
5. Fuel Loading Code	11	12	13	14	15	16	17	18	19	20	21	23	22	21	16	---
6. Uranium wt, Kg	5.821	12.677	11.924	8.662	11.875	14.558	13.286	13.670	23.400	37.860	37.564	17.155	17.360	26.843	22.646	275.301
7. Carbon wt. Fueled Element, Kg (Matrix + Coating)	28.739	60.625	54.182	38.210	50.899	57.100	50.707	50.650	78.840	123.533	118.836	50.131	53.397	84.920	88.822	989.591
8. Carbon wt. Unfueled Element, Kg (Matrix)	4.341	7.814	7.814	5.209	5.209	9.116	7.814	5.209	13.023	17.234	16.626	7.814	7.814	15.628	20.837	151.501
9. Carbon wt. Pyro-Sleeve, Kg	1.132	2.037	2.037	1.358	1.358	2.377	2.037	1.358	3.397	4.495	4.336	2.037	2.037	4.076	5.434	39.506
10. Total Carbon wt., Kg	34.212	70.476	64.033	44.777	57.466	68.593	60.558	57.217	95.260	145.262	139.798	59.982	63.248	104.624	115.093	1180.598
11. Zirconium wt. Fueled Element, Kg	3.969	8.379	7.497	5.292	7.056	7.938	7.056	7.056	11.025	17.295	16.662	7.056	7.497	11.907	12.348	138.033
12. Zirconium wt. Unfueled Element, Kg	0.305	0.549	0.549	0.366	0.366	0.641	0.549	0.366	0.915	1.211	1.168	0.549	0.549	1.098	1.464	10.645
13. Total Zirconium wt. Kg	4.274	8.928	8.046	5.658	7.422	8.579	7.605	7.422	11.940	18.506	17.830	7.605	8.046	13.005	13.812	148.678
14. Total Niobium wt., Kg	2.306	4.868	4.355	3.074	4.099	4.612	4.099	4.099	6.405	10.047	9.680	4.099	4.355	6.917	7.174	80.189

Weights for shim tubes, support stems and liner tubes in the central elements are the same as those given in DRM 52772 corrected for fueled length (51.63/52.00).

TABLE 2-6 (CRD)

R-1 AMBIENT TEMPERATURE CORE ATOM DENSITIES - GRAPHITE, 15 REGIONS (U)

Region	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
<u>Material</u>															
$N_{U235} \times 10^{-3}$ *	0.5227	0.5518	0.5710	0.5922	0.6319	0.6485	0.6695	0.7274	0.7470	0.7911	0.8142	0.8648	0.8314	0.7787	0.6026
$N_{U238} \times 10^{-4}$	0.379	0.401	0.415	0.430	0.459	0.471	0.486	0.528	0.542	0.574	0.591	0.628	0.604	0.565	0.437
$N_{C} \times 10^{-2}$	6.4553	6.4470	6.4435	6.4332	6.4258	6.4210	6.4129	6.3978	6.3902	6.3788	6.3680	6.3546	6.3652	6.3782	6.4357
$N_{Zr 90} \times 10^{-4}$	5.47	5.54	5.49	5.51	5.63	5.44	5.46	5.63	5.43	5.51	5.51	5.46	5.49	5.38	5.24
$N_{Zr 91} \times 10^{-4}$	1.19	1.20	1.19	1.20	1.22	1.18	1.19	1.22	1.18	1.20	1.20	1.19	1.19	1.17	1.14
$N_{Zr 92} \times 10^{-4}$	1.82	1.84	1.82	1.83	1.87	1.81	1.81	1.87	1.80	1.83	1.83	1.81	1.82	1.79	1.74
$N_{Zr 94} \times 10^{-4}$	1.85	1.87	1.85	1.86	1.90	1.84	1.84	1.90	1.83	1.86	1.86	1.84	1.85	1.82	1.77
$N_{Zr 96} \times 10^{-4}$	0.30	0.30	0.30	0.30	0.31	0.30	0.30	0.31	0.30	0.30	0.30	0.30	0.30	0.29	0.28
$N_{Nb} \times 10^{-4}$	5.625	5.747	5.666	5.710	5.926	5.581	5.612	5.925	5.555	5.704	5.701	5.614	5.666	5.452	5.186
$N_{Mn} \times 10^{-4}$	0.15	0.13	0.15	0.14	0.11	0.16	0.16	0.11	0.16	0.14	0.14	0.16	0.15	0.18	0.22
$N_{Cr} \times 10^{-3}$	0.157	0.137	0.151	0.144	0.112	0.164	0.159	0.112	0.168	0.145	0.145	0.159	0.151	0.183	0.224
$N_{Ni} \times 10^{-3}$	0.221	0.193	0.212	0.202	0.157	0.231	0.223	0.157	0.236	0.204	0.205	0.224	0.212	0.257	0.315
$N_{Mo 95} \times 10^{-5}$	0.1														
$N_{Mo 96} \times 10^{-5}$	0.1														
$N_{Mo 97} \times 10^{-5}$	0.1														
$N_{Mo 98} \times 10^{-5}$	0.3														
$N_{Mo 100} \times 10^{-5}$	0.1														
$N_{Ti} \times 10^{-4}$	0.22	0.19	0.21	0.20	0.16	0.23	0.22	0.16	0.23	0.20	0.20	0.22	0.21	0.25	0.31
$N_{Fe} \times 10^{-3}$	0.532	0.464	0.510	0.486	0.378	0.554	0.537	0.378	0.567	0.491	0.491	0.537	0.510	0.618	0.757
$N_{Ta} \times 10^{-4}$	0.100														

* For example, N_{U235} in region 1 = 0.0005227.

TABLE 2-7
 PRELIMINARY R-1 RADIAL NON-CORE REGIONS MATERIAL INVENTORY

REGION	Material Weights, Kg									
	Be	C	AL	Fe	Ni	Cr	Mo	Zr	Mn	Ti
1. Non-Corroding Periphery Region No. 16*	---	115.33	---	---	---	---	7.83	93.80	---	---
2. Lateral Support Region No. 17	---	178.59	---	---	---	---	---	---	---	---
3. Aluminum Cylinder Region No. 18	---	---	261.95	---	---	---	---	---	---	---
4. Plunger -1 Region No. 19	---	3.92	39.07	12.72	5.87	3.48	0.29	---	0.37	0.49
5. Plunger -2 Region No. 24	55.24	10.58	17.54	---	---	---	---	---	---	---
6. Spring Region No. 20	125.26	---	46.42	81.56	37.69	22.34	1.86	---	2.38	3.13
7. Beryllium Region No. 21	591.70	3.75	84.27	---	---	---	---	---	---	---
8. Control Vane Region No. 23	---	---	---	---	---	---	---	---	---	---
TOTALS:	772.20	312.17	449.25	94.28	43.56	25.82	9.98	93.80	2.75	3.62

* Reactivity Weighted Masses Due to Non-Uniformity in the Axial Direction.

TABLE 2-8

PRELIMINARY R-1 RADIAL NON-CORE REGIONS NUMBER DENSITIES

REGION	NUMBER DENSITIES																	
	$\left(\frac{\mu}{m}\right)$																	
	$\left(\frac{\mu}{p}\right)$																	
Be	C	AL	Fe	Ni	Cr	M ₀₉₅	M ₀₉₆	M ₀₉₇	M ₀₉₈	M ₀₁₀₀	Z _{r90}	Z _{r91}	Z _{r92}	Z _{r94}	Z _{r96}	M _n	Ti	
1. Non-Corrodible Periphery-Region 16	---	0.069206	---	---	---	---	0.000092 (4069.43) (4074.43)	0.000097 (3859.41) (3864.41)	0.000056 (6688.71) (6693.71)	0.000286 (1305.66) (1310.66)	0.000057 (6571.28) (6576.28)	0.003817 (92.00) (98.20)	0.000830 (445.42) (451.62)	0.001267 (289.65) (295.85)	0.001290 (284.38) (290.58)	0.000207 (1804.66) (1810.86)	---	---
2. Lateral Support Region 17	---	0.050289	---	---	---	---	---	---	---	---	---	---	---	---	---	---	---	---
3. Aluminum Cylinder Region 18	---	---	0.053936	---	---	---	---	---	---	---	---	---	---	---	---	---	---	---
4. Plunger-1 Region 19	---	0.010274	0.045586	0.007169	0.003147	0.002106	0.000015 (16964.3) (16969.3)	0.000016 (15903.7) (15908.7)	0.000009 (28277.1) (28282.1)	0.000046 (5538.5) (5543.5)	0.000009 (15903.7) (15908.7)	---	---	---	---	0.000211	0.000322	
5. Plunger-2 Region 24	0.055745	0.008011	0.005913	---	---	---	---	---	---	---	---	---	---	---	---	---	---	---
6. Spring Region 20	0.055686	---	0.006894	0.005851	0.002572	0.001721	0.000012 (38838.0) (38843.0)	0.000013 (35850.1) (35855.1)	0.000007 (66583.0) (66588.0)	0.000038 (12261.2) (12266.2)	0.000007 (66583.0) (66588.0)	---	---	---	---	0.000173	0.000261	
7. Beryllium Region 21	0.098326	0.000467	0.004678	---	---	---	---	---	---	---	---	---	---	---	---	---	---	---
8. Control Vane - Region 23																		
N ^{B10} = 0.023720																		
N ^{Cu} = 0.066210																		

TABLE 2-9.

R-1 COMPOSITE CORE MATERIAL INVENTORY - 5 REGIONS

<u>MATERIAL</u>	CORE REGION				
	1	2	3	4	5
URANIUM, kg	111.772	73.940	66.121	53.454	27.367
CARBON, kg	334.346	177.389	147.666	124.247	84.975
ZIRCONIUM, kg	459.426	242.644	202.336	168.917	113.309
A-286, kg	26.822	15.451	12.480	11.970	10.641
SS-304, kg	1.944	1.120	0.904	0.867	0.771
TANTALUM, kg	1.900	1.096	0.885	0.849	0.754

TABLE 2-10

R-1 COMPOSITE CORE NUMBER DENSITIES* - 5 REGIONS

MATERIAL	CORE REGION				
	1	2	3	4	5
OUTER RADIUS, cm	29.431	36.400	41.318	45.032	47.388
VOLUME, ccx10 ³	359.416	190.364	158.598	133.073	90.350
N _U **	0.0007975	0.0009961	0.0010691	0.0010301	0.0007755
N _C	0.046663	0.046743	0.046705	0.046835	0.0471770
N _{Zr}	0.0084428	0.0084188	0.0084264	0.0083840	0.0082833
N _{A-286}	0.0008136	0.0008848	0.0008578	0.0009806	0.0012839
N _{SS-304}	0.0000595	0.0000648	0.0000627	0.0000717	0.0000939
N _{Ta}	0.0000176	0.0000192	0.0000186	0.0000212	0.0000278

* The volumes and atom densities are based on a total fuel length of 52.0 inches.

** Includes both U-235 and U-238 atom densities.

TABLE 2-11

R-1 RADIAL NON-CORE REGION MATERIAL INVENTORY,
ONE-DIMENSIONAL

<u>REGION*</u>	<u>MATERIAL (kg)</u>					
	<u>Be</u>	<u>C</u>	<u>AL</u>	<u>Mo</u>	<u>Zr</u>	<u>A-286</u>
1. Periphery		115.33		7.83	93.80	
2. Lateral support		178.59				
3. Aluminum Cylinder + Plunger 1		3.92	261.95 39.07			23.22
4. Beryllium 1 + Plunger 2	180.88 55.24	10.58	25.76 17.54			
5. Beryllium 2 + Spring	410.82 125.26	3.75	58.51 46.42			148.96
TOTAL	772.20	312.17	449.25	7.83	93.80	172.18

* Region terminology is presented in Section 2, 4. 2.

TABLE 2-12
R-1 RADIAL NON-CORE REGION NUMBER DENSITIES, ONE-DIMENSIONAL

<u>REGION</u>	<u>Be</u>	<u>C</u>	<u>AL</u>	<u>Mo</u>	<u>Zr</u>	<u>A-286</u>
<u>Periphery</u> Outer Radius=49.468 cm Region Volume=83.594x10 ³ cc		0.069206		0.000588	0.007411	
<u>Lateral Support</u> Outer Radius=53.632 cm Region Volume=178.138x10 ³ cc		0.050289				
<u>Aluminum Cylinder</u> Outer Radius=56.426 cm Region Volume=127.595x10 ³ cc		0.001541	0.052684			0.0019839
<u>Beryllium 1</u> Outer Radius=60.332 cm Region Volume=189.24x10 ³ cc	0.083409	0.0028045	0.0051097			
<u>Beryllium 2</u> Outer Radius=68.377 cm Region Volume=429.66x10 ³ cc	0.083405	0.0004378	0.0054537			0.0037795

TABLE 2-13
R-1 GRAPHITE CORE MATERIAL INVENTORY* - 5 REGIONS

MATERIAL	CORE REGION				
	1	2	3	4	5
URANIUM, kg	92.473	61.260	54.719	44.203	22.646
CARBON, kg	457.332	240.522	199.78	167.872	115.093
ZIRCONIUM, kg	57.934	30.446	25.435	21.051	13.812
NIOBIUM, kg	31.512	16.452	13.779	11.272	7.174
A-286, kg	26.822	15.451	12.480	11.970	10.641
SS-304, kg	1.944	1.120	0.904	0.867	0.771
TANTALUM, kg	1.900	1.096	0.885	0.849	0.754

* The masses listed are for 51.63 inches total fuel length.

TABLE 2-14
R-1 GRAPHITE CORE NUMBER DENSITIES* - 5 REGIONS

MATERIAL	CORE REGION				
	1	2	3	4	5
OUTER RADIUS, cm	29.431	36.4	41.318	45.032	47.388
VOLUME, ccx10 ³	356.859	189.009	157.469	132.126	89.707
N _U ^{**}	0.0006645	0.0008311	0.0008911	0.0008579	0.0006474
N _C	0.064285	0.063833	0.063641	0.063733	0.064357
N _{Zr}	0.0010722	0.0010640	0.0010669	0.0010523	0.0010169
N _{Nb}	0.0005727	0.0005644	0.0005674	0.0005532	0.0005186
N _{A-286}	0.0008136	0.0008848	0.0008578	0.0009806	0.0012839
N _{SS-304}	0.0000595	0.0000648	0.0000627	0.0000717	0.0000939
N _{Ta}	0.0000176	0.0000192	0.0000186	0.0000212	0.0000278

* The volumes and atom densities in this model are based on a total fuel length of 51.63 inches.

** Includes both U-235 and U-238 atom densities.

TABLE 2-15
COMPOSITE CORE RADIAL DISPLACEMENTS

<u>Core Station (in)</u>		<u>Rated Full Power (SOL)</u>	<u>Startup Hold*</u>
(inlet)	0	- .02 in.	- .01 in.
	5	.05 in.	.02 in.
	10	.10 in.	.05 in.
	15	.14 in.	.07 in.
	20	.17 in.	.09 in.
	25	.20 in.	.11 in.
	30	.24 in.	.13 in.
	35	.26 in.	.14 in.
	40	.28 in.	.15 in.
	45	.29 in.	.15 in.
	50	.29 in.	.14 in.

* 40 seconds into the SOL startup (2520° R chamber temperature)

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TABLE 2-16

ALUMINUM BARREL RADIAL DISPLACEMENTS - COMPOSITE CORE

	<u>Core Station (in.)</u>	<u>Rated Full Power (SOL)</u>	<u>Startup Hold*</u>
(inlet)	2	- .064 in.	- .071 in.
	4	- .072 in.	- .074 in.
	8	- .075 in.	- .076 in.
	12	- .075 in.	- .076 in.
	16	- .075 in.	- .077 in.
	20	- .076 in.	- .078 in.
	24	- .077 in.	- .080 in.
	28	- .080 in.	- .083 in.
	32	- .082 in.	- .085 in.
	36	- .085 in.	- .087 in.
	40	- .088 in.	- .089 in.
	44	- .092 in.	- .092 in.
	48	- .096 in.	- .094 in.
	50	- .084 in.	- .090 in.
	52	- .071 in.	- .086 in.

* 40 seconds into the SOL startup (2520° R chamber temperature)

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TABLE 2-17

GRAPHITE CORE RADIAL DISPLACEMENTS

<u>Core Station (in.)</u>		<u>Rated Full Power (SOL)</u>	<u>Startup Hold*</u>
(inlet)	0	- .01 in.	- .01 in.
	5	.04 in.	.02 in.
	10	.07 in.	.03 in.
	15	.10 in.	.05 in.
	20	.13 in.	.07 in.
	25	.16 in.	.08 in.
	30	.19 in.	.10 in.
	35	.22 in.	.11 in.
	40	.24 in.	.11 in.
	45	.24 in.	.11 in.
	50	.25 in.	.10 in.

* 39.5 seconds into the SOL startup (2433° R chamber temperature)

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TABLE 2-18

ALUMINUM BARREL RADIAL DISPLACEMENTS - GRAPHITE CORE

<u>Core Station (in.)</u>	<u>Rated Full Power (SOL)</u>	<u>Startup Hold *</u>
(inlet) 2	- .061 in.	- .069 in.
4	- .069 in.	- .072 in.
8	- .072 in.	- .073 in.
12	- .071 in.	- .074 in.
16	- .071 in.	- .074 in.
20	- .072 in.	- .076 in.
24	- .074 in.	- .078 in.
28	- .076 in.	- .081 in.
32	- .079 in.	- .083 in.
36	- .083 in.	- .086 in.
40	- .086 in.	- .088 in.
44	- .090 in.	- .091 in.
48	- .095 in.	- .094 in.
50	- .083 in.	- .090 in.
52	- .071 in.	- .087 in.

* 39.5 seconds into the SOL startup (2433° R chamber temperature)

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TABLE 2-19

BASIC DESIGN PARAMETERS AND ASSUMPTIONS - COMPOSITE FUEL

1. Reference Fuel Element Drawing = 940J443
2. Reference Unfueled Element Drawing = 940J439
3. Reference Periphery Drawing = 939J728
4. Reference Lateral Support Drawing = 939J733
5. Reference Reflector Drawing = 939J733
6. All region and material weights and volumes based on a 52.0 inch length.
7. Use the unfueled to fueled element ratio documented in DRM-52085. This element breakup was rederived and verified.
8. Use NCI's developed for PAX-G2A which were documented in NRD-70-190.
9. Total carbon weight in the unfueled element equals 319.3 gms, (314.8 gms matrix, 4.5 gms coating).
10. Total zirconium weight in the unfueled element equals 426.5 gms. (393.5 gms matrix, 33.0 gms coating).
11. Fueled carbon density = 1.3 gms/cc.
12. Fueled zirconium density = 1.55 gms/cc.
13. 0.102 inch reamed channel diameter, 19 coolant holes.
14. Fueled element material volume = 282.3 cc.
15. R-1 has 45 instrumented centrals and 304 non-instrumented centrals. Each of the 45 instrumented centrals have 2-100 mil full length holes, for a total of 90 holes. Of these 90 holes, 45 contain thermocapsules with the rest of the holes plugged. The remaining 45 holes have thermocouples at an average axial station of 22.8" with the bottom of the holes plugged. The difference in carbon mass of assuming all 349 centrals are non-instrumentated relative to the current instrumentation plan is only +0.182 kg. Thus, this model assumes all 349 centrals are non-instrumentated.
16. Unfueled element material volume = 224.86 cc.
17. Using homogeneous cross sections and reactivity worth, a - \$1.5 worth of tantalum is smeared uniformly over the entire core regions.

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TABLE 2-19 (CONT'D.)

18. The zirconium bore and external coatings were not reactivity weighted due to the small reactivity worth of zirconium and the shapes of the reference coating profiles.
19. The control vane material, $B^{10}Cu$, has a B^{10} surface density of 0.100 gm/cm^2 .

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TABLE 2-20
CHEMICAL COMPOSITIONS A-286 AND SS-304

	<u>A-286</u>	<u>SS-304</u>
<u>Material</u>	<u>w/o</u>	<u>w/o</u>
Manganese	1.6	2.0
Chromium	15.0	20.0
Nickel	25.3	11.0
Molybdenum	1.25	0.5
Titanium	2.1	---
Iron	54.75	66.5

$$\rho = 0.286 \text{ lbs/in}^3$$

$$\rho = 0.287 \text{ lbs/in}^3$$

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TABLE 2-21
BASIC DESIGN PARAMETERS AND ASSUMPTIONS - GRAPHITE FUEL

1. Reference Fuel Element Drawing = 711J780
2. Reference Unfueled Element Drawing = 711J781
3. Core weights and volumes based on a 51.63 inch length. All other region weights and volumes developed for a 52.0 inch length.
4. Fueled element material volume = 283.7 cc, DWG. 711J780.
5. No carbon addition during coating of unfueled elements. (HED PROCESS).
No carbon addition during ZrC coatings of fueled elements (HED PROCESS). Assume a 4.0 gm carbon addition per element from the 41.0 gm Nb bore coatings (GEM PROCESS). Neglect carbon addition from Nb exterior coatings of the fuel elements due to small amount ($\sim .3$ gm/element) and axial distribution (40-52 inches).
6. Figure 2-9 shows the variation in the fueled element carbon weight versus uranium loading. This curve was obtained from the three data points (weights) of DWG. 711J780.
7. Coating masses per element given on reference fueled and unfueled drawings apply over entire element length, 52.0 and 52.58 inches respectively. These have been ratioed down to a 51.63 inch length. Note that the unfueled element graphite length is 51.38 inches; however, it is being treated as a 51.63 inch length.
8. ZrC bore and external and NbC bore coatings were not axially weighted due to small neutronic worth and axial profiles of Zr and axial profile of Nb. NbC external coating was reduced from 3.0 gm/element to 2.0 gm/element due to axial profile (40-52 inches).
9. Unfueled element material volume
= 227.3 cc (51.63 inches).
10. Assumed unfueled element carbon density
= 1.91 gm/cc, consistent with Figure 2-9.
11. Unfueled carbon weight = 434.1 gm.
12. Zirconium weight in unfueled element
= 30.5 gm (12.8 gm bore + 17.7 gm external coating).

TABLE 2-22
NEUTRON CROSS SECTION DATA TAPES EMPLOYED
IN THE REFLECTOR ASSEMBLY ANALYSIS

<u>Cross Section Tape Identification</u>	<u>Description</u>
FSGAM28, RD1, RD2.	GAM LIBRARY TAPE
CGBIT15.	BIT LIBRARY TAPE
GWORY1, RD1, RD2.	QUERY LIBRARY TAPE
CGGAS15.	GAS LIBRARY TAPE

TABLE 2-23

NUCLEAR CROSS SECTION DATA TAPES EMPLOYED
IN THE NUCLEAR SUBSYSTEM ANALYSIS

<u>Cross Section Tape Identification</u>	<u>Description</u>
FSGAM29, RD1, RD2	GAM Library Tape
CGBIT15	BIT Library Tape
GWQRY1, RD1, RD2	QUERY Library Tape
CGGAS15	GAS Library Tape

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TABLE 2-24
COMPARISON OF CALCULATED AND MEASURED ABSOLUTE REACTIVITY

Reactor	Calculated Reactivity, \$ WISDM	Corrected*	Measured*** Reactivity, \$	WISDM-Measured Difference, \$
PAX-GO	-0.086**	+0.694	0	0.694
PAX-G1(R-1 Test Sector)	-1.289	-0.509	-1.65	1.141
PAX-G2 (R-1 Test Sector)	-0.205	+0.575	-1.65	2.225
PAX-G3 (R-1 Test Sector)	+0.386	+0.926	-1.24	2.166

* Using corrections from Table 2-32.

** Calculated from Table 2-31 by weighting the KIWI Sector reactivity with 5/6 and the R-1 test sector by 1/6.

***6x worth of replacing 60° sector of PAX-G0 by test sector.

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TABLE 2-25

SUMMARY OF PAX-G2 EXPERIMENTAL
AND PREDICTED CHANNEL-10 FISSION/GRAM DATA

<u>Element Location</u>	<u>Radius, cm</u>	<u>Fissions per Gram</u>	
		<u>Predicted</u>	<u>Measurement</u>
00D-10	1.914	0.99949	1.00000
A1F-10	5.065	0.98883	0.99400
G	3.316	0.99643	0.99947
B	3.829	0.99464	0.99822
C	5.743	0.98456	0.99102
D	6.902	0.97593	0.98468
E	6.632	0.97806	0.98629
B1F-10	9.947	0.94132	0.96105
G	8.345	0.95731	0.97477
B	8.773	0.95320	0.97137
C	10.659	0.93376	0.95414
D	11.955	0.91885	0.94019
E	11.645	0.92256	0.94378
B2F-10	9.572	0.94522	0.96453
G	7.657	0.96964	0.97973
B	6.902	0.97593	0.98468
C	8.345	0.95731	0.97477
D	10.130	0.93940	0.94935
E	10.659	0.93376	0.95414
C1F-10	14.952	0.87738	0.90219
G	13.401	0.90033	0.92286
B	13.805	0.89487	0.91772
C	15.670	0.86681	0.89195
D	17.016	0.84608	0.87175
E	16.689	0.85122	0.87680
C2F-10	13.805	0.89487	0.91772

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TABLE 2-25 (CONT'D)

<u>Element Location</u>	<u>Radius, cm</u>	<u>Fissions per Gram</u>	
		<u>Predicted</u>	<u>Measurement</u>
C2G-10	11.955	0.91885	0.94019
B	11.645	0.92256	0.94378
C	13.263	0.90216	0.92463
D	14.952	0.87738	0.90219
E	15.195	0.87384	0.89882
C3F-10	14.453	0.88451	0.90908
G	12.554	0.91138	0.93332
B	11.486	0.92442	0.94542
C	12.554	0.91138	0.93332
D	14.453	0.88451	0.90908
E	15.315	0.87208	0.89698
D1F-10	19.987	0.79273	0.82373
G	18.462	0.81914	0.84899
B	18.855	0.81247	0.84264
C	20.707	0.77631	0.81157
D	22.078	0.75194	0.78812
E	21.743	0.75790	0.79396
D2F-10	18.462	0.81914	0.84899
G	16.689	0.85122	0.87680
B	16.579	0.85294	0.87847
C	18.262	0.82249	0.85222
D	19.895	0.79434	0.82541
E	19.987	0.79273	0.82373
D3F-10	18.262	0.82249	0.85222
G	16.357	0.85639	0.88178
D3B-10	15.670	0.86681	0.89195
C	17.015	0.84608	0.87175
D	18.855	0.81247	0.84264
E	19.429	0.80250	0.83308

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TABLE 2-25 (CONT'D)

Element Location	Radius, cm	Fissions per Gram	
		Predicted	Measured
D4F-10	19.429	0.80250	0.83308
G	17.546	0.83755	0.86351
B	16.357	0.85639	0.88178
C	17.230	0.84266	0.86850
D	19.144	0.80747	0.83788
E	20.169	0.78577	0.82071
E1F-10	25.034	0.69794	0.73762
G	23.524	0.72339	0.76333
B	23.911	0.71684	0.75665
C	25.756	0.68166	0.72543
D	27.141	0.66018	0.70308
E	26.802	0.66531	0.70848
E2F-10	23.290	0.72739	0.76729
G	21.574	0.76093	0.79688
B	21.574	0.76093	0.79688
C	23.290	0.72739	0.76729
D	24.887	0.70038	0.73998
E	24.887	0.70038	0.73998
E3F-10	22.570	0.74327	0.77968
G	20.707	0.77631	0.81157
B	20.260	0.78418	0.81919
C	21.743	0.75790	0.79396
D	23.524	0.72339	0.76333
E	23.911	0.71684	0.75665
E4F-10	22.973	0.73281	0.77279
G	21.058	0.77012	0.80561
B	20.169	0.78577	0.82071
C	21.318	0.76553	0.80116
D	23.211	0.72873	0.76866
E	23.987	0.71554	0.75528

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TABLE 2-25 (CONT'D)

<u>Element Location</u>	<u>Radius, cm</u>	<u>Fissions per Gram</u>	
		<u>Predicted</u>	<u>Measured</u>
E5F-10	24.441	0.70784	0.74760
G	22.570	0.74327	0.77968
B	21.318	0.76553	0.80116
C	22.078	0.75195	0.78812
D	23.987	0.71554	0.75528
E	25.107	0.69673	0.73627
F1F-10	30.087	0.61577	0.66011
G	28.588	0.63878	0.68096
B	28.970	0.63346	0.67546
C	30.809	0.60657	0.65101
D	32.205	0.58986	0.63533
E	31.862	0.59373	0.63898
F2F-10	28.201	0.64427	0.68675
G	26.527	0.66951	0.71283
B	26.596	0.66846	0.71170
C	28.330	0.64242	0.68480
D	29.904	0.61817	0.66262
E	29.842	0.61898	0.66342
F3F-10	27.141	0.66018	0.70308
G	25.325	0.69315	0.73258
B	25.034	0.69794	0.73726
C	26.595	0.66846	0.71170
D	28.330	0.64242	0.68480
E	28.588	0.63878	0.68096
F4F-10	27.006	0.66222	0.70513
G	25.107	0.69673	0.73627
B	24.441	0.70784	0.74760
C	25.756	0.68166	0.72543
D	27.610	0.65284	0.69573
E	28.201	0.64427	0.68675
F5F-10	27.808	0.64994	0.69265
G	25.897	0.67939	0.72311
B	24.887	0.70038	0.73998
C	25.897	0.67939	0.72311
D	27.808	0.64994	0.69265
E	28.716	0.63698	0.67907

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TABLE 2-25 (CONT'D)

<u>Element Location</u>	<u>Radius, cm</u>	<u>Fissions per Gram</u>	
		<u>Predicted</u>	<u>Measured</u>
F6F-10	29.472	0.62393	0.66844
G	27.610	0.65284	0.69573
B	26.319	0.67274	0.71623
C	27.006	0.66222	0.70513
D	28.907	0.63433	0.67632
E	30.087	0.61577	0.66011
G1F-10	35.144	0.56214	0.60730
G	33.652	0.57340	0.62480
B	34.031	0.57018	0.61720
C	35.866	0.55853	0.60790
D	37.269	0.55658	0.60470
E	36.924	0.55687	0.60250
G2F-10	33.158	0.57802	0.62645
G	31.515	0.59781	0.64281
B	31.631	0.59643	0.64148
C	33.379	0.57590	0.62461
D	34.934	0.56346	0.61435
E	34.829	0.56416	0.61486
G3F-10	31.862	0.59373	0.63898
G	30.087	0.61577	0.66011
B	29.904	0.61817	0.66262
C	31.515	0.59781	0.64281
D	33.213	0.57748	0.62602
E	33.379	0.57589	0.62461
G4F-10	31.340	0.54993	0.64474
G	29.472	0.62393	0.66844
B	28.970	0.63346	0.67546
C	30.390	0.61186	0.65624
D	32.205	0.58986	0.63544
E	32.657	0.58323	0.63093
G5F-10	31.631	0.59642	0.64148
G	29.719	0.62061	0.66503
B	28.907	0.63433	0.67632
C	30.087	0.61577	0.66011
D	31.977	0.59242	0.63771
E	32.713	0.58261	0.63047

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TABLE 2-25 (CONT'D)

<u>Element Location</u>	<u>Radius, cm</u>	<u>Fissions per Gram</u>	
		<u>Predicted</u>	<u>Measurement</u>
G6F-10	32.713	0.58262	0.63047
G	30.809	0.60657	0.65101
B	29.719	0.62061	0.66503
C	30.630	0.60881	0.65322
D	32.545	0.58446	0.63207
G6E-10	33.543	0.57438	0.62333
G7F-10	34.512	0.56645	0.61665
G	32.657	0.58322	0.63093
B	31.340	0.59993	0.64474
C	31.977	0.59242	0.63771
D	33.869	0.57154	0.62085
E	35.091	0.56245	0.61357
H1F-10	40.202	0.58333	0.63960
G	38.716	0.56015	0.61630
B	39.093	0.56259	0.62540
C	40.925	0.59693	0.66710
D	42.333	0.64886	0.70540
E	41.986	0.63639	0.68910
H2F-10	38.144	0.55839	0.61750
G	36.524	0.55785	0.60850
B	36.675	0.55750	0.61270
C	38.431	0.55902	0.61070
D	39.974	0.58001	0.64260
E	39.836	0.57103	0.63990
H3F-10	36.675	0.55734	0.60980
G	34.934	0.56343	0.61640
B	34.829	0.56415	0.62100
C	36.474	0.55799	0.61990
D	38.144	0.55831	0.61690
E	38.240	0.55840	0.61700

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TABLE 2-25 (CONT'D)

Element Location	Radius, cm	Fissions per Gram	
		Predicted	Measurement
H4F-10	35.866	0.55857	0.60620
G	34.031	0.57020	0.62320
B	33.652	0.57340	0.62450
H4C-10	35.144	0.56212	0.61470
D	36.923	0.55684	0.61030
E	37.269	0.55663	0.60530
H5F-10	35.764	0.55897	0.61350
G	33.869	0.57153	0.62440
B	33.213	0.57748	0.62660
C	34.512	0.56647	0.61300
D	36.373	0.55694	0.61320
E	36.973	0.55684	0.60830
H6F-10	36.373	0.55689	0.60300
G	34.459	0.56687	0.60660
B	33.543	0.57437	0.61980
C	34.618	0.56565	0.61150
D	36.524	0.55778	0.60660
E	37.367	0.55659	0.60830
H7F-10	37.661	0.55704	0.60650
G	35.764	0.55899	0.60540
B	34.618	0.56567	0.61190
C	35.455	0.56042	0.60570
D	37.367	0.55667	0.60630
E	34.431	0.55907	0.60760
H8F-10	39.559	0.56714	0.64240
G	37.709	0.55699	0.61370
B	36.373	0.55689	0.61310
C	36.973	0.55683	0.61470
D	38.858	0.56112	0.63000
E	40.111	0.58181	0.66750

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TABLE 2-25 (CONT'D)

Element Location	Radius, cm	Fissions per Gram	
		Predicted	Measurement
J3F-10	41.547	0.62157	0.69760
G	39.836	0.57068	0.63840
B	39.790	0.57028	0.64510
C	41.459	0.62015	0.70400
D	43.106	0.69851	0.80840
E	43.148	0.69902	0.80850
J4F-10	40.520	0.58891	0.65910
G	38.716	0.56014	0.62520
B	38.431	0.55886	0.61930
C	39.974	0.57949	0.63980
D	41.723	0.62669	0.68930
E	41.986	0.63586	0.70980
J5F-10	41.986	0.63751	0.73080
G	40.111	0.58218	0.66270
B	39.559	0.56752	0.64300
C	40.925	0.59781	0.68030
D	42.764	0.68328	0.79060
E	43.275	0.70880	0.81820
K	39.093	0.56281	0.63130
M	38.240	0.55857	0.61780
J6F-10	42.247	0.64533	0.73260
G	40.339	0.58529	0.66000
B	39.559	0.56724	0.64510
C	40.746	0.59369	0.68180
D	42.636	0.66339	0.74690
E	43.360	0.70987	0.81590
K	38.858	0.56111	0.62700
M	38.431	0.55888	0.62100
J7F-10	41.193	0.60418	0.66870
G	39.280	0.56444	0.61950
B	38.288	0.55852	0.60550
C	39.280	0.56422	0.62520
D	41.193	0.60340	0.67870
E	42.117	0.64090	0.69900

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TABLE 2-25 (CONT'D)

Element Location	Radius, cm	Fissions per Gram	
		Predicted	Measurement
J8F-10	42.636	0.66431	0.74300
G	40.746	0.59385	0.66540
B	39.559	0.56753	0.63210
C	40.339	0.58596	0.65030
D	42.247	0.64759	0.71710
E	43.360	0.71370	0.79210
P1F-10	45.262	0.89671	1.00640
G	43.781	0.73624	0.82120
B	44.156	0.76203	0.83390
C	45.985	0.97302	1.10100
D	47.398	1.20843	1.40950
E	47.049	1.16211	1.35610
P2F-10	44.856	0.82491	0.99900
G	43.276	0.70853	0.81040
B	43.487	0.71947	0.79920
D	46.776	1.12724	1.34430
E	46.579	1.09899	1.33490
J	41.723	0.62814	0.69400
P3F-10	44.774	0.82129	1.01230
B	43.148	0.70176	0.82330
D	46.461	1.07704	1.36290
E	46.422	1.06271	1.36450
K	46.461	1.05855	1.34220
P3L-10	46.579	1.06888	1.33210
N	43.275	0.70433	0.80210
P4F-10	45.585	0.93149	1.10660
G	43.781	0.73639	0.85640
B	43.487	0.71610	0.81770
C	45.019	0.83810	1.00100
D	46.776	1.09905	1.33580
E	47.049	1.15328	1.39480
H	42.334	0.65014	0.75440
L	45.985	0.99186	1.20220

TABLE 2-25 (CONT'D)

Element Location	Radius, cm	Fissions per Gram	
		Predicted	Measured
P5F-10	45.746	0.95930	1.13360
G	43.864	0.74617	0.86410
C	44.610	0.80997	0.96400
D	46.461	1.07736	1.30360
E	47.010	1.17984	1.43490
J	44.528	0.79797	0.93490
K	46.422	1.05444	1.25050
L	47.166	1.17228	1.40470
P6F-10	45.945	0.97434	1.17080
G	44.031	0.75440	0.87190
B	43.106	0.69546	0.78700
C	44.156	0.76204	0.89890
D	46.065	0.98399	1.16950
E	46.932	1.12723	1.36530
P7F-10	46.422	1.06859	1.22670
G	44.528	0.80183	0.89490
C	44.156	0.76920	0.87680
D	46.065	1.00602	1.15860
P7E-10	47.166	1.21106	1.40040
H	43.106	0.69794	0.77220
L	46.932	1.15029	1.36070
P8F-10	46.461	1.04963	1.21040
G	44.610	0.80032	0.89190
B	43.275	0.70525	0.78070
C	43.864	0.74453	0.82610
D	45.746	0.95683	1.08160
E	47.010	1.15616	1.32780
K	41.986	0.63672	0.71150
M	42.764	0.68013	0.73420

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TABLE 2-26

SUMMARY OF NRX-A6 AND PAX/R-1
MEASURED AND CALCULATED DRUM SPANS

Measurement Made in	Reactor Simulated in Analytical Model	EXTERMINATOR-2 Calculated Drum Span, \$	Drum Span Inferred from Measurement, \$	Ratio Measured/ Calculated
NRX-A6	NRX-A6	a 8.50	e 7.60	0.894
PAX-G0	Test Fuel Sector in PAX-G0	d 7.04*	b 6.26	0.889
PAX-G1	R-1 Sector in PAX-G1	d 8.00*	c 6.89	0.861
PAX-G2A	R-1 Sector in PAX-G2A	c 7.84	d 6.64	0.847
PAX-G3	R-1 Sector in PAX-G3	g 5.93	h 4.84	0.816

* Adjusted for $\beta = 0.0070$ (Ref. f).

- a Ref. 1
- b Ref. 13
- c Ref. 14
- d Ref. 20
- e Ref. 28
- f Ref. 30
- g Ref. 31
- h Ref. 17

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TABLE 2-27

ALLOWABLE MANUFACTURING VARIATION FOR
CONSTITUENTS IN A COMPOSITE CORE

Constituent	Allowable Variation per Fueled or Unfueled Element (+)		Core Variation (Statistical), kg
	Percent	gms	
<u>Uranium</u>	2.5 ^a	4.4 ^b	± 0.19
<u>Carbon</u>			
Fuel Matrix	2.0 ^a	7.3 ^c	$\pm 0.64^h$
Fuel Coating, First	25.0 ^d	3.0 ^a	
Fuel Coating, Second	33.0 ^d	1.0 ^a	
Unfueled Matrix	2.0 ^a	6.2 ^e	
Unfueled Coating, External	40.0 ^d	1.0 ^a	
Unfueled Coating, Internal	50.0 ^d	1.0 ^a	
<u>Zirconium</u>			
Fuel Matrix	2.0 ^a	8.8 ^f	$\pm 1.84^i$
Fuel Coating, First	20.0 ^d	19.0 ^a	
Fuel Coating, Second	33.0 ^d	7.0 ^a	
Unfueled Matrix	2.0 ^a	7.9 ^g	
Unfueled Coating, Internal	30.0 ^d	4.0 ^a	
Unfueled Coating, External	33.0 ^d	6.0 ^a	

^aDrawing 940J447 (Fuel) and 940J439 (Unfueled Element)

^bFrom (332.654 kg in Core/1878) (0.025); Mass from Reference 8

^cFrom (689.226 kg in Core/1878) (0.020); Mass from Reference 8

^dInferred from quoted tolerance and total quoted deposition (Drawing 940J447)

^eFrom (109.865/349) (0.020); Mass from Reference 8

^fFrom (821.813/1878) (0.020); Mass from Reference 8

^gFrom (137.332/349) (0.02); Mass from Reference 8

^hFrom $(7.3 + 3.0 + 1.0)\sqrt{1878} + (6.2 + 1.0 + 1.0)\sqrt{349}$

ⁱFrom $(8.8 + 19.0 + 7.0)\sqrt{1878} + (7.9 + 4.0 + 6.0)\sqrt{349}$

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TABLE 2-28

ALLOWABLE MANUFACTURING VARIATIONS
FOR CONSTITUENTS IN A GRAPHITE CORE

Constituent	Allowable Variation Fueled or Unfueled Element (+)		Core Variation (Statistical), kg
	Percent	Grams	
<u>Uranium</u>	2.5 ^a	3.7 ^e	0.16 ^f
<u>Carbon</u>			0.62 ^g
Fuel Matrix	2.0 ^a	10.5 ^e	
Fuel Coating	2.0 ^a	0.1 ^e	
Unfueled Matrix	2.0 ^c	8.7 ^e	
<u>Zirconium</u>			0.76 ^h
Fuel Coating, Internal	15.0 ^b	10.0 ^a	
Fuel Coating, External	30.0 ^b	3.0 ^a	
Unfueled Coating, Internal	30.0 ^c	4.2 ^d	
Unfueled Coating, External	33.0 ^c	6.3 ^d	
<u>Niobium</u>			0.39 ⁱ
Fuel Coating, Internal	20.0 ^b	8.0 ^a	
Fuel Coating, External	36.0 ^b	1.0 ^a	

^aDrawing 711J780 (Fuel) and 711J781 (Unfueled Element)

^bInferred from tolerance and quoted deposition

^cTaken to be the same as for Composite Fuel

^dDrawing 947C618 for total masses

^eFrom Table 1 and 3, Reference 9

^fFrom $3.7 \sqrt{1878}$, gms

^gFrom $(10.5 + 0.1) \sqrt{1878} + 8.7 \sqrt{349}$, gm

^hFrom $(10 + 3) \sqrt{1878} + (4.2 + 6.3) \sqrt{349}$, gm

ⁱFrom $(8 + 1) \sqrt{1878}$, gm

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TABLE 2-29

 REACTIVITY DIFFERENCE BETWEEN AS-BUILT FUEL CONTENTS
AND AS-SPECIFIED FUEL CONTENTS

Reactor and Material	*As-Built Content, kg	*As-Specified Content, kg	Mass Difference, kg	*Reactivity Coefficient, ¢/kg	Reactivity Difference, ¢
<u>NRX-A2</u>					
Uranium	173.620	174.456	- 0.836	13.33	- 11.1
Carbon	855.520	868.255	- 12.735	5.62	- 71.6
Niobium	66.006	61.201	4.805	- 5.42	- 26.0
					<u>- 108.7</u>
<u>NRX-A3</u>					
Uranium	171.958	172.776	- 0.818	13.33	- 10.9
Carbon	859.897	868.248	- 8.351	5.62	- 46.9
Niobium	71.143	67.662	3.481	- 5.42	- 18.9
					<u>- 76.7</u>
<u>NRX-A4</u>					
Uranium	176.037	171.511	4.526	13.33	60.3
Carbon	839.609	838.950	0.659	5.62	3.7
Niobium	73.590	76.032	- 2.442	- 5.42	13.2
					<u>77.2</u>
<u>NRX-A5</u>					
Uranium	174.987	171.287	3.700	13.33	49.3
Carbon	844.785	838.950	5.835	5.62	32.8
Niobium	91.689	87.120	4.569	- 5.42	- 24.8
					<u>57.3</u>
<u>NRX-A6</u>					
Uranium	160.389	155.054	5.335	15.01	80.1
Carbon	846.292	842.622	3.670	5.04	18.5
Niobium	80.730	79.200	1.530	- 5.55	- 8.5
					<u>90.1</u>

*Source: Data Books for designated reactors.

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TABLE 2-29 (CONT'D)
REACTIVITY DIFFERENCE BETWEEN AS-BUILT FUEL CONTENTS
AND AS-SPECIFIED FUEL CONTENTS
(Continued)

<u>Reactor and Material</u>	<u>*As-Built Content, kg</u>	<u>*As-Specified Content, kg</u>	<u>Mass Difference, kg</u>	<u>*Reactivity Coefficient, ¢/kg</u>	<u>Reactivity Difference, ¢</u>
<u>XE-2 ("Prime")</u>					
Uranium	174.384	171.281	3.103	13.33	41.4
Carbon	843.729	838.950	4.779	5.62	26.9
Niobium	83.232	79.200	4.032	- 5.42	- 21.9
					<u>46.4</u>

*Source: Data Books for designated reactors.

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TABLE 2-30

NRX MANUFACTURING VARIABILITY STATISTICS

Point	<u>Estimate Of Population Parameters</u>		
	R_i (Reactivity Difference), ¢	Absolute Value of ($R_i - M$), ¢	Cumulative Frequency
1	- 108.7	123.0	0.083
2	- 76.7	91.0	0.250
3	46.4	32.1	0.416
4	57.3	43.0	0.583
5	77.2	62.9	0.750
6	90.1	75.8	0.917

M = Estimate of Population Mean = $1/6$ (Sum of the R_i values) = $+14.3¢$

S = Estimate of the Population Standard Deviation calculated from the above sample =

$$\sqrt{\frac{\sum (R_i - M)^2}{6 - 1}} = \sqrt{\frac{35991.46}{5}} = \pm 84.8¢$$

Population Fraction and Confidence Level

Population Fraction	Confidence Level, %	K^*	$KS \pm, \$$	$M + KS, \$$	$M - KS, \$$
0.99	95	5.775	4.90	5.04	- 4.76
0.95	95	4.414	3.74	3.88	- 3.60
0.90	95	3.712	3.15	3.29	- 3.01
0.75	95	2.604	2.21	2.35	- 2.07
0.99	90	4.870	4.13	4.27	- 3.99
0.95	90	3.723	3.16	3.30	- 3.02
0.90	90	3.131	2.66	2.80	- 2.52
0.75	90	2.196	1.86	2.00	- 1.72

*Two Sided K Factors

TABLE 2-31

SUMMARY OF EXTERMINATOR-2 CALCULATIONS
FOR PAX/R-1 ABSOLUTE REACTIVITY

Reactor Simulated in EXTERMINATOR-2 Calculation	Computer Run ID	Eigenvalue	Reactivity *, \$
PAX-GO KIWI 300° Sector)	267 10/24/70	0.999006	-0.142
PAX-GO (R-1 Test** Sector)	099 10/17/70	1.001344	+0.192
PAX-G1 (R-1 Test Sector)	086 10/10/70	0.990977	-1.289
PAX-G2 (R-1 Test Sector)	807 9/14/70	0.998565	-0.205
PAX-G3 (R-1 Test Sector)	678 9/21/70	1.002701	+0.386

* Converted with $\beta = 0.0070$ (Ref. 39).

** Model composed of 6 test sectors.

TABLE 2-32

SUMMARY OF CORRECTIONS TO TWO DIMENSIONAL CALCULATIONS
FOR THE PAX-G REACTOR SERIES

<u>Items not Included in EXTERMINATOR-2 Calculation</u>	<u>Effect on Total Reactivity if Item had been included in the EXTERMINATOR-2 Calculations, δ</u>	
	<u>PAX - G0, G1, G2</u>	<u>PAX- G3</u>
Pressure Vessel Simulation	56 ^a	32 ^c
Detector	5 ^b	5
Walls	10 ^b	10
Test Stand	7 ^b	7
	<hr/>	<hr/>
TOTAL	78	54

^aReference 32

^bReference 33, page 89

^cSee PAX-G3 Drawings 388D914, 388D915 and 388D949, and PAX Drawing 710J344

TABLE 2-33

STATISTICAL TREATMENT OF ERROR IN ABSOLUTE REACTIVITY
CALCULATION WITH WISDM

Estimate of Population Parameters

Point	E _i (Error in Absolute Reactivity), \$	Absolute Value of (E _i -M)	Cumulative Frequency
1	0.694	0.866	0.125
2	1.141	0.419	0.375
3	2.166	0.606	0.625
4	2.225	0.665	0.875

M = Estimate of population mean = 1/4 (Sum of E_i Values) = \$1.56

S = Estimate of population standard deviation calculated from the above sample =

$$\sqrt{\frac{\sum (E_i - M)^2}{(4-1)}} = \sqrt{\frac{1.7350}{3}} = \pm 0.76 \$$$

Population Fraction and Confidence Level

Population Fraction	Confidence Level, %	K*	KS, \$	M + KS	M-KS
0.99	95	8.299	6.31	7.87	-4.75
0.95	95	6.370	4.84	6.40	-3.28
0.90	95	5.369	4.08	5.64	-2.52
0.75	95	3.779	2.87	4.43	-1.31
0.99	90	6.440	4.89	6.45	-3.33
0.95	90	4.943	3.76	5.32	-2.20
0.90	90	4.166	3.17	4.73	-1.61
0.75	90	2.932	2.23	3.79	-0.67

* Two sided K factors.

TABLE 2-34

SUMMARY OF ERRORS IN NRX PREDICTIONS
OF ABSOLUTE REACTIVITY

Reactor	Ref.	Predicted Critical Control Drum Position, Degrees	Measured Critical Control Drum Position, Degrees	Measured Drum Span, °	Analytical Bias , \$*
NRX-A2	35	90.0	97.8	9.0	- 0.564
NRX-A3	34	90.0	89.1	8.8	+ 0.069
NRX-A4	29	94.0	99.1	8.8	- 0.373
NRX-A5	36	93.0	99.4	8.2	- 0.398
NRX-A6	28	88.0	97.2	7.6	- 0.648
XE-Prime	25	94.4	98.8	8.2	- 0.256

*Computed as $0.5 (\text{Measured Drum Span}) (\cos \theta_M - \cos \theta_p)$

The uncertainty in the measured drum span itself has only a second order effect on the listed analytical bias.

TABLE 2-35

NRX CRITICALITY ERROR STATISTICS

Estimate Of Population Parameters

<u>Point</u>	<u>B_i (Analytical Bias), \$</u>	<u>Absolute Value of (B_i - M), \$</u>	<u>Cumulative Frequency</u>
1	+ 0.069	0.431	0.083
2	- 0.256	0.106	0.250
3	- 0.373	0.011	0.416
4	- 0.398	0.036	0.583
5	- 0.564	0.202	0.750
6	- 0.648	0.286	0.917

M = Estimate of Population Mean = 1/6 (Sum of the B_i values) = -0.362\$

S = Estimate of the Population Standard Deviation calculated from the above sample =

$$\sqrt{\frac{\sum(B_i - M)^2}{6-1}} = \sqrt{\frac{0.321014}{5}} = \pm 0.253\$$$

Population Fraction and Confidence Level

<u>Population Fraction</u>	<u>Confidence Level, %</u>	<u>K*</u>	<u>KS, \$</u>	<u>M - KS, \$</u>	<u>M + KS, \$</u>
0.99	95	5.775	1.461	- 1.823	1.109
0.95	95	4.414	1.117	- 1.479	0.755
0.90	95	3.712	0.939	- 1.301	0.577
0.75	95	2.604	0.659	- 1.021	0.297
0.99	90	4.870	1.232	- 1.594	0.870
0.95	90	3.723	0.942	- 1.304	0.580
0.90	90	3.131	0.792	- 1.154	0.430
0.75	90	2.196	0.556	- 0.918	0.194

*Two Sided K Factors

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TABLE 2-36
FULL POWER REACTIVITY CHANGE DATA

Reactor	EP	Ambient Critical Position, degrees	"Full Power" CRT, seconds	Drum Position at CRT, degrees	Corrected Drum Position at CRT, degrees
NRX-A2	IV	97.5	14085	103.2	103.2
NRX-A3	IV	89.6	21550	98.6	^a 96.9
	V	89.2	12880	99.8	^b 98.3
NRX/EST	III	^e 98.3	^f 22430	^f 107.5	^f 107.5
	IV	^e 99.2	^f 11189	^f 110.2	^k 110.0
	IVA	^e 105.6	^f 9298	^f 116.5	^l 116.1
NRX-A5	^g III	99.3	25300	108.8	-----
	^g IV	102.4	12975	115.0	-----
NRX-A6	IIIA	93.4	14403	ⁱ 104.4	ⁱ 104.4
XE-P	5C	95.0	38935	ⁱ 103.6	
Design Point	-----	-----	-----	-----	-----

a) Reference 34, page 3-41

Delayed neutron effect	- 2 β
Corrosion to CRT	-11.2 β
Total	-13.2 β

This reactivity loss compensated by outward drum motion at 7.56 β /degrees of 1.7°. Therefore, corrected hot critical is 98.6° - 1.7° = 96.90.

b) Reference 34, page 3-42

Delayed neutron effect	- 2 β
Corrosion to CRT	- 9.7 β
Total	-11.7 β

This reactivity loss compensated by outward drum motion at 7.56 β /degrees of 1.5°. Therefore, corrected hot critical is 99.8° - 1.5° = 98.3°.

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TABLE 2-36 (CONT'D)

EP	<u>Best Estimate of Thermal Conditions</u>					
	<u>Cold to Hot Swing, degrees</u>	<u>Drum Worth, °/degrees</u>	<u>Cold to Hot Swing, cents</u>	<u>Chamber Pressure (P_c), psia</u>	<u>Chamber Temperature (T_c), °R</u>	<u>P_c/T psia/°R</u>
IV	5.7	7.7	-43.9	^d 581	^d 3815	0.1523
IV	7.3	7.56	-55.2	^c 569	^c 3883	0.1465
V	9.1	7.56	-68.8	^c 556	^c 3922	0.1418
III	9.2	7.6	-68.8	^f 538	^f 3991	0.1347
IV	10.8	7.6	-80.0	^f 548	^f 4006	0.1367
IVA	10.5	7.6	-74.8	^f 567	^f 4144	0.1369
9III	-----	-----	-53.9	555	3854	0.1440
9IV	-----	-----	-79.3	570	4169	0.1367
IIIA	11.0	-----	-72.0	^h 586	^h 4131	0.1419
5C	EXCLUDE DUE TO X LOSS			ⁱ 555	ⁱ 4098	0.1354
Design Point	-----	-----	-----	550	4090	0.1345

c) Page 5-18 for chamber pressure data (Ref. 35)
Page 5-13 for chamber temperature data-
Value shown is average of T_{cc1} and T_{cc2}.

d) Pages 4-7, 4-20, 4-26 (Ref. 35)
T value is calculated and considered the best
estimate of temperature in chamber. Measured
value is 3480° R.

e) Reference 29, Page 6-45

f) Reference 29, Page 6-79

g) Reference 36, Page 7-71

Cold-To-Hot Swings:

$$-65.9^\circ + 12^\circ = -53.9^\circ$$

$$-85.2^\circ + 5.9^\circ = -79.3^\circ$$

h) Reference 28, Page 3-39

i) Reference 28, Page 7-79

j) NULIST Data Reduction in Pittsburgh

k) $110.2^\circ - 1.6^\circ/7.6^\circ \text{ degrees} = 110.0^\circ$

l) $116.5^\circ - 3.2^\circ/7.6^\circ \text{ degrees} = 116.1^\circ$

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TABLE 2-37
COLD-TO-HOT SWING DATA CORRECTED TO DESIGN CONDITIONS

Reactor	EP	CRT, seconds	A "Observed" Cold-To-Hot Swing, cents	C Feedback from Formula*, cents	B Change in Feedback to reach Design Conditions (D-C), cents	(A+B) Condition Normalized Cold-To-Hot Swing, cents
NRX-A2	IV	14085	-43.9	-46.1	-34.0	-77.9
NRX-A3	IV	21550	-55.2	-55.9	-24.2	-79.4
	V	12880	-68.8	-62.8	-17.3	-86.1
NRX/EST	III	22430	-68.8	-73.9	- 6.2	-75.0
	IV	11189	-80.0	-72.9	- 7.2	-87.2
	IVA	9298	-74.8	-81.0	+ 0.9	-73.9
NRX-A5	III	25300	-53.9	-56.6	-23.5	-77.4
	IV	12975	-79.3	-82.7	+ 2.6	-76.7
NRX-A6	IIIA	14403	-72.0	-71.4	- 4.4	-76.4

Design Point

NRX-A2 to NRX-A5	-	-	-80.1	} D
NRX-A6	-	-	-75.8	

Average Cold-To-Hot Swing (Mean of Column headed by (A+B)) = -78.9¢

Estimate of Standard Deviation = +4.7¢ ("S") -

* Formula used for correction has general form $A + BT_c + C (P_c/T_c)$.

Constants for NRX-A2 through NRX-A5 are: A = 34.2, B = -0.0601, C = 978.

Constants for NRX-A6 are: A = 31.9, B = -0.0560, C = 902.

TABLE 2-38
STATISTICAL TREATMENT OF NRX COLD-TO-HOT REACTIVITY SWING DATA

<u>Population Fraction</u>	<u>Confidence Level, %</u>	<u>K*</u>	<u>KS, ¢</u>
0.99	95	4.631	21.8
0.95	95	3.532	16.6
0.90	95	2.967	13.9
0.75	95	2.078	9.8
0.99	90	4.098	19.3
0.95	90	3.125	14.7
0.90	90	2.626	12.3
0.75	90	1.839	8.6

* Two Sided Factors, Reference 26.

TABLE 2-39
PRELIMINARY NERVA MULTIZONED FUEL DISTRIBUTION - COMPOSITE

Fuel Loading Code	Number of Fueled Elements	Loading Mg/cc *** of Fueled Graphite	Total Uranium 52.0 inches ₃ gm	Total* Carbon 52.0 inches ₃ gm	Total** Zirconium 52.0 inches ₃ gm
11	54	463	130.7	382.0	552.6
12	114	476	134.4	382.0	552.6
13	102	502	141.7	382.0	552.6
14	72	515	145.4	382.0	552.6
15	96	529	149.3	382.0	552.6
16	276	577	162.9	382.0	552.6
17	96	593	167.4	382.0	552.6
18	96	608	171.6	382.0	552.6
19	150	667	188.3	382.0	552.6
20	235.3	688	194.2	382.0	552.6
21	388.7	709	200.2	382.0	552.6
22	102	730	206.1	382.0	552.6
23	96	765	216.0	382.0	552.6

* Includes 12.0 grams/element from bore deposition and 3.0 grams/element from external coating.

** Includes 93.0 grams/element from bore deposition and 22.0 grams/element from external coating.

*** Loadings do not include the additional 30 MG/cc to allow for shimming by the additional element selection procedure.

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TABLE 2-40

POWER FACTORS FOR THE R-1 REACTOR WITH THE COMPOSITE
FUELED CORE CONTROL DRUMS AT 50 DEGREES

1	Q11D***	1040	1040	1040	1043	1043	1043	1043	1046	1046	1046
19	1	1046	1046	1055	1055	1055	1055	1063	1063	1063	
1	A 1F***	1037	1036	1034	1040	1039	1038	1037	1044	1043	1042
19	1	1041	1040	1052	1051	1050	1049	1061	1060	1058	
1	A 1G***	1040	1040	1040	1043	1043	1043	1043	1046	1046	1046
19	1	1046	1045	1054	1054	1054	1053	1062	1062	1061	
1	A 1H***	1039	1040	1040	1041	1042	1043	1043	1043	1044	1045
19	1	1046	1046	1052	1053	1054	1054	1060	1061	1062	
1	A 1C***	1031	1033	1034	1034	1036	1037	1038	1038	1039	1040
19	1	1041	1043	1047	1049	1050	1051	1057	1058	1060	
1	A 1D***	1026	1027	1027	1030	1031	1031	1032	1035	1036	1036
19	1	1036	1037	1045	1046	1046	1046	1055	1056	1056	
1	A 1E***	1029	1029	1027	1034	1033	1032	1031	1039	1038	1037
19	1	1036	1035	1048	1047	1046	1045	1057	1056	1056	
1	B 1F***	1045	1042	1039	1050	1047	1044	1040	1055	1052	1049
29	1	1045	1042	1062	1059	1056	1052	1069	1066	1062	
1	B 1G***	1054	1053	1052	1055	1054	1057	1056	1060	1060	1059
29	1	1058	1057	1067	1066	1065	1064	1073	1072	1071	
1	B 1H***	1051	1053	1051	1051	1053	1055	1054	1052	1054	1056
29	1	1058	1060	1060	1062	1064	1066	1068	1070	1072	
1	B 1C***	1031	1035	1038	1034	1037	1040	1043	1036	1040	1043
29	1	1046	1049	1048	1051	1054	1058	1059	1062	1065	
1	B 1D***	1019	1020	1021	1024	1025	1026	1027	1030	1031	1032
29	1	1033	1034	1042	1043	1044	1045	1054	1055	1056	

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TABLE 2-40 (CONT'D)

1 B 1E***	1025	1023	1021	1032	1030	1028	1026	1039	1037	1035
29	1	1033	1030	1049	1047	1045	1043	1059	1057	1055
1 B 2F***	1045	1042	1040	1050	1048	1045	1042	1056	1054	1051
29	1	1049	1046	1065	1062	1060	1057	1073	1071	1068
1 B 2G***	1030	1028	1026	1033	1031	1029	1031	1036	1034	1033
19	1	1034	1032	1043	1041	1042	1040	1049	1051	1049
1 B 2B***	1032	1033	1033	1033	1034	1034	1035	1035	1036	1036
19	1	1036	1037	1042	1043	1043	1043	1049	1050	1050
1 B 2C***	1050	1053	1052	1052	1054	1057	1056	1054	1056	1059
29	1	1058	1060	1063	1066	1068	1067	1073	1075	1078
1 B 2D***	1034	1037	1039	1038	1041	1043	1045	1043	1045	1047
29	1	1049	1051	1055	1057	1059	1061	1066	1069	1070
1 B 2E***	1031	1031	1030	1037	1037	1037	1036	1044	1044	1043
29	1	1043	1042	1055	1055	1054	1054	1067	1066	1065
1 C 1F***	1050	1045	1040	1056	1051	1045	1040	1061	1056	1051
39	1	1046	1040	1067	1062	1057	1051	1073	1067	1062
1 C 1G***	1021	1020	1018	1021	1020	1018	1017	1022	1020	1019
29	1	1017	1015	1026	1024	1023	1021	1030	1028	1026
1 C 1B***	1011	1014	1017	1009	1013	1016	1019	1008	1011	1015
29	1	1018	1021	1015	1018	1022	1025	1022	1025	1029
1 C 1C***	1028	1033	1038	1030	1035	1040	1045	1032	1037	1042
39	1	1048	1052	1044	1050	1055	1060	1057	1062	1067
1 C 1D***	1009	1009	1011	1014	1016	1018	1019	1021	1023	1025
39	1	1027	1028	1035	1037	1039	1041	1050	1051	1053

TABLE 2-40 (CONT'D)

1 C 1E***	1018	1015	1012	1027	1024	1020	1017	1036	1033	1029
39	1	1026	1022	1047	1043	1040	1036	1058	1054	1050
1 C 2F***	1011	1006	1002	1017	1013	1008	1003	1023	1019	1015
29	1	1010	1005	1030	1026	1022	1017	1038	1033	1029
1 C 2G***	1033	1031	1028	1035	1033	1030	1027	1037	1035	1032
29	1	1030	1027	1042	1039	1037	1034	1047	1044	1041
1 C 2R***	1033	1035	1036	1032	1034	1035	1037	1031	1033	1035
29	1	1037	1038	1038	1039	1041	1043	1044	1046	1047
1 C 2C***	1010	1014	1018	1010	1015	1019	1023	1011	1016	1020
29	1	1024	1028	1022	1026	1031	1035	1032	1037	1041
1 C 2D***	1034	1037	1040	1039	1042	1045	1048	1045	1048	1051
39	1	1054	1057	1059	1062	1065	1068	1073	1076	1079
1 C 2E***	1035	1033	1031	1043	1041	1040	1038	1051	1050	1048
39	1	1046	1044	1064	1062	1060	1058	1076	1074	1072
1 C 3F***	1049	1045	1041	1057	1053	1049	1045	1055	1061	1057
39	1	1053	1049	1075	1071	1066	1062	1084	1080	1075
1 C 3G***	1027	1023	1020	1030	1027	1023	1019	1034	1031	1027
29	1	1023	1019	1039	1036	1032	1028	1044	1041	1037
1 C 3B***	1036	1037	1035	1036	1036	1036	1036	1036	1036	1036
29	1	1036	1036	1041	1041	1041	1041	1046	1046	1046
1 C 3C***	1020	1023	1027	1019	1023	1027	1030	1019	1023	1027
29	1	1031	1034	1028	1032	1036	1039	1037	1041	1044
1 C 3D***	1041	1045	1049	1045	1049	1053	1057	1049	1053	1057
39	1	1061	1065	1062	1066	1071	1075	1075	1080	1084

TABLE 2-40 (CONT'D)

1 C 3E***	1031	1031	1031	1039	1039	1039	1039	1 46	1047	1047
39	1	1047	1046	1060	1060	1060	1060	1073	1073	1073
1 D 1F***	1010	1003	998	1015	1009	1004	997	1021	1015	1008
49	1	1003	996	1026	1019	1014	1007	1030	1024	1018
1 D 1G***	1035	1033	1032	1036	1034	1031	1029	1035	1032	1031
49	1	1028	1026	1037	1035	1033	1031	1039	1037	1035
1 D 1B***	1022	1027	1031	1019	1024	1028	1032	1016	1021	1025
49	1	1030	1034	1023	1027	1032	1036	1029	1034	1038
1 D 1C***	1007	1013	1020	1008	1015	1021	1028	1010	1016	1023
59	1	1030	1035	1023	1030	1036	1042	1035	1043	1048
1 D 1D***	984	987	989	992	991	993	995	997	999	1001
59	1	1003	1005	1012	1014	1016	1018	1027	1029	1031
1 D 1E***	995	991	987	1005	1001	996	992	1015	1011	1007
59	1	1002	998	1026	1022	1018	1013	1037	1033	1028
1 D 2F***	1029	1023	1017	1036	1030	1024	1017	1041	1037	1031
49	1	1025	1018	1048	1043	1037	1031	1055	1049	1043
1 D 2G***	1034	1031	1027	1035	1032	1028	1025	1036	1033	1029
39	1	1026	1022	1039	1035	1032	1028	1041	1038	1034
1 D 2B***	1030	1033	1036	1027	1030	1033	1036	1025	1028	1031
39	1	1033	1036	1030	1033	1036	1039	1036	1039	1041
1 D 2C***	1021	1027	1033	1021	1027	1033	1038	1021	1027	1034
49	1	1038	1044	1033	1039	1045	1050	1045	1051	1055
1 D 2D***	992	995	998	998	1001	1005	1007	1004	1008	1010
49	1	1013	1016	1018	1021	1025	1028	1033	1036	1040

TABLE 2-40 (CONT'D)

1 D 2E***	996	993	991	1004	1003	1000	997	1014	1011	1008
49	1	1007	1004	1026	1023	1021	1018	1038	1035	1032
1 D 3F***	1028	1023	1017	1035	1031	1025	1020	1043	1039	1034
49	1	1028	1022	1051	1047	1041	1036	1059	1055	1049
1 D 3G***	1038	1033	1028	1040	1035	1031	1026	1043	1038	1034
39	1	1029	1024	1046	1041	1037	1032	1049	1044	1040
1 D 3B***	1044	1045	1046	1042	1043	1044	1045	1040	1041	1042
39	1	1043	1044	1044	1045	1047	1048	1048	1050	1051
1 D 3C***	1015	1021	1027	1014	1020	1026	1031	1015	1019	1025
39	1	1031	1036	1023	1029	1035	1041	1033	1039	1045
1 D 3D***	1007	1011	1016	1011	1016	1020	1024	1016	1021	1025
49	1	1030	1034	1031	1035	1040	1044	1045	1050	1054
1 D 3E***	1002	1001	1000	1010	1009	1008	1006	1019	1018	1017
49	1	1016	1014	1032	1031	1030	1029	1046	1044	1043
1 D 4F***	1008	1004	1000	1017	1012	1008	1004	1027	1022	1017
49	1	1012	1008	1036	1031	1026	1021	1046	1041	1035
1 D 4G***	1021	1015	1009	1025	1019	1013	1009	1029	1023	1018
39	1	1014	1008	1033	1027	1023	1018	1036	1031	1027
1 D 4B***	1038	1037	1036	1036	1035	1035	1034	1034	1034	1034
39	1	1033	1032	1038	1037	1037	1036	1041	1040	1040
1 D 4C***	1016	1021	1026	1014	1019	1024	1029	1013	1017	1022
39	1	1027	1032	1022	1025	1030	1035	1030	1033	1039
1 D 4D***	1004	1009	1015	1007	1012	1018	1023	1010	1015	1021
49	1	1027	1032	1023	1029	1035	1040	1037	1043	1049

TABLE 2-40 (CONT'D)

1 D 4E***	1013	1014	1014	1021	1022	1022	1023	1030	1031	1032
59	1	1032	1032	1044	1044	1045	1045	1058	1059	1059
1 E 1F***	1035	1026	1022	1041	1033	1024	1019	1048	1040	1031
69	1	1026	1017	1052	1043	1034	1029	1055	1046	1041
1 E 1G***	1071	1068	1065	1068	1065	1062	1059	1066	1063	1061
69	1	1057	1054	1067	1064	1061	1057	1067	1064	1061
1 E 1B***	1051	1057	1063	1046	1052	1058	1064	1041	1047	1053
69	1	1059	1065	1047	1053	1059	1065	1053	1059	1065
1 E 1C***	1026	1035	1044	1027	1036	1045	1049	1028	1037	1046
79	1	1050	1059	1043	1052	1061	1065	1058	1067	1072
1 E 1D***	988	991	994	997	1000	1006	1009	1010	1013	1016
79	1	1019	1021	1027	1031	1034	1036	1045	1049	1052
1 E 1E***	1010	1004	995	1022	1016	1010	1005	1035	1029	1023
79	1	1017	1011	1047	1041	1035	1029	1059	1053	1047
1 E 2F***	1066	1057	1049	1073	1065	1057	1048	1078	1073	1065
69	1	1057	1048	1083	1078	1070	1061	1092	1083	1075
1 E 2G***	1017	1013	1010	1017	1013	1009	1005	1017	1013	1009
59	1	1006	1002	1018	1014	1011	1007	1019	1015	1011
1 E 2B***	1010	1013	1017	1005	1009	1013	1017	1002	1006	1009
59	1	1013	1017	1007	1011	1014	1018	1011	1015	1019
1 E 2C***	1049	1057	1066	1048	1057	1065	1073	1048	1057	1065
69	1	1073	1078	1061	1070	1078	1083	1075	1083	1092
1 E 2D***	1014	1018	1022	1022	1021	1026	1029	1025	1030	1034
69	1	1038	1042	1042	1047	1051	1055	1060	1064	1068

TABLE 2-40 (CONT'D)

1 E 2E***	1022	1018	1014	1029	1026	1021	1022	1042	1038	1034
69	1	1030	1025	1055	1051	1047	1042	1068	1064	1060
1 E 3F***	990	987	979	998	992	988	980	1007	1000	993
59	1	989	982	1014	1007	1003	995	1020	1013	1009
1 F 3G***	1029	1025	1020	1030	1026	1021	1016	1031	1028	1023
59	1	1018	1013	1034	1030	1025	1020	1036	1031	1026
1 E 3H***	1031	1034	1036	1029	1030	1032	1034	1025	1028	1030
59	1	1032	1033	1029	1032	1034	1036	1033	1035	1037
1 E 3C***	995	1002	1009	994	1001	1008	1015	992	1000	1007
59	1	1013	1020	1003	1010	1018	1024	1014	1021	1028
1 E 3D***	1037	1043	1048	1043	1049	1054	1059	1049	1055	1061
69	1	1066	1071	1067	1072	1078	1083	1084	1089	1095
1 E 3E***	1035	1032	1030	1046	1044	1041	1039	1057	1055	1053
69	1	1050	1048	1072	1070	1068	1065	1087	1085	1082
1 E 4F***	1064	1057	1050	1071	1068	1060	1053	1081	1075	1071
69	1	1064	1056	1090	1083	1080	1072	1099	1092	1088
1 E 4G***	1023	1017	1011	1026	1020	1014	1008	1029	1023	1018
59	1	1011	1005	1032	1026	1020	1014	1034	1028	1022
1 E 4H***	1035	1036	1036	1032	1032	1033	1033	1030	1031	1032
59	1	1032	1032	1033	1034	1034	1035	1036	1036	1037
1 E 4C***	1005	1012	1018	1003	1009	1016	1022	1000	1007	1013
59	1	1020	1026	1010	1016	1023	1029	1019	1025	1032
1 E 4D***	1044	1051	1057	1048	1055	1062	1068	1053	1060	1066
69	1	1073	1076	1070	1076	1083	1086	1086	1093	1096

TABLE 2-40 (CONT'D)

1 E 4E***	1031	1030	1030	1041	1041	1040	1040	1052	1052	1051	
69	1	1051	1050	1068	1068	1067	1067	1084	1084	1083	
1 E 5F***	1033	1026	1020	1044	1038	1031	1024	1055	1049	1043	
69	1	1036	1029	1066	1059	1053	1046	1076	1070	1063	
1 E 5G***	998	991	987	1002	995	992	984	1007	1000	993	
59	1	989	982	1010	1003	999	991	1012	1009	1001	
1 E 5B***	1020	1019	1018	1018	1017	1016	1014	1015	1014	1013	
59	1	1012	1011	1017	1016	1015	1014	1019	1018	1017	
1 E 5C***	996	1002	1008	993	999	1004	1010	992	995	1001	
59	1	1007	1013	1000	1003	1009	1015	1007	1010	1016	
1 E 5D***	1031	1039	1046	1033	1041	1049	1056	1036	1044	1051	
69	1	1059	1067	1052	1059	1067	1075	1067	1075	1083	
1 E 5E***	1011	1012	1013	1021	1022	1023	1024	1027	1028	1029	
69	1	1031	1031	1043	1045	1046	1047	1060	1062	1063	
1 F 1F***	1066	1056	1046	1070	1063	1053	1043	1078	1071	1060	
99	1	1051	1041	1083	1073	1063	1053	1086	1075	1065	
1 F 1G***	1017	1013	1010	1013	1010	1007	1003	1010	1007	1004	
89	1	1000	997	1009	1006	1003	999	1008	1005	1004	
1 F 1B***	995	1001	1008	988	995	1002	1008	985	989	995	
89	1	1002	1008	991	994	1001	1007	996	1003	1006	
1 F 1C***	1024	1034	1043	1025	1034	1043	1053	1025	1035	1044	
99	1	1054	1064	1041	1050	1060	1069	1056	1065	1075	
1 F 1D***	990	993	996	999	1002	1005	1007	1008	1011	1014	
99	1	1017	1019	1025	1028	1031	1034	1043	1046	1049	

TABLE 2-40 (CONT'D)

1 F 1E***	1008	1003	997	1020	1014	1009	1003	1033	1027	1021
99	1	1015	1010	1045	1039	1033	1027	1057	1051	1045
1 F 2F***	1015	1005	996	1023	1013	1004	994	1035	1022	1012
89	1	1003	993	1035	1026	1016	1006	1039	1029	1019
1 F 2G***	1042	1037	1032	1040	1036	1031	1026	1039	1034	1029
79	1	1024	1019	1037	1033	1028	1023	1036	1031	1027
1 F 28***	1030	1035	1040	1023	1029	1034	1039	1017	1022	1028
79	1	1033	1038	1021	1026	1032	1037	1025	1030	1036
1 F 2C***	992	1002	1011	991	1000	1010	1020	990	1000	1009
89	1	1019	1032	1004	1014	1023	1033	1018	1027	1037
1 F 2D***	1039	1043	1047	1047	1051	1056	1060	1056	1060	1065
99	1	1069	1074	1075	1079	1084	1085	1090	1095	1100
1 F 2E***	1050	1045	1040	1063	1058	1053	1048	1073	1071	1066
99	1	1061	1056	1087	1082	1080	1075	1101	1096	1091
1 F 3F***	1015	1006	994	1025	1015	1006	994	1035	1025	1016
79	1	1003	994	1040	1031	1021	1008	1045	1036	1026
1 F 3G***	1062	1056	1050	1062	1056	1050	1048	1062	1057	1050
79	1	1049	1043	1062	1060	1054	1048	1066	1060	1054
1 F 3B***	1061	1065	1068	1055	1059	1062	1066	1053	1053	1056
79	1	1060	1064	1057	1060	1060	1063	1060	1063	1067
1 F 3C***	1013	1023	1032	1011	1020	1030	1039	1009	1018	1028
79	1	1037	1047	1021	1031	1040	1050	1033	1043	1053
1 F 3D***	985	990	995	991	997	1002	1008	998	1004	1009
89	1	1015	1020	1016	1022	1027	1033	1034	1040	1049

TABLE 2-40 (CONT'D)

1 F 3E***	986	983	982	998	994	991	988	1010	1007	1004
89	1	1000	997	1025	1022	1018	1015	1040	1036	1032
1 F 4F***	1015	1006	994	1025	1017	1008	996	1036	1028	1019
79	1	1010	998	1044	1035	1026	1017	1051	1042	1033
1 F 4G***	1039	1032	1025	1041	1034	1027	1024	1043	1036	1029
69	1	1026	1019	1043	1036	1034	1026	1043	1041	1033
1 F 4B***	1049	1051	1053	1044	1046	1047	1049	1039	1041	1043
69	1	1044	1046	1041	1043	1045	1046	1042	1044	1046
1 F 4C***	1035	1044	1048	1031	1040	1049	1053	1028	1037	1046
79	1	1050	1059	1038	1048	1057	1061	1049	1059	1068
1 F 4D***	1001	1008	1014	1006	1013	1020	1030	1011	1018	1025
89	1	1036	1042	1029	1039	1046	1054	1050	1057	1065
1 F 4E***	992	990	988	1003	1002	1000	998	1015	1014	1012
89	1	1010	1009	1031	1030	1028	1026	1051	1049	1044
1 F 5F***	1013	1006	998	1028	1017	1009	1001	1040	1032	1021
89	1	1013	1005	1050	1041	1030	1021	1059	1050	1038
1 F 5G***	1048	1044	1035	1051	1047	1039	1030	1055	1051	1043
79	1	1034	1026	1060	1052	1043	1035	1061	1052	1044
1 F 5B***	1042	1043	1042	1038	1038	1038	1038	1033	1034	1034
69	1	1034	1033	1034	1034	1034	1034	1039	1039	1039
1 F 5C***	1035	1044	1048	1030	1039	1047	1051	1026	1034	1043
79	1	1051	1055	1035	1043	1052	1060	1044	1052	1061
1 F 5D***	998	1006	1013	1001	1009	1017	1028	1005	1013	1021
89	1	1032	1040	1021	1030	1041	1050	1038	1050	1059

TABLE 2-40 (CONT'D)

1 F 5E***	982	982	982	989	990	990	989	1001	1001	1001
89	1	1001	1001	1017	1018	1018	1017	1034	1034	1034
1 F 6F***	977	971	984	986	982	976	969	998	992	988
89	1	981	974	1009	1002	995	991	1019	1012	1005
1 F 6G***	1013	1004	992	1018	1009	997	988	1024	1015	1002
79	1	993	984	1025	1016	1004	995	1026	1014	1005
1 F 6H***	1045	1044	1042	1041	1039	1038	1036	1037	1036	1034
79	1	1032	1030	1037	1035	1033	1031	1036	1034	1032
1 F 6C***	1015	1023	1030	1009	1017	1024	1032	1000	1011	1019
79	1	1026	1034	1007	1018	1026	1033	1014	1025	1033
1 F 6D***	979	984	993	980	986	995	1003	982	988	997
89	1	1006	1014	995	1004	1013	1022	1011	1020	1029
1 F 6E***	1036	1037	1039	1046	1048	1049	1051	1057	1059	1060
99	1	1062	1064	1075	1077	1079	1081	1091	1093	1094
1 G 1F***	993	981	969	997	984	972	969	1001	988	975
A9	1	972	969	993	980	974	971	984	979	973
1 G 1G***	1032	1026	1020	1026	1024	1018	1013	1019	1018	1016
A9	1	1011	1006	1007	1005	1008	1003	994	997	1000
1 G 1B***	1004	1011	1017	999	1003	1010	1015	993	997	1001
A9	1	1007	1012	985	990	1000	1005	978	988	998
1 G 1C***	960	967	972	956	965	969	979	952	961	967
A9	1	977	986	953	961	971	980	956	966	975
1 G 1D***	953	953	952	958	958	957	965	962	962	962
B9	1	969	976	963	963	973	980	964	973	983

TABLE 2-40 (CONT'D)

1 G 1E***	974	968	961	985	970	964	963	999	981	966
B9	1	965	964	992	974	972	972	985	983	970
1 G 2F***	1026	1017	1008	1033	1024	1015	1006	1041	1032	1023
A9	1	1014	1005	1045	1036	1027	1018	1049	1039	1030
1 G 2G***	1042	1037	1032	1039	1035	1030	1026	1038	1033	1029
99	1	1024	1019	1036	1032	1028	1023	1035	1031	1027
1 G 2B***	1028	1033	1039	1021	1027	1032	1038	1015	1021	1026
99	1	1032	1037	1020	1025	1030	1036	1024	1029	1035
1 G 2C***	1001	1010	1019	1001	1010	1019	1028	1001	1009	1018
A9	1	1027	1037	1014	1023	1032	1041	1028	1037	1046
1 G 2D***	965	969	972	972	976	980	983	980	984	983
A9	1	991	995	997	1000	1004	1008	1013	1017	1021
1 G 2E***	976	971	967	987	983	978	974	998	994	990
A9	1	985	981	1011	1006	1002	997	1023	1019	1014
1 G 3F***	1022	1013	1004	1031	1021	1012	1003	1040	1031	1021
99	1	1012	1003	1045	1035	1026	1017	1050	1040	1031
1 G 3G***	1071	1068	1062	1070	1067	1061	1055	1073	1067	1060
99	1	1054	1048	1071	1065	1059	1053	1070	1064	1058
1 G 3B***	1066	1071	1076	1062	1067	1072	1073	1056	1060	1065
99	1	1069	1074	1059	1063	1068	1072	1062	1066	1071
1 G 3C***	1013	1023	1032	1012	1021	1030	1040	1010	1019	1029
99	1	1038	1048	1023	1032	1042	1051	1035	1045	1055
1 G 3D***	998	1003	1007	1005	1010	1014	1019	1012	1017	1022
A9	1	1027	1032	1029	1034	1039	1044	1047	1052	1057

TABLE 2-40 (CONT'D)

1 G 3E***	1002	998	995	1013	1010	1006	1003	1025	1022	1018
A9	1	1015	1011	1039	1036	1032	1028	1054	1050	1046
1 G 4F***	1030	1021	1012	1040	1031	1022	1013	1051	1042	1032
99	1	1023	1015	1058	1048	1039	1030	1064	1054	1045
1 G 4G***	997	990	983	997	991	984	980	998	992	988
89	1	981	974	997	994	987	980	999	992	986
1 G 4B***	1003	1005	1008	996	999	1002	1004	990	993	995
89	1	998	1001	995	994	997	1000	996	999	1001
1 G 4C***	1041	1050	1060	1037	1047	1057	1067	1034	1044	1054
99	1	1064	1074	1046	1056	1065	1076	1058	1067	1077
1 G 4D***	990	996	1002	996	1002	1008	1014	1002	1008	1014
99	1	1020	1026	1019	1025	1031	1038	1036	1043	1049
1 G 4E***	1014	1012	1010	1026	1024	1021	1019	1038	1036	1033
A9	1	1031	1028	1053	1051	1048	1046	1069	1066	1064
1 G 5F***	1021	1013	1004	1032	1023	1015	1007	1043	1035	1026
99	1	1018	1010	1051	1043	1034	1026	1059	1050	1042
1 G 5G***	1078	1069	1064	1080	1072	1066	1057	1083	1074	1069
99	1	1060	1052	1083	1077	1069	1060	1085	1077	1068
1 G 5B***	1006	1007	1008	1000	1001	1002	1003	994	996	997
89	1	998	999	995	996	997	998	998	1000	1001
1 G 5C***	1051	1060	1067	1046	1055	1065	1071	1042	1051	1060
99	1	1070	1077	1052	1061	1071	1081	1062	1072	1081
1 G 5D***	996	1003	1010	1000	1007	1014	1021	1004	1011	1019
99	1	1026	1033	1021	1028	1036	1043	1038	1046	1054

TABLE 2-40 (CONT'D)

1	G	5E***	1011	1010	1009	1022	1021	1020	1019	1034	1033	1032
	A9	1	1031	1030	1050	1049	1048	1047	1066	1066	1064	
1	G	6F***	1024	1017	1009	1035	1028	1020	1013	1047	1040	1032
	A9	1	1025	1017	1057	1049	1041	1034	1066	1058	1051	
1	G	6G***	1054	1045	1036	1058	1049	1040	1031	1062	1053	1044
	99	1	1035	1026	1063	1054	1045	1036	1063	1054	1045	
1	G	6B***	1078	1077	1077	1072	1072	1071	1070	1070	1070	1069
	99	1	1068	1068	1070	1069	1069	1068	1069	1069	1068	
1	G	6C***	1042	1050	1059	1036	1045	1053	1062	1031	1040	1048
	99	1	1057	1066	1040	1048	1057	1065	1048	1057	1065	
1	G	6D***	1014	1022	1029	1017	1024	1032	1040	1020	1028	1036
	A9	1	1044	1052	1036	1044	1052	1061	1052	1061	1069	
1	G	6E***	993	994	994	1003	1004	1004	1004	1014	1014	1015
	A9	1	1015	1016	1030	1031	1031	1032	1047	1048	1048	
1	G	7F***	986	980	974	996	990	984	979	1008	1002	996
	A9	1	990	984	1019	1012	1006	1000	1029	1022	1016	
1	G	7G***	1040	1031	1022	1046	1036	1028	1019	1051	1042	1033
	A9	1	1024	1016	1053	1044	1035	1026	1055	1046	1037	
1	G	7B***	1045	1043	1040	1040	1038	1036	1034	1036	1034	1032
	99	1	1030	1028	1036	1034	1032	1030	1035	1033	1031	
1	G	7C***	1015	1023	1030	1010	1017	1024	1032	1004	1011	1019
	99	1	1026	1033	1011	1018	1026	1033	1018	1025	1032	
1	G	7D***	989	997	1005	990	998	1006	1015	992	1000	1008
	A9	1	1017	1025	1007	1015	1024	1032	1022	1031	1039	

TABLE 2-40 (CONT'D)

1 G 7E***	962	964	966	972	974	975	977	981	983	985
A9 1	986	988	997	999	1001	1002	1014	1015	1017	
1 H 1F***	1008	996	992	1010	993	990	989	1011	1006	988
C9 1	987	984	1008	990	984	982	1003	986	979	
1 H 1G***	951	951	950	949	949	948	947	947	946	946
B9 1	945	944	943	942	941	940	938	937	935	
1 H 1B***	954	951	948	950	952	949	948	944	947	949
B9 1	949	949	941	944	947	947	926	941	945	
1 H 1C***	1003	1015	1017	987	1005	1007	1008	971	980	996
C9 1	997	997	968	984	996	996	972	985	996	
1 H 1D***	924	921	916	921	920	925	925	925	925	925
D9 1	924	923	925	924	926	925	922	924	925	
1 H 1E***	945	938	922	943	935	928	928	938	931	923
D9 1	925	926	945	930	929	931	942	935	935	
1 H 2F***	964	961	958	966	963	960	957	966	964	961
B9 1	959	956	962	959	959	956	957	957	956	
1 H 2G***	999	1000	1001	994	992	993	991	989	987	982
B9 1	983	983	982	980	973	973	977	970	964	
1 H 2B***	986	990	993	980	984	991	993	975	981	985
B9 1	988	992	971	976	980	982	966	970	974	
1 H 2C***	943	949	953	948	944	949	956	952	949	945
B9 1	952	959	951	948	951	959	951	955	958	
1 H 2D***	1019	1007	995	1022	1011	998	996	1024	1012	1001
C9 1	1009	1006	1016	1015	1013	1009	1017	1014	1011	

TABLE 2-40 (CONT'D)

1 H 2E***	936	950	963	956	936	950	965	963	954	945
B9 1	951	966	959	950	947	964	954	951	948	
1 H 3F***	995	983	973	1001	987	977	970	1007	995	981
B9 1	974	967	1002	988	982	975	998	990	983	
1 H 3G***	1006	1004	1002	1003	998	996	995	1000	995	991
A9 1	989	987	993	989	983	981	987	981	975	
1 H 3B***	1010	1015	1020	1002	1004	1009	1012	994	996	999
A9 1	1001	1004	988	990	994	996	981	985	989	
1 H 3C***	991	997	1006	986	995	1004	1009	981	990	1000
B9 1	1008	1012	982	991	1005	1010	983	994	1007	
1 H 3D***	952	950	947	952	957	953	957	952	956	960
B9 1	964	967	955	959	970	973	958	968	979	
1 H 3E***	952	948	943	956	955	951	947	959	959	958
B9 1	954	950	965	964	964	960	970	970	971	
1 H 4F***	969	964	956	979	967	961	953	989	977	964
A9 1	958	949	984	972	967	959	979	975	970	
1 H 4G***	1014	1014	1014	1017	1013	1012	1004	1019	1015	1010
A9 1	1002	993	1007	1003	988	979	995	980	965	
1 H 4B***	1031	1036	1041	1021	1023	1029	1036	1010	1013	1016
A9 1	1023	1030	1002	1006	1011	1018	995	1001	1006	
1 H 4C***	976	987	998	976	981	992	1001	976	981	986
A9 1	995	1005	977	983	991	1000	979	987	995	
1 H 4D***	963	968	973	968	971	975	984	973	975	977
B9 1	986	996	980	982	984	994	985	990	993	

TABLE 2-40 (CONT'D)

1 H 4E***	958	958	958	963	961	961	961	966	965	963
B9 1	964	964	975	974	970	971	984	980	977	
1 H 5F***	974	970	962	981	975	971	959	986	982	976
A9 1	968	957	984	979	974	966	981	977	972	
1 H 5G***	1027	1016	1004	1028	1021	1009	998	1029	1021	1014
A9 1	1002	991	1021	1013	1011	1000	1012	1010	1008	
1 H 5B***	1045	1046	1047	1034	1034	1035	1038	1022	1022	1023
A9 1	1026	1029	1012	1013	1018	1021	1003	1008	1012	
1 H 5C***	987	1003	1020	985	988	1005	1021	983	987	990
A9 1	1006	1022	982	986	991	1007	981	987	993	
1 H 5D***	943	949	954	945	954	959	967	948	956	967
A9 1	972	978	953	963	970	975	959	966	972	
1 H 5E***	972	970	968	973	973	971	974	972	973	974
B9 1	977	980	978	979	980	983	986	987	988	
1 H 6F***	958	943	930	962	955	940	933	966	959	952
A9 1	944	937	963	956	952	944	960	957	953	
1 H 6G***	1001	996	990	1005	991	986	981	1008	995	981
A9 1	977	972	997	984	973	968	986	975	964	
1 H 6B***	1029	1026	1023	1022	1019	1016	1014	1014	1012	1010
A9 1	1007	1004	1004	1001	997	995	993	989	985	
1 H 6C***	977	989	1001	978	982	994	1001	978	983	987
A9 1	994	1001	972	976	987	994	965	976	987	
1 H 6D***	945	952	958	941	949	955	961	937	945	952
A9 1	958	961	945	953	964	967	953	965	973	

TABLE 2-40 (CONT'D)

1	H	6E***	946	945	945	954	956	956	950	962	964	966
		B9	1	961	955	964	966	969	964	966	969	971
1	H	7F***	956	952	949	966	956	953	948	974	966	957
		B9	1	951	946	977	968	960	955	979	972	964
1	H	7G***	980	971	961	980	973	963	960	979	972	965
		A9	1	962	956	976	968	964	958	971	968	961
1	H	7B***	1006	1005	1005	995	997	996	992	984	986	988
		A9	1	984	980	978	980	979	975	971	971	970
1	H	7C***	970	976	982	965	969	976	983	960	965	969
		A9	1	976	983	960	964	967	974	959	962	965
1	H	7D***	947	955	962	947	955	963	972	948	955	963
		B9	1	972	981	955	963	969	978	962	969	974
1	H	7E***	951	947	943	951	946	942	945	951	946	940
		B9	1	943	946	959	953	948	951	967	961	955
1	H	8F***	942	945	957	945	950	953	958	946	953	958
		B9	1	965	958	955	961	965	972	963	968	972
1	H	8G***	969	964	959	969	968	962	954	970	968	966
		B9	1	957	949	980	978	971	963	989	983	976
1	H	8B***	983	978	973	978	975	970	969	973	970	966
		A9	1	963	962	969	966	966	964	967	967	966
1	H	8C***	979	987	997	971	981	989	997	963	973	983
		B9	1	988	996	971	981	988	993	978	986	993
1	H	8D***	953	954	954	952	958	958	957	950	956	961
		B9	1	961	960	960	965	969	968	969	973	977

TABLE 2-40 (CONT'D)

1	H	8E***	1034	1019	1004	1026	1034	1018	1010	1015	1023	1031
		C9	1	1023	1014	1037	1045	1046	1037	1045	1046	1046
1	J	3F***	993	991	995	997	992	997	1003	1000	1005	999
		C9	1	1006	1011	1002	1005	1001	1007	1003	1000	1003
1	J	3G***	948	948	947	948	947	946	942	948	946	944
		B9	1	929	925	945	931	935	930	929	932	935
1	J	38***	961	957	953	958	958	954	949	943	943	954
		B9	1	950	946	941	941	939	947	938	936	935
1	J	3C***	1033	1029	1032	1029	1022	1026	1025	1024	1019	1010
		C9	1	1020	1018	1018	1011	1015	1014	1011	1008	1011
1	J	3D***	1027	1014	1000	1002	998	982	978	975	969	961
		B9	1	957	960	959	951	955	950	948	943	938
1	J	3E***	995	1007	1017	968	979	992	997	945	949	960
		B9	1	968	974	942	955	952	961	939	945	951
1	J	4F***	1016	1017	1015	1016	1009	1010	1008	1026	1009	1002
		C9	1	1002	992	1021	1004	999	998	1004	1001	995
1	J	4G***	967	965	963	960	962	961	958	954	957	958
		B9	1	956	952	950	951	954	951	944	947	948
1	J	48***	961	966	970	957	959	963	964	952	955	957
		B9	1	957	957	951	954	955	955	949	951	952
1	J	4C***	1000	1005	1020	999	999	1015	1017	998	999	997
		C9	1	1013	1015	997	997	1011	1014	996	1000	1013
1	J	4D***	950	950	947	948	946	943	947	945	942	938
		D9	1	936	941	944	939	945	942	940	946	942

TABLE 2-40 (CONT'D)

1 J 4E***	970	967	955	960	960	953	956	947	948	949
D9 1	948	947	946	948	949	948	953	954	947	
1 J 5F***	964	967	967	964	970	971	972	970	969	973
D9 1	976	968	985	982	983	985	990	992	992	
1 J 5G***	1033	1032	1019	1036	1035	1022	1022	1040	1039	1024
C9 1	1024	1022	1054	1039	1039	1037	1053	1053	1052	
1 J 5B***	952	955	959	957	956	960	961	961	960	959
B9 1	960	962	960	959	975	977	969	974	979	
1 J 5C***	1011	1013	1013	1010	1012	1014	1015	1016	1012	1013
C9 1	1016	1017	1031	1035	1029	1032	1045	1051	1045	
1 J 5D***	1025	1020	1011	1012	1009	1000	1001	998	994	987
D9 1	989	979	996	999	998	988	1002	1000	996	
1 J 5E***	990	1002	1013	975	977	989	995	954	957	960
B9 1	966	972	965	959	963	971	967	971	975	
1 J 5K***	951	955	953	955	954	958	960	946	958	957
B9 1	959	961	960	959	955	958	961	959	954	
1 J 5M***	963	966	963	969	964	961	964	970	965	959
B9 1	962	963	965	959	964	965	960	964	967	
1 J 6F***	957	967	967	958	960	970	970	956	961	963
D9 1	973	973	968	972	974	983	979	983	984	
1 J 6G***	1009	1000	1001	1021	1005	1007	1008	1033	1017	1012
C9 1	1013	1013	1032	1026	1022	1022	1040	1036	1030	
1 J 6B***	959	959	958	960	961	961	957	959	960	962
B9 1	958	955	961	963	960	969	973	971	969	

TABLE 2-40 (CONT'D)

1 J 6C***	1021	1026	1029	1019	1023	1027	1027	1007	1021	1024	
C9 1	1026	1025	1018	1032	1032	1033	1028	1040	1040		
1 J 6D***	970	967	961	972	960	962	970	974	961	955	
D9 1	962	967	976	970	967	971	985	982	976		
1 J 6E***	1004	1001	998	978	981	979	983	947	951	955	
B9 1	960	965	950	954	956	962	959	962	964		
1 J 6K***	956	962	967	958	957	962	963	958	958	957	
B9 1	958	959	955	955	958	959	952	956	960		
1 J 6M***	965	957	949	963	962	954	948	962	960	959	
B9 1	953	947	959	958	959	953	957	958	958		
1 J 7F***	984	978	979	984	985	979	976	983	985	985	
C9 1	975	972	981	983	979	970	979	977	973		
1 J 7G***	946	942	936	939	941	936	932	932	934	935	
B9 1	932	916	929	930	933	918	925	928	919		
1 J 7B***	951	954	958	950	946	949	951	948	944	941	
B9 1	942	944	939	935	936	937	929	929	930		
1 J 7C***	941	944	947	941	942	945	944	930	942	943	
B9 1	942	940	926	938	939	938	922	934	935		
1 J 7D***	993	995	1004	989	993	1002	1004	985	989	1000	
C9 1	1002	1002	977	988	995	995	975	983	988		
1 J 7E***	936	936	937	939	937	938	941	931	930	928	
D9 1	932	936	928	926	928	932	922	924	926		
1 J 8F***	937	946	952	947	948	948	953	944	948	949	
B9 1	948	954	955	959	957	955	966	966	964		

TABLE 2-40 (CONT'D)

1 J 8G***	1006	1009	1009	1012	1004	1005	1006	1018	1010	1001	
C9 1	1002	993	1022	1013	1013	1004	1025	1025	1015		
1 J 8B***	952	950	947	951	949	946	950	948	946	944	
B9 1	948	952	960	947	941	944	959	953	947		
1 J 8C***	984	989	1005	988	990	996	1011	990	994	997	
C9 1	1002	1017	1001	1006	1008	1013	1012	1016	1019		
1 J 8D***	938	941	933	942	945	937	936	946	949	941	
D9 1	941	938	961	955	952	950	968	966	962		
1 J 8E***	958	961	965	943	944	947	947	924	925	926	
B9 1	925	923	930	931	928	927	942	940	938		
1 P 1F***	997	1017	1040	976	993	1013	1045	952	972	984	
B9 1	1019	1048	966	978	1007	1038	977	1002	1027		
1 P 1G***	908	908	906	913	922	920	927	916	924	931	
B9 1	937	942	942	947	960	964	961	973	983		
1 P 1B***	915	907	897	932	917	910	905	945	932	918	
B9 1	917	913	964	952	938	938	981	970	957		
1 P 1C***	893	866	831	902	867	834	812	910	875	837	
69 1	816	791	886	849	828	805	861	841	817		
1 P 1D***	1013	1008	1003	983	966	962	970	957	940	923	
69 1	927	930	934	912	901	904	900	889	877		
1 P 1E***	969	987	1007	926	940	965	984	881	896	910	
69 1	939	960	874	889	901	931	866	879	892		
1 P 2F***	1027	1052	1068	1002	1021	1038	1072	977	998	1014	
B9 1	1044	1075	995	1013	1036	1070	1011	1042	1064		

TABLE 2-40 (CONT'D)

1 P 2G***	950	957	960	953	951	956	983	946	945	951	
B9 1	981	1004	940	939	970	996	925	959	987		
1 P 2B***	907	906	912	927	917	915	915	942	935	926	
B9 1	928	928	965	959	948	951	986	978	969		
1 P 2D***	1031	1005	975	1012	983	948	932	981	951	920	
69 1	903	886	946	914	904	886	907	896	884		
1 P 2F***	998	1032	1061	952	964	999	1021	901	916	929	
69 1	953	976	895	911	926	950	887	906	923		
1 P 2J***	928	941	946	938	935	949	956	946	944	942	
D9 1	950	965	950	949	954	962	953	959	965		
1 P 3F***	1039	1017	993	1070	1040	1019	1014	1091	1069	1040	
B9 1	1037	1033	1098	1071	1065	1063	1104	1095	1084		
1 P 3B***	973	971	967	976	970	976	966	983	979	974	
B9 1	965	964	1006	1004	994	986	1028	1021	1012		
1 P 3D***	1099	1065	1030	1064	1032	996	979	1028	995	961	
69 1	942	922	992	957	939	917	951	931	910		
1 P 3E***	1033	1064	1094	984	1002	1034	1069	930	952	971	
69 1	1008	1044	925	948	963	1001	918	936	953		
1 P 3K***	1090	1058	1025	1048	1026	992	975	1004	982	958	
69 1	941	921	970	947	938	917	933	923	912		
1 P 3L***	993	1030	1063	948	970	1007	1036	901	924	945	
69 1	976	1004	899	924	942	972	895	917	937		
1 P 3N***	935	936	927	972	941	933	931	1003	976	946	
B9 1	937	926	1005	979	955	947	1006	985	963		

TABLE 2-40 (CONT'D)

1 P 4F***	930	939	947	904	899	909	922	873	869	864
69 1	879	893	867	858	859	875	856	857	857	
1 P 4G***	942	944	946	974	956	959	971	1000	985	968
89 1	980	993	1016	1001	996	1010	1030	1026	1021	
1 P 4B***	934	927	929	947	942	936	933	956	953	949
89 1	947	944	978	976	974	973	997	997	995	
1 P 4C***	1060	1045	1023	1063	1034	1018	1000	1064	1038	1012
89 1	996	976	1059	1030	1013	995	1052	1030	1012	
1 P 4D***	1039	1021	992	1015	991	962	945	987	959	930
69 1	914	896	950	921	913	894	912	903	891	
1 P 4E***	1023	1033	1045	995	987	998	1018	958	949	940
69 1	966	992	935	926	927	953	911	913	913	
1 P 4H***	956	961	966	965	970	976	973	971	977	982
D9 1	981	979	985	991	1001	1001	1007	1009	1010	
1 P 4L***	954	977	1000	918	927	951	970	875	886	896
69 1	917	938	872	883	891	914	866	875	884	
1 P 5F***	926	929	931	894	900	904	913	856	863	869
69 1	881	891	842	849	864	876	830	846	856	
1 P 5G***	954	949	944	982	962	958	953	1004	986	967
89 1	963	959	1014	997	977	974	1022	1004	985	
1 P 5C***	1032	1022	1009	1038	1018	1007	994	1048	1023	1003
89 1	993	978	1050	1031	1011	1000	1051	1039	1018	
1 P 5D***	1022	985	947	995	972	933	915	968	944	920
69 1	900	876	943	919	910	885	917	907	994	

TABLE 2-40 (CONT'D)

1 P 5E***	1026	1032	1038	994	998	1005	1016	949	953	957	
69 1	973	990	929	933	942	959	907	917	926		
1 P 5J***	955	957	958	975	972	974	988	990	988	987	
B9 1	1000	1014	1010	1009	1014	1028	1023	1029	1034		
1 P 5K***	1003	968	930	990	953	915	902	975	938	899	
69 1	885	867	944	905	885	867	911	890	867		
1 P 5L***	1017	1023	1028	988	983	988	1001	950	946	942	
69 1	958	972	927	922	923	939	902	904	904		
1 P 6F***	892	921	946	879	888	917	944	863	875	884	
69 1	915	942	866	878	893	924	869	887	901		
1 P 6G***	933	960	982	937	951	973	1006	940	954	964	
B9 1	998	1026	971	981	1001	1030	997	1017	1033		
1 P 6B***	946	946	946	962	961	961	959	968	967	966	
D9 1	964	962	996	995	993	991	1019	1018	1016		
1 P 6C***	1005	987	964	1022	995	976	961	1037	1013	985	
B9 1	973	957	1036	1011	996	984	1034	1022	1006		
1 P 6D***	936	916	894	937	909	887	881	938	910	879	
69 1	873	865	924	893	876	868	907	891	871		
1 P 6E***	1018	1021	1024	989	983	986	999	948	941	935	
69 1	948	961	931	924	926	939	913	915	917		
1 P 7F***	916	947	976	880	896	927	958	837	858	875	
69 1	909	940	841	862	875	908	845	862	875		
1 P 7G***	937	950	960	938	942	952	977	938	943	944	
B9 1	970	988	960	961	974	992	977	990	996		

TABLE 2-40 (CONT'D)

1	P 7C***	930	927	925	956	945	942	949	979	967	955
	B9 1	962	967	992	981	979	985	1002	1001	999	
1	P 7D***	914	912	910	887	887	886	896	858	858	858
	69 1	868	879	846	845	852	862	830	836	842	
1	P 7E***	997	994	990	967	961	958	958	934	926	919
	69 1	917	914	907	899	894	892	878	873	869	
1	P 7H***	937	936	935	946	947	945	946	944	946	946
	D9 1	947	948	975	976	972	973	1000	997	993	
1	P 7L***	999	998	996	975	966	964	977	938	928	919
	69 1	932	945	919	909	909	922	899	899	898	
1	P 8F***	911	946	979	877	893	928	959	835	856	874
	69 1	907	938	840	862	878	911	845	865	882	
1	P 8G***	922	933	942	919	932	942	961	916	930	941
	B9 1	960	983	937	948	971	989	955	978	993	
1	P 8B***	941	937	934	934	928	924	925	918	920	922
	B9 1	922	922	945	946	944	944	966	964	962	
1	P 8C***	921	916	908	945	925	921	918	963	948	929
	B9 1	929	927	974	959	952	953	984	980	974	
1	P 8D***	901	866	824	906	870	829	812	911	876	834
	69 1	817	801	892	852	832	817	869	851	828	
1	P 8E***	999	995	992	970	957	953	963	936	919	901
	69 1	911	921	909	891	889	900	881	880	877	
1	P 8K***	943	949	963	946	947	954	971	946	949	950
	D9 1	961	970	949	952	958	969	943	959	965	

TABLE 2-40 (CONT'D)

1	P	8M***	930	932	924	944	933	925	928	958	947	925
		D9	1	930	930	958	945	932	933	956	942	936

TABLE 2-41
R-1 COMPOSITE CORE REACTOR, FRACTIONAL FISSION POWER PERTURBATION
PER DOLLAR REACTIVITY, AT CORE EDGE

Component	Fractional Fission Power Change Per Dollar Reactivity at the Core Edge		
	Calculated With No Control Drum Bank Representation	Calculated With Control Drum Bank Representa- tion Worth of \$2.77	Extrapolated To Control Drum Bank Representation at Nom- inal Hot Critical, \$5.59
Outward Drum Rotation	+ 0.09103		
Seal Hydrogen	+ 0.3166	+ 0.3278	+ 0.3393
Cylinder Hydrogen	+ 0.4998	+ 0.4431	+ 0.3852
Reflector Hydrogen	+ 1.567	+ 1.676	+ 1.787
Reflector Temperature	+ 0.01404	+ 0.03767	+ 0.06180
Core Temperature	- 0.001436	- 0.0006557	~ 0.0
Core Expansion	+ 0.01866	+ 0.01479	+ 0.01084
Cylinder Contraction	+ 0.02597	+ 0.03672	+ 0.04770
Core Hydrogen	- 0.01231	- 0.01052	- 0.008698

079 10/17/70
100 10/17/70
083 10/17/70
084 10/17/70
525 10/23/70
522 10/23/70
086 10/17/70
093 10/17/70

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TABLE 2-42
DISTRIBUTION OF FISSION POWER IN FUEL CLUSTER

Fission Densities Relative to Channel Row Number Three									
Channel Row	R-1**	PAX/G2 Experimental Results(Unshimmed)				PAX/G2 Calculations			
		Dry	With Poly*	Min. Flow*	Max. Flow*	Min. Flow No Shim	Shim†	Max. Flow No Shim	Shim†
1	1.018	1.016	1.047	1.021	1.033	1.013	1.012	1.024	1.021
2	1.009	--	--	--	--	1.005	1.004	1.009	1.008
3	1.000	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
4	0.9965	--	--	--	--	0.9956	0.9961	0.9928	0.9925
5	0.9935	0.9941	0.9880	0.9932	0.9854	0.9917	0.9926	0.9870	0.9870

- * "With Poly" measurements provided 4.52 gm H in support stem, "Min, Flow" and "Max Flow" data are linear interpolations between "Dry" measurements and "With Poly" measurements. Min. Flow assumed 0.672 gm H and Max. Flow assumed 2.34gm H in the stem to agree with the PAX calculations
- + "Shim" calculations include a TaC shim tube of a nominal 3 mil coating thickness, approximately 3l gms. TaC
- ** Assumed gradient for preliminary loading distribution described in Section 2.4.2.

TABLE 2-43
R-1 COMPOSITE CORE REACTOR
FRACTIONAL FISSION POWER PERTURBATION PER DOLLAR REACTIVITY
AT CORE EDGE

<u>Fractional Fission Power Change Per Dollar Reactivity</u>		
<u>Component</u>	<u>Calculated With No Control Drum Bank Representation</u>	<u>Calculated With Control Drum Bank Representation Worth of \$2.77.</u>
Outward Drum Rotation		+ 0.09103
SSA Flow	- 0.01231	- 0.01052
Corrosion	+ 0.008681	+ 0.008846
Fuel Burnup	- 0.002075	- 0.002618
Fission Product Poisoning	- 0.01027	- 0.01332

086	10/17/70
093	10/17/70
082	10/20/70
081	10/20/70

TABLE 2-44

R-1 COMPOSITE CORE REACTOR, THROTTLED MODE OF OPERATION

Component	Operating Mode Start of Life	Throttled Mode Start of Life	Reactivity Change, β	Fractional Fission Power Change Per Dollar Reactivity at Core Edge, at Nominal Hot Critical	Fractional Fission Power Change at Core Edge, at Nominal Hot Critical In Going From Operating to Throttle Mode
Seal Hydrogen	0.292 lbs.	0.188 lbs.	- 0.055	+ 0.3393	- 0.01866
Cylinder Hydrogen	0.405 lbs.	0.214 lbs.	- 0.07	+ 0.3852	- 0.02696
Reflector Hydrogen	1.208 lbs.	0.660 lbs.	- 0.017	+ 1.787	- 0.03038
Reflector Temperature	252° R	283° R	~ 0.0	+ 0.0618	~ 0
TOTAL					- 0.076

158 10/2/70
100 10/17/70

TABLE 2-45
SUMMARY RESULTS FOR NERVA FISSION PRODUCT POISONING
AND FUEL BURNUP CHARACTERISTICS

I. FISSION PRODUCTS

<u>Design</u>	<u>Reactivity Effects</u>	
	<u>Xenon Buildup</u>	<u>Stable Fission Products Including Samarium -149</u>
A. Composite Fuel	-10.9 Cents	-1.2 Cents
B. Graphite Fuel	-12.0 Cents	-1.3 Cents

II. FUEL BURNUP

<u>Design</u>	<u>Reactivity Loss For 10 Hrs. Operation</u>
A. Composite Fuel	-11 Cents
B. Graphite Fuel	-12 Cents

TABLE 2-46
SUMMARY OF NERVA CONTROL REQUIREMENTS

<u>Design Item Identification</u>	<u>Reactivity Requirement For Composite Fuel</u>	<u>Reactivity Requirement For Graphite Fuel</u>
1. Corrosion Compensation	\$1.00	\$1.00
2. Fuel Burnup and Fission Product Poisoning	\$0.23	\$0.25
3. Cold-To-Hot Reactivity Feedback	\$1.15	\$0.80
4. Ambient and Operating Uncertainties		
4a. NERVA Flight Engine		
4.a.1 Cold Critical Uncertainties	+ \$0.15	+ \$0.15
4.a.2 Hot Operating Uncertainties	+ \$0.35	+ \$0.35
4b. R-1		
4.b.1 Cold Critical Uncertainties	+ \$0.15	+ \$0.15
4.b.2 Hot Operating Uncertainties	+ \$0.80	+ \$0.80
5. Safety Shutdown Requirements		
5a. Non-Operating	\$1.5	\$1.5
5b. Operating	\$0.25	\$0.25
6. Operating Mode Requirements		
6a. Dry Startup	\$0.70	\$0.70
6b. Thermal Startup	\$0.20	\$0.20
6c. Controllability Margin	\$0.10	\$0.10
7. Space Storage	0.0	0.0
8. Cold Flow Testing	0.0 (\$1.25 Max. Available)	0.0 (\$1.25 Max. Available)
9. Actuator Malfunction Allowance	\$0.25	\$0.25

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TABLE 2-47
RELATIVE CONTRIBUTIONS TO THE CORE NEUTRONIC TEMPERATURE
REACTIVITY FEEDBACK IN THE COMPOSITE R-1 REACTOR *

<u>Effect</u>	<u>Reactivity Feedback at 2880° R Core Temperature</u>
Uranium Resonance Doppler Broadening	- \$0.085
Zirconium Resonance Doppler Broadening	- \$0.269
Thermal Neutron Energy Base Shift	- \$0.059
 Total Core Neutronic Feedback to 2880° R	 - \$0.413

* Calculated with control drums removed.

106 10/17/70
081 10/17/70
082 10/17/70
080 10/17/70
083 10/17/70

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TABLE 2-48
DEVELOPMENT OF REACTIVITY WEIGHTED EFFECTIVE CYLINDER
DISPLACEMENTS FOR THE COMPOSITE R-1 REACTOR

L=Core Station (in.)	ΔL =Interval (in.)	$W=\sin \frac{\pi L}{52}$	Displacements (in.)	
			$D_{\text{SOL}}^{\text{Rated Full Power SOL}}$	$D_{\text{ST. UP}}^{\text{Start Up Hold}^+}$
1	2	.060	-.060*	-.0695*
4	4	.239	-.072	-.074
8	4	.465	-.075	-.076
12	4	.663	-.075	-.076
16	4	.823	-.075	-.077
20	4	.935	-.076	-.078
24	4	.993	-.077	-.080
28	4	.993	-.080	-.083
32	4	.935	-.082	-.085
36	4	.823	-.085	-.087
40	4	.663	-.088	-.089
44	4	.465	-.092	-.092
48	4	.239	-.096	-.094
51	2	.060	-.0775**	-.088**

$$\sum_i W_i \Delta L_i = 33.184, \sum_i D_i^{\text{SOL}} W_i \Delta L_i = -2.6612, \sum_i D_i^{\text{ST. UP}} W_i \Delta L_i = -2.7265$$

In general, $D_{\text{EFF}} = \left(\sum_i D_i W_i \Delta L_i \right) / \left(\sum_i W_i \Delta L_i \right)$

Then,

$$D_{\text{EFF}}^{\text{SOL}} = -2.6612/33.184 = -.080, D_{\text{EFF}}^{\text{ST. UP}} = -2.7265/33.184 = -.082$$

⁺ 40 seconds into the SOL Start Up

* Extrapolated from displacement data of TABLE 2-23

** Interpolated from displacement data of TABLE 2-23

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TABLE 2-49
DEVELOPMENT OF REACTIVITY WEIGHTED EFFECTIVE CORE
DISPLACEMENTS FOR THE COMPOSITE R-1 REACTOR

L=Core Station (in.)	ΔL =Interval (in.)	$W=\sin \frac{\pi L}{52}$	Displacements (in.)	
			$D_{\text{SOL}}^{\text{Rated Full Power SOL}}$	$D_{\text{ST.UP}}^{\text{Start Up Hold}^+}$
1.25	2.5	.076	-.0025*	-.0025*
5	5	.297	.05	.02
10	5	.568	.10	.05
15	5	.787	.14	.07
20	5	.935	.17	.09
25	5	.998	.20	.11
30	5	.971	.24	.13
35	5	.856	.26	.14
40	5	.663	.28	.15
45	5	.410	.29	.15
49.75	4.5	.135	.29	.14

$$\sum_i W_i \Delta L_i = 33.223, \quad \sum_i D_i^{\text{SOL}} W_i \Delta L_i = 6.6783, \quad \sum_i D_i^{\text{ST.UP}} W_i \Delta L_i = 3.5365$$

In general, $D_{\text{EFF}} = \left(\sum_i D_i W_i \Delta L_i \right) / \left(\sum_i W_i \Delta L_i \right)$

Then,

$$D_{\text{EFF}}^{\text{SOL}} = 6.6783/33.223 = 0.201 \text{ in.} = 1.03\%^{**}$$

$$D_{\text{EFF}}^{\text{ST.UP}} = 3.5365/33.223 = 0.106 \text{ in.} = 0.54\%^{**}$$

⁺ 40 seconds into the SOL Start Up (2520° R Chamber Temperature)

* Interpolated From Displacement Data of TABLE 2-16

** Based on an ambient core radius of 19.48 inches

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TABLE 2-50
ADDITIVITY OF REACTIVITY FEEDBACK EFFECTS
IN THE COMPOSITE R-1 REACTOR WITHOUT CONTROL DRUMS

<u>Effect</u>	<u>Calculated Feedback (\$)</u>	<u>Summed Feedback (\$)</u>	<u>Difference</u>
H_c - 1.4 lbs. core hydrogen	+ 2.98		
T_c - 2880° R core temperature	- 0.41		
$H_c + T_c$	+ 2.44	+ 2.57	+ 0.13
H_s - 0.29 lbs. seal hydrogen	+ 0.16		
H_{cyl} - 0.41 lbs. cylinder hydrogen	+ 0.12		
H_R - 1.22 lbs. reflector hydrogen	+ 0.04		
$H_p = H_s + H_{cyl} + H_R$	+ 0.33	+ 0.32	- 0.01
T_R - 139.3° R beryllium temperature	- 0.17		
$H_p + T_R$	+ 0.12	+ 0.16	+ 0.04
$H_c + T_R$	+ 2.87	+ 2.81	- 0.06
$H_c + (H_p + T_R)$	+ 3.11	+ 3.10	- 0.01
$T_c + H_p$	- 0.07	- 0.08	- 0.01
$T_c + (H_p + T_R)$	- 0.35	- 0.29	+ 0.06

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TABLE 2-51
ADDITIVITY OF CONTROL DRUM INTERACTION
EFFECTS IN THE COMPOSITE R-1 REACTOR

<u>Effect</u>	<u>Calculated Drum Worth Change (%)</u>	<u>Summed Drum Worth Change (%)*</u>	<u>Difference</u>
H_c - 1.4 lbs. core hydrogen	- 6.5		
T_c - 2880° R core temperature	- 1.6		
$H_c + T_c$	not calculated	- 8.0	
H_s - 0.29 lbs. seal hydrogen	- 0.1		
H_{cyl} - 0.41 lbs. cylinder hydrogen	- 0.8		
H_R - 1.22 lbs. reflector hydrogen	+ 0.2		
$H_p = H_s + H_{cyl} + H_R$	- 0.6	- 0.7	- 0.1
T_R - 139.3° R beryllium temperature	- 4.0		
$H_p + T_R$	- 5.4	- 4.6	+ 0.8
$H_c + T_R$	- 9.0	- 10.2	- 1.2
$H_c + (H_p + T_R)$	- 10.8	- 11.5	- 0.7
$T_c + H_p$	not calculated	- 2.2	
$T_c + (H_p + T_R)$	not calculated	- 6.9	

* Drum worth changes are fractionally multiplied, i.e., in the case of core hydrogen and core temperature the "summed" result in $(1 - .065)(1 - .016) - 1 = -.08$.

TABLE 2-52

DEVELOPMENT OF CORE HYDROGEN FEEDBACK WITH SHIM TUBE
INTERACTION EFFECTS TO NOMINAL RATED FULL POWER
START OF LIFE IN THE COMPOSITE R-1 REACTOR

	<u>Homogeneous Calc. With \$3.0 Shim</u>	<u>Homogeneous Calc. With \$1.50 Shim</u>	<u>Heterogeneous With Stem Interaction Effects</u>
Propellant	\$1.08	\$1.13	\$1.14
SSA at Min. Stem Flow	1.12	1.17	1.11
SSA Hot Critical Reactivity Adjustment	<u>0.45</u>	<u>0.45</u>	<u>0.45</u>
Total Core Hydrogen Feedback	\$2.65	\$2.75	\$2.70

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TABLE 2-53

DEVELOPMENT OF THE REACTIVITY FEEDBACK
NOMINAL RATED TO FULL POWER START OF LIFE
FOR THE COMPOSITE R-1 REACTOR

Operating Change	Magnitude	Feedback	Modified Per This Section
		as Calculated Per Section 2.5.1.1,2.1	
Propellant Hydrogen	0.464 lb *	+\$1.08	+\$1.14
Support Stem Hydrogen	0.671 lb *	+\$1.57	+\$1.56
Core Neutronic Temperature	3173°R	-\$0.34	- \$0.60
Core Expansion	1.03%	-\$1.50	As Calculated
Seal Hydrogen	0.277 lb	+\$0.15	
Cylinder Hydrogen	0.416 lb	+\$0.15	
Reflector Hydrogen	1.24 lb	+\$0.03	
Reflector Temperature	245°R	0	
Cylinder Contraction	-0.08 in. (Radial)	+\$0.15	
Total Cold-to-Hot Feedback		+\$1.29	+\$1.08

*These are reactivity weighted effective masses as indicated from TNT run TNTMI58.

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TABLE 2-54

DEVELOPMENT OF THE CONTROL DRUM WORTH CHANGE
TO NOMINAL RATED FULL POWER START OF LIFE FOR THE
COMPOSITE R-1 REACTOR

<u>Operating Change</u>	<u>Magnitude</u>	Drum Worth Change	
		<u>As Calculated</u> <u>Per Section 2.5.1.1.2.1</u>	<u>Modified</u> <u>Per This Section</u>
Propellant Hydrogen	0.464 lb *	-2.2%	-2.3%
Support Stem Hydrogen	0.671 lb *	-3.2%	-3.2%
Core Neutronic Temperature	3173°R	-1.7%	-3.0%
Core Expansion	1.03%	+5.5%	As Calculated
Seal Hydrogen	0.277 lb	0	
Cylinder Hydrogen	0.416 lb	-0.8%	
Reflector Hydrogen	1.24 lb	0	
Reflector Temperature	245°R	-2.4%	
Cylinder Contraction	-0.08 in. (Radial)	+1.7%	
		3.3%	-4.7%

* Reactivity Weighted Effective Masses

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TABLE 2-55

NUCLEAR AND RADIATION ENVIRONMENTAL DRAWING
Nuclear Data - Composite Core (Sheet 3)

I Major Nuclear Parameters

1. Control Drum Span (Ambient Conditions) = (\$4.50 Required, \$5.40 Calculated), 80% Probability, 95% Confidence
2. Reduction in Drum Span (Ambient-to-Operating Conditions) = 6%
3. Nominal Cold Critical Drum Position = 80°
4. Nominal Cold-to-Hot Reactivity Change = \$1.08
5. Nominal Hot Critical Drum Position (Orifice Angle, SSCV Flow = 7.1 lb/sec) = 47°
6. Minimum Shutdown Reactivity from Cold Critical = \$1.70
7. Support Stem Reactivity Swing (Min. SSCV Setting at Full Power to Maximum SSCV Setting Throttled) = \$2.69

II Critical Control Drum Angles

Power Test Identification	Full Power Operating Time, sec	Cold Start of Life	Hot Start of Life (100% Power)	Hot Start of Life (60% Power, Throttled)	Hot End of Life (100% Power)	Hot End of Life (60% Power, Throttled)	Cold End of Life	Cold End of Life (11 Hrs after Shutdown)	Cooldown and Space Storage
Initial Power Run	3600	80°	47°	47°	47°	47°	82.5°	82.5°	55°
1st Restart									
2nd Restart									

III Power Distribution (Without Drum Movement)

Power Test Identification	Hot Start of Life (100% Power)				Hot End of Life (100% Power)			
	Max Power Factor in Core	Min Power Factor in Core	Max Peripheral Power Factor	Min Peripheral Power Factor	Max Power Factor in Core	Min Power Factor in Core	Max Peripheral Power Factor	Min Peripheral Power Factor
Initial Power Run	1.10	0.70	1.0	0.70	1.10	0.70	1.0	0.70
1st Restart								
2nd Restart								

Power Test Identification	Hot Start of Life (60% Power, Throttled)				Hot End of Life (60% Power, Throttled)			
	Max Power Factor in Core	Min Power Factor in Core	Max Peripheral Power Factor	Min Peripheral Power Factor	Max Power Factor in Core	Min Power Factor in Core	Max Peripheral Power Factor	Min Peripheral Power Factor
Initial Power Run	1.10	0.64	0.92	0.64	1.10	0.64	0.92	0.64
1st Restart								
2nd Restart								

TABLE 2-56

FEEDBACK REACTIVITY UTILIZATION

<u>Stem Flow Condition</u>	<u>Feedback Reactivity Utilization</u>	<u>Cumulative Feedback Reactivity Utilization</u>
Minimum Flow	\$ 0.63	\$ 0.63
Nominal Flow, SOL	0.45	1.08
Compensation for Cold-to-Hot Uncertainty	0.35	1.43
Compensation for Corrosion Losses, EOL	1.00	<u>2.43</u>
Total Feedback Reactivity Utilization:		\$ 2.43

TABLE 2-57
 PRELIMINARY NERVA MULTIZONED FUEL DISTRIBUTION - GRAPHITE

FUEL LOADING CODE	NUMBER OF FUELED ELEMENTS	LOADING**** MG/CC OF FUELED GRAPHITE	TOTAL URANIUM 51.63 INCHES CM	TOTAL* CARBON 51.63 INCHES CM	TOTAL** ZIRCONIUM 51.63 INCHES CM	TOTAL*** NIOBIUM 51.63 INCHES CM
11	54	380	107.8	532.2	73.5	42.7
12	114	392	111.2	531.8	73.5	42.7
13	102	412	116.9	531.2	73.5	42.7
14	72	424	120.3	530.7	73.5	42.7
15	96	436	123.7	530.2	73.5	42.7
16	276	475	134.8	528.7	73.5	42.7
17	96	488	138.4	528.2	73.5	42.7
18	96	502	142.4	527.6	73.5	42.7
19	150	550	156.0	525.6	73.5	42.7
20	235.3	567	160.9	525.0	73.5	42.7
21	388.7	584	165.7	524.2	73.5	42.7
22	102	600	170.2	523.5	73.5	42.7
23	96	630	178.7	522.2	73.5	42.7

* Includes 4.0 grams/element from bore deposition.

** Includes 59.6 grams/element from bore deposition and 13.9 grams/element from external coating.

*** Includes 40.7 grams/element from bore deposition and 2.0 grams/element from external coating.

**** Loadings do not include the additional 30 CM/CC to allow for shimming by the additional element selection procedure.

TABLE 2-58

R-1 GRAPHITE CORE REACTOR, FRACTIONAL FISSION POWER PERTURBATION
PER DOLLAR REACTIVITY, AT CORE EDGE

Fractional Fission Power Change Per Dollar Reactivity at the Core Edge

<u>Component</u>	<u>Calculated With No Control Drum Bank Representation</u>	<u>Calculated With Control Drum Bank Representa- tion Worth of \$2.58</u>	<u>Extrapolated To Control Drum Bank Representation at Nom- inal Hot Critical, \$5.16</u>
Outward Drum Rotation	+ 0.09725		
Seal Hydrogen	+ 0.3166	+ 0.3278	+ 0.3393
Cylinder Hydrogen	+ 0.4998	+ 0.4431	+ 0.3852
Reflector Hydrogen	+ 1.567	+ 1.676	+ 1.787
Reflector Temperature	+ 0.01404	+ 0.03767	+ 0.06180
Core Temperature	- 0.03712	- 0.03017	- 0.02323
Core Expansion	+ 0.01245	+ 0.01204	+ 0.01163
Cylinder Contraction	+ 0.03064	+ 0.04107	+ 0.05148
Core Hydrogen	- 0.01112	- 0.009444	- 0.007766

085 10/17/70
092 10/17/70
079 10/17/70
100 10/17/70
112 10/17/70
102 10/17/70
527 10/23/70
524 10/23/70

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TABLE 2-59

R-1 GRAPHITE CORE REACTOR
FRACTIONAL FISSION POWER PERTURBATION PER DOLLAR REACTIVITY
AT CORE EDGE

Fractional Fission Power Change Per Dollar Reactivity

<u>Component</u>	<u>Calculated with no Control Drum Bank Representation</u>	<u>Calculated with Control Drum Bank Representation Worth of \$2.58</u>
Outward Drum Rotation		+0.09725
SSA Flow	-0.01112	-0.009444
Corrosion	+0.009345	+0.009304
Fuel Burnup	-0.003195	-0.004275
Fission Product Poisoning	-0.01325	- 0.03212

085 10/17/70
092 10/17/70
084 10/20/70
083 10/20/70

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TABLE 2-60

RELATIVE CONTRIBUTIONS TO THE CORE NEUTRONIC TEMPERATURE
REACTIVITY FEEDBACK IN THE GRAPHITE R-1 REACTOR*

Effect	Reactivity Feedback at 2880° R Core Temperature
Uranium Resonance Doppler Broadening	- \$0.223
Niobium Resonance Doppler Broadening	- \$0.173
Thermal Neutron Energy Base Shift	- \$0.057
Total Core Neutronic Feedback to 2880° R	- \$0.453

*Calculated with control drums removed.

089 10/17/70
088 10/17/70
097 10/17/70
087 10/17/70
112 10/17/70

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TABLE 2-61
DEVELOPMENT OF REACTIVITY WEIGHTED EFFECTIVE CYLINDER
DISPLACEMENTS FOR THE GRAPHITE R-1 REACTOR

L=Core Station (in.)	ΔL =Interval (in.)	$W=\sin \frac{\pi L}{52}$	Displacements (in.)	
			$D_{\text{Rated Full Power SOL}}^{\text{SOL}}$	$D_{\text{Start Up Hold}}^{\text{ST. UP}}+$
1	2	.060	-.057*	-.0675*
4	4	.239	-.069	-.072
8	4	.465	-.072	-.073
12	4	.663	-.071	-.074
16	4	.823	-.071	-.074
20	4	.935	-.072	-.076
24	4	.993	-.074	-.078
28	4	.993	-.076	-.081
32	4	.935	-.079	-.083
36	4	.823	-.083	-.086
40	4	.663	-.086	-.088
44	4	.465	-.090	-.091
48	4	.239	-.095	-.094
51	2	.060	-.077**	-.0885**

$$\sum_i W_i \Delta L_i = 33.184, \quad \sum_i D_i^{\text{SOL}} W_i \Delta L_i = -2.5581, \quad \sum_i D_i^{\text{ST. UP}} W_i \Delta L_i = -2.6650$$

$$\text{In general, } D_{\text{EFF}} = \left(\sum_i D_i W_i \Delta L_i \right) / \left(\sum_i W_i \Delta L_i \right)$$

$$\text{Then, } D_{\text{EFF}}^{\text{SOL}} = -2.5581/33.184 = -.077, \quad D_{\text{EFF}}^{\text{ST. UP}} = -2.6650/33.184 = -.080$$

+ 39.5 seconds into the SOL Start Up

* Extrapolated from displacement data of TABLE 2-18

** Interpolated from displacement data of TABLE 2-18

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TABLE 2-62
DEVELOPMENT OF REACTIVITY WEIGHTED EFFECTIVE CORE
DISPLACEMENTS FOR THE GRAPHITE R-1 REACTOR

Core Station (in.)	$\Delta L = \text{Interval (in.)}$	$W = \sin \frac{\pi L}{52}$	Displacements (in.)	
			$D^{\text{SOL}}_{\text{Rated Full Power SOL}}$	$D^{\text{ST. UP}}_{\text{Start Up Hold}^+}$
1.25	2.5	.076	+.0025*	-.0025*
5	5	.297	.04	.02
10	5	.568	.07	.03
15	5	.787	.10	.05
20	5	.935	.13	.07
25	5	.998	.16	.08
30	5	.971	.19	.10
35	5	.856	.22	.11
40	5	.663	.24	.11
45	5	.410	.24	.11
49.75	4.5	.135	.25	.10

$$\sum_i W_i \Delta L_i = 33.223, \quad \sum_i D_i^{\text{SOL}} W_i \Delta L_i = 5.4912, \quad \sum_i D_i^{\text{ST. UP}} W_i \Delta L_i = 2.6448$$

$$\text{In general, } D_{\text{EFF}} = \left(\sum_i D_i W_i \Delta L_i \right) / \left(\sum_i W_i \Delta L_i \right)$$

$$\text{Then, } D_{\text{EFF}}^{\text{SOL}} = 5.4912/33.223 = 0.165 \text{ in.} = 0.85\% \text{ **}$$

$$D_{\text{EFF}}^{\text{ST. UP}} = 2.6448/33.223 = 0.080 \text{ in.} = 0.41\% \text{ **}$$

+ 39.5 seconds into the SOL Start Up (2433° R Chamber Temperature)

* Interpolated From Displacement Data of TABLE 2-17

** Based on an ambient core radius of 19.48 inches

TABLE 2-63
ADDITIVITY OF REACTIVITY FEEDBACK EFFECTS
IN THE GRAPHITE R-1 REACTOR WITHOUT CONTROL DRUMS

<u>Effect</u>	<u>Feedback (\$)</u>	<u>Feedback (\$)</u>	<u>Difference</u>
H_c - 1.4 lbs. core hydrogen	+ 3.04		
T_c - 2880° R core temperature	- 0.48		
$H_c + T_c$	+ 2.43	+ 2.56	+ 0.13
H_s - 0.29 lbs. seal hydrogen	+ 0.16		
H_{cyl} - 0.41 lbs. cylinder hydrogen	+ 0.12		
H_R - 1.22 lbs. reflector hydrogen	+0.04		
$H_p = H_s + H_{cyl} + H_R$	+ 0.33	+ 0.32	- 0.01
T_R - 139.3° R beryllium temperature	- 0.17		
$H_p + T_R$	+ 0.12	+ 0.16	+ 0.04
$H_c + T_R$	+ 2.87	+ 2.81	- 0.06
$T_c + H_p$	- 0.21	- 0.15	+ 0.06
$T_c + (H_p + T_R)$	- 0.46	- 0.36	+ 0.10

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TABLE 2-64

DEVELOPMENT OF THE REACTIVITY FEEDBACK TO NOMINAL
RATED FULL POWER START OF LIFE FOR THE GRAPHITE R-1 REACTOR

<u>Operating Change</u>	<u>Magnitude</u>	<u>- Feedback -</u>	
		<u>As Calculated</u> <u>Per Section 2.5.2.1.2.1</u>	<u>Modified</u> <u>Per This Section</u>
Propellant Hydrogen	0.459 lb *	+ \$1.10	+ \$1.16
Support Stem Hydrogen	0.685 lb *	+ \$1.60	+ \$1.59
Core Neutronic Temperature	3150° R	- \$0.30	- \$0.80
Core Expansion	0.85%	- \$1.20	As Calculated
Seal Hydrogen	0.292 lb	+ \$0.16	
Cylinder Hydrogen	0.405 lb	+ \$0.15	
Reflector Hydrogen	1.208 lb	+ \$0.03	
Reflector Temperature	252° R	-0-	
Cylinder Contraction	-0.077 in. (Radial)	+ \$0.14	
		+ \$1.68	+ \$1.23

* These are reactivity weighted effective masses as indicated from TNT run TNTG381.

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TABLE 2-65

DEVELOPMENT OF THE CONTROL DRUM WORTH CHANGE TO NOMINAL
RATED FULL POWER START OF LIFE FOR THE GRAPHITE R-1 REACTOR

- Drum Worth Change -			
<u>Operating Change</u>	<u>Magnitude</u>	<u>As Calculated Per Section 2.5.2.1.2.1</u>	<u>Modified Per This Section</u>
Propellant Hydrogen	0.459 lb *	- 2.2 %	- 2.3 %
Support Stem Hydrogen	0.685 lb *	- 3.2 %	- 3.2 %
Core Neutronic Temperature	3150° R	- 3.0 %	- 8.0 %
Core Expansion	0.85%	+ 5.0 %	As Calculated
Seal Hydrogen	0.292 lb	0	
Cylinder Hydrogen	0.405 lb	- 0.8 %	
Reflector Hydrogen	1.208 lb	0	
Reflector Temperature	252° R	- 2.4 %	
Cylinder Contraction	-0.077 in. (Radial)	+ 1.5 %	
		- 5.2 %	
			-10.2 %

* Reactivity Weighted Effective Masses

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TABLE 2-66

NUCLEAR AND RADIATION ENVIRONMENTAL DRAWING
Nuclear Data - Graphite Core (Sheet 3)

I Major Nuclear Parameters

1. Control Drum Span (Ambient Conditions) = (\$4.50 Required, \$5.35 Calculated), 80% Probability, 95% Confidence
2. Reduction in Drum Span (Ambient-to-Operating Conditions) = 6%
3. Nominal Cold Critical Drum Position = 80°
4. Nominal Cold-to-Hot Reactivity Change = .75\$
5. Nominal Hot Critical Drum Position (Orifice Angle, SSCV Flow = 7.8 lb/sec) = 58°
6. Minimum Shutdown Reactivity from Cold Critical = \$ 1.70
7. Support Stem Reactivity Swing (Min SSCV Setting at Full Power to Max, SSCV Setting at Throttled) = \$2.23

II Critical Control Drum Angles

Power Test Identification	Full Power, Operating Time, sec	Cold Start of Life	Hot Start of Life (100% Power)	Hot Start of Life (60% Power, Throttled)	Hot End of Life (100% Power)	Hot End of Life (60% Power, Throttled)	Cold End of Life	Cold End of Life (11 Hrs after Shutdown)	Cooldown and Space Storage
Initial Power Run	3600	80°	58°	58°	58°	58°	82.5°	82.5°	55°
1st Restart									
2nd Restart									

III Power Distribution (Without Drum Movement)

Power Test Identification	Hot Start of Life (100% Power)				Hot End of Life (100% Power)			
	Max Power Factor in Core	Min Power Factor in Core	Max Peripheral Power Factor	Min Peripheral Power Factor	Max Power Factor in Core	Min Power Factor in Core	Max Peripheral Power Factor	Min Peripheral Power Factor
Initial Power Run	1.10	0.70	1.0	0.70	1.10	0.70	1.0	0.70
1st Restart								
2nd Restart								

Power Test Identification	Hot Start of Life (60% Power, Throttled)				Hot End of Life (60% Power, Throttled)			
	Max Power Factor in Core	Min Power Factor in Core	Max Peripheral Power Factor	Min Peripheral Power Factor	Max Power Factor in Core	Min Power Factor in Core	Max Peripheral Power Factor	Min Peripheral Power Factor
Initial Power Run	1.10	0.64	0.92	0.64	1.10	0.64	0.92	0.64
1st Restart								
2nd Restart								

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Figure 2-1. Flow Chart for Nuclear, Thermal, and Structural Analysis

Figure 2-2. Nuclear and Radiation Design NSS Analysis (Block 11)

Figure 2-3. Flow chart for the Calculation of Converged Spectra

Figure 2-4. Flow chart for the Calculation of Reactivity - Maximum Uranium Loading

Figure 2-5. Flowchart for Calculation of Power Map in Final Design at Test Conditions

Figure 2-6. Flow chart for the Calculation of Total Control Drum Span

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Figure 2-7. Flow Chart for NSS Nuclear Analytical Procedure Analysis

Figure 2-8, Outer Radial Region Geometry for R- θ EXTERMINATOR Calculations, Outer Radius, cm.

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Atomic Energy Act of 1954

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Atomic Energy Act of 1954

Figure 2-9. Fueled Carbon Weight versus Uranium Loading, Graphite Core Model

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Atomic Energy Act of 1954

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Figure 2-10. PAX-G1 Reflector Assembly

Figure 2-11. R-1 Reflector Mock-up PAX-G3 Configuration

Figure 2-12. PAX-G2A Central Channel Radial NC_i Variation

Figure 2-13. Comparison of Ambient and Full Power (EP-IV) Drum Worth Measurements

Figure 2-14. PHOEBUS-2A Correlation of Reactivity Discrepancy With Reflector Hydrogen Gas Temperature

Figure 2-15. Examination of NRX Manufacturing Variability Data for Normality

WITH THE SPECIFIED CONFIDENCE LEVEL AT LEAST P PERCENT OF
THE REACTIVITY DIFFERENCES WILL BE LESS THAN THAT SHOWN

Figure 2-16. Fuel Manufacturing Reactivity Differences from NRX Experience (R-1, Graphite Fueled Reactor)

Figure 2-17. Examination of the WISDM Absolute Reactivity Error Data for Normality

WITH THE SPECIFIED CONFIDENCE LEVEL, AT LEAST P PERCENT OF
THE REACTIVITY ERRORS WILL BE LESS THAN SHOWN

Figure 2-18. Absolute Reactivity Error from PAX-G/WISDM Experience (R-1, Graphite Fueled Reactor)

Figure 2-19. Examination of NRX Reactivity Error Data for Normality

WITH THE SPECIFIED CONFIDENCE LEVEL, AT LEAST P PERCENT OF
THE REACTIVITY ERRORS WILL BE LESS THAN THAT SHOWN

Figure 2-20. Absolute Reactivity Error from NRX Experience (Flight, Graphite Fueled Reactor)

Figure 2-21. Examination of the NRX Corrected Cold-To-Hot Reactivity Swing Data for Normality

WITH THE SPECIFIED CONFIDENCE LEVEL AT LEAST P PERCENT OF
THE VARIATIONS IN THE COLD-TO-HOT SWINGS WILL BE LESS THAN THAT SHOWN

Figure 2-22. Cold-To-Hot Reactivity Swing Variation from NRX Experience (Flight, Graphite Fueled Reactor)

Figure 2-23. Schematic of Engine Reactor Assembly Operation

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WITH THE SPECIFIED CONFIDENCE LEVEL, AT LEAST P PERCENT OF
THE REACTIVITY ERRORS WILL BE LESS THAN THAT SHOWN

Figure 2-24. R-1 Absolute Reactivity Error Envelopes

Figure 2-25. Radial Distribution of the Preliminary R-1 Fuel Loadings

RADIAL DISTANCE FROM CORE CENTERLINE - CM
Figure 2-26. R-1 Radial Fission Distribution WITH ORIFICE ANGLE OF 50°

Figure 2-27. PAX-G2A Central Channel Radial NC_f Distribution

Figure 2-28. R-1 Radial Fission Distribution Required by Exterminator Orifice Angle = 50°

Figure 2-30. R-1 Central Element Power Perturbation

Figure 2-31. Relative Radial Distribution of Photon and Neutron Kinetic Heating in the Core, Regions 1-15

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Figure 2-32. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation For One Dollar Outward Drum Rotation

Figure 2-33. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Seal Hydrogen Flow With No Drum Bank Representation

Figure 2-34. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Seal Hydrogen Flow With Drum Bank Representation Worth of \$2.77

Figure 2-35. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Cylinder Hydrogen Flow With No Drum Bank Representation

Figure 2-36. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Cylinder Hydrogen Flow With Drum Bank Representation Worth of \$2.77

Figure 2-37. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Reflector Hydrogen Flow With No Drum Bank Representation

Figure 2-38. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Reflector Hydrogen Flow With Drum Bank Representation Worth of \$2.77

Figure 2-39. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Peripheral Hydrogen Flow With No Drum Bank Representation

Figure 2-40. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Peripheral Hydrogen Flow With Drum Bank Representation Worth of \$2.77

Figure 2-41. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Reflector Temperature Change With No Drum Bank Representation

Figure 2-42. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Reflector Temperature Change With Drum Bank Representation Worth of \$2.77

Figure 2-43. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Peripheral Hydrogen Flow and Reflector Temperature Change With No Drum Bank Representation

Figure 2-44. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Peripheral Hydrogen Flow and Reflector Temperature Change With Drum Bank Representation Worth of \$2.77

Figure 2-45. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Core Temperature Change With No Drum Bank Representation

Figure 2-46. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Core Temperature Change With Drum Bank Representation Worth of \$2.77

Figure 2-47. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Core Expansion with No Drum Bank Representation

Figure 2-48. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Core Expansion with Drum Bank Representation Worth of \$2.77

Figure 2-49. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Cylinder Contraction With No Drum Bank Representation

Figure 2-50. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Cylinder Contraction With Drum Bank Representation Worth of \$2.77

Figure 2-51. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Core Hydrogen Flow With No Drum Bank Representation

Figure 2-52. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Core Hydrogen Flow With Drum Bank Representation Worth of \$2.77

Figure 2-53. R-1 Composite Core Reactor, Radial Fission Power Profile Change in Going From Ambient Conditions to Nominal Rated Full Power at Start of Life

Figure 2-54. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation For One Dollar Outward Drum Rotation

Figure 2-55. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of SSA Flow With No Drum Bank Representation

Figure 2-56. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of SSA Flow With Drum Bank Representation Worth \$2.77

Figure 2-57. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Corrosion With No Drum Bank Representation

Figure 2-58. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Corrosion With Drum Bank Representation Worth \$2.77

Figure 2-59. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Fuel Burnup With No Drum Bank Representation

Figure 2-60. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Fuel Burnup With Drum Bank Representation Worth \$2.77

Figure 2-61. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Fission Product Poisoning With No Drum Bank Representation

Figure 2-62. R-1 Composite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Fission Product Poisoning With Drum Bank Representation Worth \$2.77

Figure 2-63. R-1 Composite Core Reactor, Radial Fission Power Profile Change at the End of Ten Hour Reactor Operating Lifetime

Figure 2-64. Summary of NERVA Total Control Drum Span Utilization - Composite Fuel

Figure 2-65. Summary of NERVA Total Control Drum Span Utilization - Graphite Fuel

Figure 2-66. Reactivity Feedback Variation with Uniform Core Hydrogen Mass Content in the Composite R-1 Reactor

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Figure 2-67. Reactivity Feedback Variation with Average Core Temperature
(Excluding Core Expansion Effects) in the Composite R-1 Reactor

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Figure 2-68. Reactivity Feedback Variation with Combined Filler Strip and Seal Hydrogen Mass Content in the Composite R-1 Reactor

Figure 2-69. Reactivity Feedback Variation with Support Cylinder Hydrogen Mass Content in the Composite R-1 Reactor

Figure 2-70. Reactivity Feedback Variation With Reflector Hydrogen Mass Content in the Composite R1 Reactor

Figure 2-71. Reactivity Feedback Variation With Beryllium Temperature in the Composite R1 Reactor

Figure 2-72. Reactivity Feedback Variation with Aluminum Support Cylinder Inward Displacement in the Composite R-1 Reactor

Figure 2-73. Reactivity Feedback Variation with Percent Core Expansion in the Composite R-1 Reactor

Figure 2-74. Control Drum Worth Variation with Uniform Core Hydrogen Mass Content in the Composite R-1 Reactor

Figure 2-75. Control Drum Worth Variation With Average Core Temperature (Excluding Expansion Effects) in the Composite RI Reactor

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Figure 2-76. Control Drum Worth Variation with Support Cylinder Hydrogen Mass Content in the Composite R-1 Reactor

Figure 2-77. Control Drum Worth Variation with Beryllium Temperature in the Composite R-1 Reactor

Figure 2-78. Control Drum Worth Variation with Aluminum Support Cylinder Inward Displacement in the Composite R-1 Reactor

Figure 2-79., Control Drum Worth Variation with Percent Core Expansion in the Composite R-1 Reactor

Figure 2-80.. Peripheral Hydrogen Reactivity Feedback Dependence on Species and Gas Temperature in the Composite R-1 Reactor

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Figure 2-81. Dependence of Control Drum Worth on Peripheral Hydrogen Species and Gas Temperature in the Composite R-1 Reactor

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Figure 2-82. NRX/EST Measured Core Temperature Reactivity Feedback

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Figure 2-83. Operating Profiles of Power, Reactivity, Drum Position, and Support System Flow for Composite Core

Figure 2-84. Reactivity Maps ($T_c - P_c$) for R-1 Composite Core

Figure 2-86. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation For One Dollar Outward Drum Rotation

Figure 2-87. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Seal Hydrogen Flow With No Drum Bank Representation

Figure 2-88. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Seal Hydrogen Flow With Drum Bank Representation Worth of \$2.58

Figure 2-89. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Cylinder Hydrogen Flow With No Drum Bank Representation

Figure 2-90. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Cylinder Hydrogen Flow With Drum Bank Representation Worth of \$2.58

Figure 2-91. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Reflector Hydrogen Flow With No Drum Bank Representation

Figure 2-92. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Reflector Hydrogen Flow With Drum Bank Representation Worth of \$2.58

Figure 2-93. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Peripheral Hydrogen Flow With No Drum Bank Representation

Figure 2-94. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Peripheral Hydrogen Flow With Drum Bank Representation Worth of \$2.58

Figure 2-95: R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Reflector Temperature Change With No Drum Bank Representation

Figure 2-96. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Reflector Temperature Change With Drum Bank Representation Worth of \$2.58

Figure 2-97. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Peripheral Hydrogen Flow and Reflector Temperature Change With No Drum Bank Representation

Figure 2-98. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Peripheral Hydrogen Flow and Reflector Temperature Change With Drum Bank Representation Worth of \$2.58

Figure 2-99. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Core Temperature Change With No Drum Bank Representation

Figure 2-100. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Core Temperature Change With Drum Bank Representation Worth of \$2.58

Figure 2-101. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Core Expansion with No Drum Bank Representation

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Figure 2-102. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Core Expansion with Drum Bank Representation Worth of \$2.58

Figure 2-103. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Cylinder Contraction With No Drum Bank Representation

Figure 2-104. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Cylinder Contraction With Drum Bank Representation Worth of \$2.58

Figure 2-105. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Core Hydrogen Flow With No Drum Bank Representation

Figure 2-106. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Core Hydrogen Flow With Drum Bank Representation Worth of \$2.58

Figure 2-107. R-1 Graphite Core Reactor, Radial Fission Power Profile Change in Going From Ambient Conditions to Nominal Rated Full Power at Start of Life

Figure 2-108. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation For One Dollar Outward Drum Rotation

Figure 2-109. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of SSA Flow With No Drum Bank Representation

Figure 2-110. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of SSA Flow With Drum Bank Representation Worth \$2.58

Figure 2-111. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Corrosion With No Drum Bank Representation

Figure 2-112. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Corrosion With Drum Bank Representation Worth \$2.58

Figure 2-113. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Fuel Burnup With No Drum Bank Representation

Figure 2-114. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Fuel Burnup With Drum Bank Representation Worth \$2.58

Figure 2-115. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Fission Product Poisoning With No Drum Bank Representation

Figure 2-116. R-1 Graphite Core Reactor, Radial Fission Power Profile Perturbation Per Dollar of Fission Product Poisoning With Drum Bank Representation Worth \$2.58

Figure 2-117. R-1 Graphite Core Reactor, Radial Fission Power Profile Change at the End of Ten Hour Reactor Operating Lifetime

Figure 2-118. Reactivity Feedback Variation with Uniform Core Hydrogen Mass Content in the Graphite R-1 Reactor

Figure 2-119. Reactivity Feedback Variation with Average Core Temperature
(Excluding Core Expansion Effects) in the Graphite R-1 Reactor

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Figure 2-120. R-1 Reactivity Feedback Variation With Combined Filler Strip and Seal Hydrogen Mass Content in the Graphite R-1 Reactor

Figure 2-121. R-1 Reactivity Feedback Variation With Support Cylinder Hydrogen Mass Content in the Graphite R-1 Reactor

Figure 2-122. R-1 Reactivity Feedback Variation With Reflector Hydrogen Mass Content in the Graphite R-1 Reactor

Figure 2-123. R-1 Reactivity Feedback Variation With Beryllium Temperature in the Graphite R-1 Reactor

Figure 2-124. Reactivity Feedback Variation with Aluminum Support Cylinder Inward Displacement in the Graphite R-1 Reactor

Figure 2-125. Reactivity Feedback Variation with Percent Core Expansion in the Graphite R-1 Reactor

Figure 2-126. Control Drum Worth Variation with Uniform Core Hydrogen Mass Content in the Graphite R-1 Reactor

Figure 2-127. R-1 Control Drum Worth Variation With Average Core Temperature
(Excluding Expansion Effects) in the Graphite R-1 Reactor

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Figure 2-128. R-1 Control Drum Worth Variation With Support Cylinder Hydrogen Mass Content in the Graphite R-1 Reactor

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Figure 2-129. R-1 Control Drum Worth Variation With Beryllium Temperature in the Graphite R-1 Reactor

Figure 2-130. R-1 Control Drum Worth Variation With Aluminum Support Cylinder Inward Displacement in the Graphite R-1 Reactor

Figure 2-131. Control Drum Worth Variation with Percent Core Expansion in the Graphite R-1 Reactor

Figure 2-132. Operating Profiles of Power, Reactivity, Drum Position, and Support System Flow for Graphite Core

3.0 FUEL ELEMENTS

3.1 INTRODUCTION

The Fuel Element design has not been selected. Therefore, information will be presented in S-054 TS 772 - Fuel Element Trade Study, R-1 Selection.

4.0 CLUSTER HARDWARE

4.1 INTRODUCTION

The Cluster Hardware design has not been selected. Therefore, information will be presented in S-054 TS 774 - NERVA Cluster and Hot End Hardware, R-1 Selection.

5.0 CORE PERIPHERY

5.1 INTRODUCTION

The Core Periphery design has not been selected. Therefore, information will be presented in S-054, TS 778 - Core Periphery Trade Study for R-1 Selection.

6.0 SUPPORT PLATE AND PLENA

6.1 INTRODUCTION

6.1.1 Description of Support Plate and Plena

The SPP consists of a) a core support plate with three integral plena (an SSCA bypass flow plenum, a support stem inlet flow plenum and a plenum which combines the bypass flow with the support stem effluent, b) a piping arrangement which distributes the SSCA outlet flow to the support plate, c) flow separation baffles and associated hardware which separate the flow to the turbine from the turbine exit flow, d) a diffuser or baffle and associated hardware which distributes the turbine exit flow to the central shield.

The major components of the SPP from a nuclear consideration are 1) the aluminum 6061 structure, and 2) the void space that is filled with hydrogen coolant during operation. All of the remaining components were included in the analysis in a homogeneous form and were not specifically modeled.

6.1.2 Content of Chapter

This chapter summarizes the neutronic effects of the SUPPORT PLATE AND PLENA. The effect of the SPP on core reactivity and axial distribution is evaluated for both composite and graphite fuel in both the ambient and design operating conditions. The requirements on the SPP of a nuclear nature are reviewed and related analysis identified.

The method of analysis employed in performing this design evaluation is presented in detail including data on the analytical models developed and method flow charts. In addition, the status/maturity of the input data is discussed.

The validity of the method is determined by comparison with experimental results.

6.2 SUMMARY OF RESULTS AND CONCLUSIONS

This section presents a summary of the results and conclusions regarding the nuclear analysis of the SPP. The nuclear design requirements on the SPP are related to their reactivity contribution and their effect on the axial fission density distribution. The effects tabulated below in these two areas will be factored into the fuel loading and power distribution analysis. These results are similar to previously reported values and have no significant design implications.

6.2.1 Reactivity Effect **

For a composite fueled reactor the reactivity contribution of the SPP was found to be $+87.6 \text{ } \mu \pm 10 \text{ } \mu$ at ambient conditions. At rated conditions, the reactivity worth decreased to $+71.9 \text{ } \mu \pm 10 \text{ } \mu$.

For a graphite fueled reactor the reactivity contribution of the SPP was found to be $+75.2 \text{ } \mu \pm 10 \text{ } \mu$ at ambient conditions. At rated conditions, the reactivity worth decreased to $60.7 \text{ } \mu \pm 10 \text{ } \mu$.

6.2.2 Fission Density Effect

The presence of the SPP with a composite fueled reactor was found to increase the core inlet ambient fission density by 57% to a value of $0.29^* \pm 0.03$. At rated conditions, this increase was 70% to a value of $0.49^* \pm 0.05$.

The presence of the SPP with a graphite fueled reactor was found to increase the core inlet ambient fission density by 52% to a value of $\pm 0.26^* \pm 0.03$. At rated conditions, this increase was 63% to a value of $0.42^* \pm 0.04$.

*Relative to an axial average of unity.

**The uncertainty in these effects are current best estimates for R-1 and are expected to be refined in the future.

6.3 REQUIREMENTS

6.3.1 Specification Identification

The governing specification for this component is Specification Number EC-677561, "Part 1, Performance/Design and Qualification Requirements for Support Plate and Plena, Forming a part of CP-677555, NERVA NUCLEAR SUBSYSTEM. "

6.3.2 Applicable Requirements

The applicable requirements are quoted below.

3.2.1 Interface Requirements

Functional, dimensional, physical, and procedural interactions between the SPP and the other Engineering Critical Components which comprise the NSS, as defined in 3.2.2.2 (Engineering Critical Components List) of CP-677555, shall not degrade performance below that specified in 3.1 (Performance) of CP-677555 and its Engineering Critical Components. The following sections interpret this requirement for the SPP.

1) Absolute Reactivity

The reactivity contribution of the SPP to the NSS total reactivity must be such that CP-677555 requirement 3.3.1.3.3, "Nuclear Criticality," can be satisfied.

2) Core Axial Power Distribution

The contribution of the SPP to the NSS core axial power distribution must be such that the requirements of EC-677566 (Fuel Element Detail Specification), Tables 1 through 9 and Tables 20.1 through 20.9 can be satisfied.

The above interface requirements are met in that the axial reactivity and fission density results are included in the derivation of the absolute fuel loading distribution.

6.4 METHOD OF ANALYSIS - DESIGN PRACTICE

The method used in the Nuclear Design Analysis of the SPP is presented in this section. The discussion includes a description of 1) the analytical procedures on which the analysis has been based, 2) the models and computer programs used, and 3) the status/maturity of the basic data. In the concluding subsection, the validity of the analysis is assessed.

6.4.1 Analytical Procedures

The analytical procedure followed for the analysis reported here is illustrated by the flow chart of Figure 6-1.* Cross sections for each region in the "Nuclear Model" were developed by weighting the basic "Nuclear Properties" with the appropriate "Neutron Flux and Current Energy Spectrum." This process was carried out by standard WISDM procedures⁽⁷⁾ in the GAMBIT Code (Block 11-1).* The resultant 16-group regional macroscopic cross sections were then used by the transport theory ANISN computer code to solve the neutron transport problem and thus allow calculation of the effect of the various non-core regions on the core reactivity and axial power distribution. This process was followed for both the ambient reactor and the reactor at start-of-life design operation. The transverse neutron leakage effect (radial buckling) in all ANISN calculations was chosen to give an essentially zero reactivity for the axially-bare composite core. The reactivity effect and the effect of each non-core region on the core axial power distribution is taken to be the difference between the ANISN calculation with the particular region in place to that without the region. Thus, the computed effects of the Support Plate and the Support Plate Plenum are taken to be the difference between 1) a simulation which includes all regions in Table 6-2 through Core Plenum 3, and 2) a simulation identical to 1) but with the Support Plate and Support Plate Plenum added.

6.4.2 Nuclear Analytical Model

The analytical model for the study reported here was assembled from a detailed examination of the reference NSS design.⁽¹⁾ The source of the material masses and atom densities for the various regions in the model is summarized in Table 6-1. The model itself is illustrated schematically in Table 6-2. The sections that follow describe the modeling for the composite fueled core, the graphite fueled core and the non-core regions.

*Operation block called out from Figure 2-1. For discussion refer to Section 2.4.1.

6.4.2.1 Composite Core

(U) The core has been divided into four basic axial zones to permit a detailed simulation of the axial variation in hydrogen propellant density and material temperature in the design operating condition. These have been designated 1M, 2M, 3M, and 4M in Table 6-2 for a core composed of the reference design composite fuel. For simplicity, the same geometric model was used for all ANISN calculation planned in this study. As can be seen in Table 6-2, this resulted in a number of duplicate adjacent regions. This has no effect on the results and is done merely for convenience in geometric modeling.

(U) Atom densities for the constituents of each of the four composite fuel regions defined in Table 6-2 are tabulated in Tables 6-6 for both the ambient condition and full power design operation. Atom densities appearing in Table 6-6 have been taken from core radial Region 8 in Reference 3 for the composite core. This radial region was chosen for the axial calculations because its uranium density was characteristic of the core average.

(U) Hydrogen propellant conditions were taken from TNT Code computer calculations.⁽⁵⁾ Also taken from this computer run were the region average material temperatures for design operation. These are summarized in Table 6-7. The material temperatures were taken directly from the TNT results with the exception of those shown for the core where radial averaging and axial integration over appropriate intervals (See Table 6-2.) was employed to identify typical regional averages for use in neutronic calculations.

6.4.2.2 Graphite Core

(CRD) The core with graphite fuel was also separated into four basic axial fuel zones. However, for the graphite alternate fuel design, the core has been further divided into two additional axial zones located at the hot end of the reactor. The first, Region 1G, simulates the NbC/C unfueled tip on the graphite fuel and unfueled central elements. The second additional zone, Region 2G, is included to define the remainder of the unfueled element NbC/C tip which extends for 0.25 inches into the fuel material envelope. The composition of this region differs significantly only in its Nb content from Region 3G. Region 3G, analogous to Composite Fuel Region 1M, is the longest axial zone (31.38 inches) and is located toward the hot end of the core. The remaining Regions 4G, 5G, and 6G are comparable to

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regions 2M, 3M, and 4M in the composite fuel model. For simplicity, the same geometric model was used for all ANISN calculations planned in this study. As can be seen in Table 6-2, this resulted in a number of duplicate adjacent regions. This has no effect on the result and is merely done for convenience in geometric modeling.

(U) Atom densities for the constituents of each of the four basic graphite axial zones are tabulated in Table 6-6 for both the ambient condition and full power design operation. As in the case of composite fuel, atom densities appearing in Table 6-6 have been taken from core radial region 8 (Reference 4). This radial region was chosen for the calculations because its uranium density was characteristic of the core average. Table 6-8 defines the composition of the two thin regions 1G and 2G at the aft end of the graphite fueled core. The hydrogen propellant conditions were taken from TNT Code computer calculations for composite and graphite fuel (Reference 6).

(U) Also taken from these computer runs were the region average material temperatures for design operation. These are summarized in Table 6-7. The material temperatures were taken directly from the TNT results with the exception of those shown for the core where radial averaging and axial integration over appropriate intervals (See Table 6-2.) was employed to identify typical regional averages for use in neutronic calculations.

6.4.2.3 Non-Fuel Regions

(U) All non-fuel atom densities are summarized in Table 6-9. As noted in earlier sections, the aft end of the core was subdivided into duplicate adjacent regions for convenience in modeling. This affects only the model of the Cluster Hot End Support Hardware (CHESH).

(CRD) In the regions forward of the core, the components and void between the support plate and the forward edge of the fuel were separated into 5 zones. Three of these, Core Plenum 1, 2, and 3, have been assigned the same homogenized material densities. The other two simulate the thin, 60-mil Copper-Natural Boron Cluster plate and disc. The remaining regions simulate the support plate, its plenum, and the central internal shield. In the analytical work reported here, the effects of appropriate groups of these regions are investigated

(U) Tables 6-6, 6-8, and 6-9 provide a composition to simulate design operation by the addition of an appropriate hydrogen atom density to the ambient regional atom densities.

Note that in the case of the BATH Central Shield there is hydrogen present in the ambient condition as well.

6.4.2.4 Problem and Tape Log

A list of the various program tapes, the problem log, and corresponding output tapes pertinent to this analysis can be found in DRM-53365⁽¹³⁾. Source references to these problems are found in the lower right corner of all tables and figures where applicable.

6.4.3 Status/Maturity of Input Data

6.4.3.1 Composite Core

6.4.3.1.1 Material and Coolant Design Power Conditions

Reference 5 contains the material temperatures and design power coolant conditions.

6.4.3.1.2 Nuclear Cross Section Data

Table 6-3 lists the basic neutron cross section data tapes employed in the SPP analysis.

6.4.3.2 Graphite Core

6.4.3.2.1 Material and Coolant Design Power Conditions

Reference 6 contains the material temperatures and design power coolant conditions.

6.4.3.2.2 Nuclear Cross Section Data

Table 6-3 lists the basic neutron cross section data tapes employed in the SPP analysis.

6.4.4 Validity of Analysis

The validity of the analytical modeling and procedures discussed in Sections 6.4.1 and 6.4.2 is presented here. Previously published analyses^(9, 10) assessed the accuracy of a similar technique. The results obtained have been included here for convenience and to illustrate by example the accuracy of analytical procedures typical of those shown in Figure 6-1. These results are based on a graphite core. It is expected that this validity is equally applicable to a composite fueled reactor.

6.4.4.1 Reactivity Effects

A series of calculations was made to duplicate experiments performed on PAX and KIWI-B4. These experiments consisted of measuring the reactivity worths of various materials on the top of the PAX and KIWI-B4 reactors. A comparison of the predictions and measurements is shown in Tables 6-4 and 6-5. As can be seen, agreement is within 6¢ in the case of PAX data and within 11¢ for the much larger KIWI-B4 effects. Based on these results, it is estimated that the uncertainty in these calculations is $\pm 10\%$.

6.4.4.2 Fission Density

A comparison is presented of the calculated and measured axial fission distribution in the PAX-GOC reactor. This has been reproduced in Figure 6-2. Although the calculation was performed with the DOT two-dimensional (r,z) code instead of the ANISN one dimensional (z) code, it is appropriate to quote these results for two reasons. First, the solution in both codes is based on neutron transport theory, and the codes have, in fact, given identical results for an analytical model which could be simulated by both.⁽¹¹⁾ Second, the axial traverse shown in Figure 6-2 is on the reactor centerline where axial effects are dominant. As is evident from Figure 6-2, the calculated fission density profile agrees very well with the measurement.

This agreement at the mocked-up inlet of the reactor is very good. It is estimated that the uncertainty in the calculated fission distribution at the inlet end is $\pm 5\%$ to $\pm 10\%$.

6.5 RESULTS, ANALYSIS AND INTERPRETATION

The discussion in this section has been divided into composite core and graphite core effects. For each of these fuel types, the effect of the SPP on the ambient reactor and the reactor at design operation is presented.

6.5.1 Composite Core

6.5.1.1 Steady State

Two steady state operational modes are examined in this section. The first relates to the ambient reactor. Full power design operation is the second mode.

6.5.1.1.1 Ambient Conditions

The ANISN calculations for the NSS at ambient conditions have been based on the fuel atom density model appearing in Table 6-6 and the non-fuel atom density model in Table 6-9. As was noted above, the computed effect of the SPP and, any non-fuel region, is obtained in a cumulative manner. Thus, the computed effect of the Support Plate and the Support Plate Plenum are taken to be the difference between 1) a simulation which includes all regions in Table 6-2 through Core Plenum 3, and 2) a simulation identical to 1) but with the Support Plate and Support Plate Plenum added. Reactivity effects for all non-fuel regions are summarized in Table 6-10. Effects have been computed on both a cumulative and on a differential basis. The combined reactivity worth of the support plate and support plate plenum is seen to be $+87.6\% \pm 10\%$. The effect of the addition of the support plate and support plate plenum on the core axial fission density is found by comparison of the fission distributions from ANISN Cases D and E noted on Table 6-10. The two fission density profiles are shown in Figure 6-3. The computed inlet value is seen to be 0.29 ± 0.03 , an increase of 57% over the inlet value without the SPP. This value of 0.286 is relative to an axial average of unity.

6.5.1.1.2 Design Operation

In a manner analogous to that described in Section 6.5.1.1.1, the reactivity worth and the effect of the SPP on the reactor's fission density distribution at

design operation has been calculated. Reactivity effects for all non-fuel regions are summarized in Table 6-11. Effects have been computed on both a cumulative and on a differential basis. The combined reactivity worth of the support plate and support plate plenum is seen to be $+71.9\% \pm 10\%$. The effect of the addition of the support plate and support plate plenum on the core axial fission density is found by comparison of the fission distributions from ANISN Cases I and J noted on Table 6-11. The two fission density profiles are shown in Figure 6-4. The inlet value is seen to be 0.49 ± 0.05 , an increase of 70% over the inlet value without the SPP. Again the value of 0.49 for the inlet power is relative to an axial average of unity.

6.5.2 Graphite Core

6.5.2.1 Steady State

Two steady state operational modes are examined in this section. The first relates to the ambient reactor. Full power design operation is the second mode. The treatment is analogous to that appearing in Section 6.5.1.

6.5.2.1.1 Ambient Conditions

The ANISN calculations for the NSS at ambient conditions have been based on the fuel atom density model appearing in Table 6-6 and the non-fuel atom density model in Table 6-9. As was noted above, the computed effect of the SPP and, in general, any non-fuel region, is obtained in a cumulative manner. Thus, the computed effects of the Support Plate and the Support Plate Plenum are taken to be the difference between 1) a simulation which includes all regions in Table 6-2 through Core Plenum 3, and 2) a simulation identical to 1) but with the Support Plate and Support Plate Plenum added. Reactivity effects for all non-fuel regions are summarized in Table 6-12. Effects have been computed on both a cumulative and on a differential basis. The combined reactivity worth of the Support Plate and Support Plate Plenum is seen to be $+75.2\% \pm 10\%$. The effect of the addition of the Support Plate and Support Plate Plenum on the core axial fission density is found by comparison of the fission distributions from ANISN Cases DD and EE noted on Table 6-12. The two fission density profiles are shown in Figure 6-5. The inlet value is seen to be 0.26 ± 0.03 (normalized to an axial average of unity), an increase of 52% over the inlet value without the SPP.

6.5.2.1.2 Design Operation

In a manner analogous to that described in Section 6.5.2.1.1 the reactivity worth and the effect of the SPP on the reactor's fission density distribution at design operation has been calculated. Reactivity effects for all non-fuel regions are summarized in Table 6-13. Effects have been computed on both a cumulative and on a differential basis. The combined reactivity worth of the support plate and support plate plenum is seen to be $+60.7\% \pm 10\%$.

The effect of the addition of the support plate and support plate plenum on the core axial fission density is found by comparison of the fission distributions from ANISN Cases II and JJ noted on Table 6-13. The two fission density profiles are shown in Figure 6-6. The inlet value is seen to be 0.42 ± 0.04 (normalized to an axial average of unity).an increase of 63% over the inlet value without the SPP.

6.6 REFERENCES

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TABLE 6-1
DATA SOURCES FOR THE ANALYTICAL
MODEL USED IN AXIAL ANISN CALCULATIONS

Region	Source
CHESH	DRM-52842 (Ref. 2)
Composite Fuel Core	
Axial Regions 1-4	DRM-52772 (Ref. 3) ^{a,c}
Graphite Fuel Core	
Unfueled Tip (Region 1)	DWG 711J781 and DRM-52842 (Nb C/C Properties) ^d
Axial Region 2	DWG 711J781 and DRM-52774 (Ref. 4) ^{a, b, d}
Axial Regions 3-6	DRM-52774 ^{a, d}
Core Plenum	DRM-52842 ^b
Poison Disc	DRM-52842 ^b
Poison Plate	DRM-52842 ^b
Support Plate	DRM-52842
Support Plate Plenum	DRM-52842
Aft Central Shield Plate	DRM-52842
Bath Central Shield	DRM-52842 ^b
Lead Central Shield	DRM-52842

a. Radial Region 8 was used from these DRMs.

b. Revised to give more detailed simulation

c. Hydrogen conditions and Material Temperatures from TNTM158 (Ref. 5)

d. Hydrogen conditions and Material Temperatures from TNTG381 (Ref. 6)

TABLE 6-2
GEOMETRY MODEL FOR AXIAL ANISN CALCULATIONS

Station, cm	Region	
0.0	Composite Fuel*	Graphite Fuel
5.413	CHESH	CHESH
6.172	Core 1M	Core 1G
6.353	Core 1M	Core 1G
6.988	Core 1M	Core 2G
7.112	Core 1M	Core 3G
7.747	Core 1M	Core 3G
86.693	Core 1M	Core 3G
124.793	Core 2M	Core 4G
134.953	Core 3M	Core 5G
137.493	Core 4M	Core 6G
138.598	Core Plenum 1	
138.750	Poison Disc	
139.144	Core Plenum 2	
139.296	Poison Plate	
146.383	Core Plenum 3	
161.623	Support Plate	
165.001	Support Plate Plenum	
167.541	Aft Central Shield Plate	
193.195	BATH Central Shield	
196.497	Lead Central Shield	

* Designates the reference design.

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TABLE 6-3
NEUTRON CROSS SECTION DATA TAPES
EMPLOYED IN THE SPP ANALYSIS

<u>Cross Section Tape Identification</u>	<u>Description</u>
FSGAM28, RD1, RD2.	GAM LIBRARY TAPE
CGBIT15.	BIT LIBRARY TAPE
GWQRY1, RD1, RD2.	QUERY LIBRARY TAPE
CGGAS15.	GAS LIBRARY TAPE

TABLE 6-4
WORTH OF MATERIAL ON TOP OF PAX*

<u>Material</u>	<u>Thickness (cm)</u>	<u>Measured Worth (¢) Cumulative</u>	<u>Calculated Worth (¢)</u>
Graphite	5.715		
Spacer	2.286		
Stainless Steel	0.198		
Polyethylene	0.005	6.40	6.90
Gadolinium and Stainless Steel	0.318		
Polyethylene	0.005	8.84	14.91
Polyethylene	16.000		
Aluminum	0.635	27.51	36.01
Aluminum	0.635	28.86	37.31
Aluminum	6.350	38.57	46.38

*Source: Reference 9, Table 12-1

TABLE 6-5

CUMULATIVE WORTH OF MATERIAL ON TOP OF KIWI-B4 CORE*

<u>Material</u>	<u>Thickness (cm)</u>	<u>Measured Worth (£) Cumulative</u>	<u>Calculated Worth (£)</u>
Niobium	0.0185		
Spacer	0.6350		
Graphite	5.0800	111.1	108.3
Niobium	0.0721		
Spacer	2.6988		
Graphite	4.2863	177.3	166.5
Niobium	0.1854		
Inconel	0.3175	185.0	174.6
Iron	0.3175	188.4	179.0
Spacer	0.6350		
Graphite	1.2700	198.6	187.9

*Source: Reference 9, Table 12-2

TABLE 6-6
 AMBIENT AND DESIGN OPERATION
 ATOM DENSITIES FOR CORE REGIONS
 - COMPOSITE & GRAPHITE FUEL

Chemical Element	Region 1M, 2M, 3M, 4M	Region 3G, 4G, 5G, 6G
U-235	0.8703×10^{-3}	0.7274×10^{-3}
U-238	0.6320×10^{-4}	0.5280×10^{-4}
C	4.6486×10^{-2}	6.3978×10^{-2}
Zr 90	4.3790×10^{-3}	5.6300×10^{-4}
Zr 91	0.9520×10^{-3}	1.2200×10^{-4}
Zr 92	1.4540×10^{-3}	1.8700×10^{-4}
Zr 94	1.4800×10^{-3}	1.9000×10^{-4}
Zr 96	0.2380×10^{-3}	0.3100×10^{-4}
Nb		5.9250×10^{-4}
Mn	0.1100×10^{-4}	0.1100×10^{-4}
Cr	0.1120×10^{-3}	0.1120×10^{-3}
Ni	0.1570×10^{-3}	0.1570×10^{-3}
Mo 95	0.1000×10^{-5}	0.1000×10^{-5}
Mo 96	0.1000×10^{-5}	0.1000×10^{-5}
Mo 97	0.1000×10^{-5}	0.1000×10^{-5}
Mo 98	0.3000×10^{-5}	0.4000×10^{-5}
Mo 100	0.1000×10^{-5}	
Ti	0.1600×10^{-4}	0.1600×10^{-4}
Fe	0.3780×10^{-3}	0.3780×10^{-3}
Ta	0.1000×10^{-4}	0.1000×10^{-4}
H	0.2650×10^{-3} (1M)	0.2650×10^{-3} (3G)
	0.5000×10^{-3} (2M)	0.5000×10^{-3} (4G)
	0.9840×10^{-3} (3M)	0.9840×10^{-3} (5G)
	1.2400×10^{-3} (4M)	1.2400×10^{-3} (6G)
	0 (1M-4M Ambient)	0 (3G-6G Ambient)

TABLE 6-7

REGION AVERAGE TEMPERATURES FOR
DESIGN OPERATION CALCULATIONS

<u>Region</u>	SOL Material Temperature, °R	BIT Kernel ⁽⁸⁾	
		<u>Hydrogen</u>	<u>Carbon</u>
CHESH*	550	54	5
Core 1M	3935	56	29
Core 2M	2250	55	3
Core 3M	1025	50	33
Core 4M	550	51	2
Core 1G	3935	56	29
Core 2G	3935	56	29
Core 3G	3935	56	29
Core 4G	2250	55	3
Core 5G	1025	50	33
Core 6G	550	51	2
Core Plenum (1, 2, 3)	540	-	-
Core Plenum Hydrogen	285	54	-
Poison Disc	1200	-	-
Poison Plate	1200	-	-
Support Plate	360	52	2
Support Plate Plenum	280	54	-
Aft Central Shield Plate	280	54	-
BATH Central Shield	280	54	2
Lead Central Shield	280	54	-

* Due to the function of the CHESH, some of the constituent materials perform at a significantly different temperature than is listed in the table. They are H at 270° R, C at 2600° R, and W at 4200° R.

TABLE 6-8

AMBIENT AND DESIGN OPERATION
ATOM DENSITIES FOR CORE REGIONS 1G AND 2G

Chemical Element	Atom Density, Atoms per barn centimeter	
	Region 1 G	Region 2G
U-235		0.7274×10^{-3}
U-238		0.5280×10^{-4}
C	5.5476×10^{-2}	6.3465×10^{-2}
Zr-90		5.6300×10^{-4}
Zr-91		1.2200×10^{-4}
Zr-92		1.8700×10^{-4}
Zr-94		1.9000×10^{-4}
Zr-96		0.3100×10^{-4}
Nb	0.9727×10^{-2}	1.5179×10^{-3}
Mn		0.1100×10^{-4}
Cr		0.1120×10^{-3}
Ni		0.1570×10^{-3}
Mo-95		0.1000×10^{-5}
Mo-96		0.1000×10^{-5}
Mo-97		0.1000×10^{-5}
Mo-98		0.4000×10^{-5}
Ti		0.1600×10^{-4}
Fe		0.3780×10^{-3}
Ta		0.1000×10^{-4}
H	0.2650×10^{-3}	0.2650×10^{-3}

TABLE 6-9
 AMBIENT AND DESIGN OPERATION
 ATOM DENSITIES* FOR NON-FUEL REGIONS

Chemical Element	REGION									
	CHESH 433-G02 738-G02	CHESH 433-G01 738-G01	Core Plenum 1, 2, & 3	Poison Disc	Poison Plate	Support Plate	Support Plate Plenum	Aft Central Shield Plate	BATH Central Shield	Lead Central Shield
H ambient	-	-	-	-	-	-	-	-	1.7840-2	-
operation	9.5262-4	9.6755-4	3.4609-3	-	-	6.2524-3	4.9994-3	6.4823-4	1.8206-2	6.5319-4
B nat'l	-	-	-	6.0750-3	8.6590-3	-	-	-	5.6696-3	-
C	1.0424-2	6.4323-3	-	-	-	3.1099-4	-	-	1.4190-3	-
Th-232	5.4759-5	1.0502-4	-	-	-	-	-	-	-	-
O-16	1.0833-4	2.0775-4	-	-	-	-	-	-	-	-
Mg	-	-	-	-	-	3.0982-4	-	5.6322-4	1.7374-6	1.3611-6
Al	-	-	-	-	-	2.7638-2	-	5.0242-2	3.6926-2	1.2142-4
Cr	8.6086-4	8.6086-4	2.7050-3	-	-	1.6255-3	2.0795-3	-	-	-
Mn	8.3010-5	8.3010-5	2.7119-4	-	-	1.6169-4	2.0712-4	-	-	-
Fe	2.9815-3	2.9815-3	9.3809-3	-	-	5.5425-3	7.0830-3	-	-	-
Ni	1.2774-3	1.2774-3	1.6209-3	-	-	8.1297-4	9.3816-4	-	-	-
Cu	-	-	-	1.6400-2	2.3375-2	-	-	-	-	-
Mo	3.7977-5	3.7977-5	2.7037-3	-	-	3.9312-6	-	-	-	-
Pb	-	-	-	-	-	-	-	-	-	2.7516-2
Nb	9.8439-4	2.9054-4	-	-	-	-	-	-	-	-
Ti	1.2721-4	1.2721-4	6.0081-5	-	-	1.3228-5	2.4590-6	-	9.9204-3	-
W	3.8482-3	7.3802-3	-	-	-	-	-	-	-	-

* Atoms per barn centimeter in exponential notation, i.e., 1.9×10^{-3} is entered in table as 1.9-3.

TABLE 6-10

AMBIENT REACTIVITY WORTH OF AFT AND
FORWARD NON-FUEL COMPONENTS
(COMPOSITE FUEL)

<u>Component/Region</u>	<u>Case Number</u>	<u>Eigenvalue</u>	<u>Reactivity Change*</u> <u>Relative to Case B, %</u>	
			<u>Cumulative</u>	<u>Differential</u>
Bare Core	B	0.999806	0	-
CHESH	C1	1.003678	55.3	55.3
Core Plenum 1	D	1.008397	122.7	67.4
Poison Disc				
Core Plenum 2				
Poison Plate				
Core Plenum 3				
Support Plate	E	1.014526	210.3	87.6
Support Plate Plenum				
Aft Central Shield Plate	F	1.015903	230.0	19.7
BATH Central Shield				
Lead Central Shield				

697 11/3/70

*Based on an effective delayed neutron fraction of 0.007 (Reference 12)

TABLE 6-11

DESIGN OPERATION REACTIVITY WORTH OF AFT AND
FORWARD NON-FUEL COMPONENTS
(COMPOSITE FUEL)

<u>Component/Region</u>	<u>Case Number</u>	<u>Eigenvalue</u>	<u>Reactivity Change*</u> <u>Relative to Case G, %</u>	
			<u>Cumulative</u>	<u>Differential</u>
Bare Core	G	1.019945	0	--
CHESH	H1	1.023533	51.3	51.3
Core Plenum 1	I	1.029034	129.8	78.5
Poison Disc				
Core Plenum 2				
Poison Plate				
Core Plenum 3	J	1.034063	201.7	71.9
Support Plate				
Support Plate Plenum	K	1.034468	207.5	5.8
Aft Central Shield Plate				
BATH Central Shield				
Lead Central Shield				

110 11/3/70

*Based on an effective delayed neutron fraction of 0.007 (Reference 12)

TABLE 6-12
AMBIENT REACTIVITY WORTH OF AFT AND
FORWARD NON-FUEL COMPONENTS
(GRAPHITE FUEL)

<u>Component/Region</u>	<u>Case Number</u>	<u>Eigenvalue</u>	<u>Reactivity Change* Relative to Case BB, %</u>	
			<u>Cumulative</u>	<u>Differential</u>
Fuel without Unfueled Tip	BB1	1.024055	- 22.4	22.4
Fuel with Unfueled Tip	BB	1.025620	0	-
CHESH	CC1	1.029094	49.6	49.6
Core Plenum 1	DD	1.033399	111.1	61.5
Poison Disc				
Core Plenum 2				
Poison Plate				
Core Plenum 3	EE	1.038662	186.3	75.2
Support Plate				
Support Plate Plenum	FF	1.039781	202.3	16.0
Aft Central Shield Plate				
BATH Central Shield				
Lead Central Shield				

697 11/3/70

*Based on an effective delayed neutron fraction of 0.007 (Reference 12)

TABLE 6-13
DESIGN OPERATION REACTIVITY WORTH OF AFT AND
FORWARD NON-FUEL COMPONENTS
(GRAPHITE FUEL)

<u>Component/Region</u>	<u>Case Number</u>	<u>Eigenvalue</u>	<u>Reactivity Change* Relative to Case GG, %</u>	
			<u>Cumulative</u>	<u>Differential</u>
Fuel without Unfueled Tip	GG1	1.041932	- 20.6	20.6
Fuel with Unfueled Tip	GG	1.043372	0	-
CHESH	HH1	1.046649	46.8	46.8
Core Plenum 1	II	1.051621	117.8	71.0
Poison Disc				
Core Plenum 2				
Poison Plate				
Core Plenum 3	JJ	1.055866	178.5	60.7
Support Plate				
Support Plate Plenum	KK	1.056201	183.3	4.8
Aft Central Shield Plate				
BATH Central Shield				
Lead Central Shield				

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698 11/3/70

*Based on an effective delayed neutron fraction of 0.007 (Reference 12).

Figure 6-1 Flow chart for Axial Calculations

Figure 6-2. Comparison of Calculated and Measured Axial Fission Distribution with BATH Shield Mockups in Place*

*Source: Reference 10, Figure 5-11.

Figure 6-3. Effect of SPP on Axial Fission Density Distribution (Ambient, Composite Core)

Figure 6-4. Effect of SPP on Axial Fission Density Distribution (Hot, Composite Core)

Figure 6-5. Effect of SPP on Axial Fission Density Distribution (Ambient, Graphite Core)

Figure 6-6. Effect of SPP on Axial Fission Density Distribution (Hot, Graphite Core)

7.0 INTERNAL SHIELD

7.1 INTRODUCTION

7.1.1 Description of Internal Shield

The Internal Shield consists of a Lead and Boron Carbide-Aluminum-Titanium Hydride (BATH) central shield located forward of the core extending in radius to the outer radius of the core, and peripheral shield located above the reflector extending from the inner radius of the reflector to the inner radius of the pressure vessel.

The major components of the Internal Shield (IS) from a nuclear design consideration are 1) the aluminum aft central shield plate, 2) the BATH central shield and 3) the lead central shield. With the exception of the peripheral shield, all of the remaining components were included in the analysis in a homogeneous form and were not specifically modeled. Since the analysis here is one dimensional, the peripheral shield has not been simulated in the model.

7.1.2 Content of Chapter

This chapter summarizes the neutronic effects of the INTERNAL SHIELD. The effect of the IS on core reactivity and axial power distribution is evaluated for both composite and graphite fuel in both the ambient and design operating conditions. The requirements on the SPP of a nuclear nature are reviewed and related analysis identified.

The method of analysis employed in performing this design evaluation is presented in detail including data on the analytical models developed and method flow charts. In addition, the status/maturity of the input data is discussed.

The validity of the method is determined by comparison with experimental results.

7.2 SUMMARY OF RESULTS AND CONCLUSIONS

This section presents a summary of the results and conclusions regarding the nuclear analysis of the IS. The nuclear design requirements on the IS are related to their reactivity contribution and their effect on the axial fission density distribution. The effects tabulated below in these two areas will be factored into the fuel loading and power distribution analysis. The results are similar to previously reported values and have no significant design implications.

7.2.1 Reactivity Effect **

For a composite fueled reactor the reactivity contribution of the IS was found to be $+19.7 \text{ } \mu\text{K} \pm 10 \text{ } \mu\text{K}$ at ambient conditions. At rated conditions, the reactivity worth decreased to $+5.8 \text{ } \mu\text{K} \pm 10 \text{ } \mu\text{K}$.

For a graphite fueled reactor the reactivity contribution of the IS was found to be $+16.0 \text{ } \mu\text{K} \pm 10 \text{ } \mu\text{K}$ at ambient conditions. At rated conditions, the reactivity worth decreased to $4.8 \text{ } \mu\text{K} \pm 10 \text{ } \mu\text{K}$.

7.2.2 Fission Density Effect

The presence of the IS with a composite fueled reactor was found to increase the core inlet ambient fission density to $0.32^* \pm 0.03$. At rated conditions, this increases to a value of $0.59^* \pm 0.06$. This latter value also includes the effect of neutrons streaming through the inlet poison plates.

The presence of the IS with a graphite fueled reactor was found to increase the core inlet ambient fission density to $0.29^* \pm 0.03$. At rated conditions, this increases to a value of $0.51^* \pm 0.05$. This latter value also includes the effect of neutrons streaming through the inlet poison plates.

*Relative to an axial average of unity.

**The uncertainty in these effects are current best estimates for R-1 and are expected to be refined in the future.

7.3 REQUIREMENTS

7.3.1 Specification Identification

The governing specification for this component is Specification Number EC-677562, "Part I, Performance/Design and Qualification Requirements for Internal Shield, Forming a Part of CP-677555, NERVA NUCLEAR SUBSYSTEM. "

7.3.2 Applicable Requirements

The applicable requirements are quoted below.

3.2.1 Interface Requirements

Functional, dimensional, physical, and procedural interactions between the Internal Shield and the other Engineering Critical Components which comprise the NSS, as defined in 3.2.2.2 (Engineering Critical Components List) of CP-677555, shall not degrade performance below that specified in 3.1 (Performance) of CP-677555 and its Engineering Critical Components. The following sections interpret this requirement for the Internal Shield.

1) Absolute Reactivity

The reactivity contribution of the IS to the NSS total reactivity must be such that CP-677555 requirement 3.3.1.3.3, "Nuclear Criticality," can be satisfied.

2) Core Axial Power Distribution

The contribution of the IS to the NSS core axial power distribution must be such that the requirements of EC-677566 (Fuel Element Detail Specification), Tables 1 through 9 and Tables 20.1 through 20.9 can be satisfied.

The above interface requirements are met in that the axial reactivity and fission density results are included in the derivation of the absolute fuel loading distribution.

7.4 METHOD OF ANALYSIS – DESIGN PRACTICE

The method used in the Nuclear Design Analysis of the IS is presented in this section. The discussion includes a description of 1) the analytical procedures on which the analysis has been based, 2) the models and computer programs used, and 3) the status/maturity of the basic data. In the concluding subsection, the validity of the analysis is assessed.

7.4.1 Analytical Procedures

The analytical procedures used in the Analysis of the IS are in conformance with the standard WISDM ⁽¹⁾ procedures and are the same as those used for the SPP. Refer to Section 6.4.1 for a description of these procedures.

7.4.2 Nuclear Analytical Model

The analytical model for the study reported here was assembled from a detailed examination of the reference NSS design.⁽²⁾ The source of the material masses and atom densities for the IS and for the various other regions in the NSS model is summarized in Table 6-1. The NSS model itself, identical to that for the SPP analysis, is illustrated in Table 6-2. The sections that follow describe the modeling for the composite fueled core, the graphite fueled core and the non-core regions.

7.4.2.1 Composite Core

The analytical model for the composite core is the same as that described in Section 6.4.2.1.

7.4.2.2 Graphite Core

The analytical model for the graphite core is the same as that described in Section 6.4.2.2.

7.4.2.3 Non-Fueled Regions

The analytical model for the IS and other non-fueled regions in the NSS is the same as that described in Section 6.4.2.3.

7.4.2.4 Problem and Tape Log

A list of the various program tapes, the problem log, and corresponding output tapes pertinent to this analysis can be found in DRM-53365⁽⁶⁾. Source reference to these problems are found in the lower right corner of all tables and figures where applicable.

7.4.3 Status/Maturity of Input Data

7.4.3.1 Composite Core

7.4.3.1.1 Material and Coolant Design Power Conditions

Reference 3 contains the material temperatures and design power coolant conditions.

7.4.3.1.2 Nuclear Cross Section Data

Table 6-3 lists the basic neutron cross section data tapes employed in the IS analysis.

7.4.3.2 Graphite Core

7.4.3.2.1 Material and Coolant Design Power Conditions

Reference 4 contains the material temperatures and design power coolant conditions.

7.4.3.2.2 Nuclear Cross Section Data

Table 6-3 lists the basic neutron cross section data tapes employed in the IS analysis.

7.4.4 Validity of Analysis

The validity of the analytical procedures and models used here to assess the reactivity worth and fission density effect of the IS and, in general, any non-fuel component is discussed in Section 6.4.4.

7.5 RESULTS, ANALYSIS AND INTERPRETATION

The discussion on this section has been divided into composite core and graphite core effects. For each of these fuel types, the effect of the IS on the ambient reactor and the reactor at design operation is presented.

7.5.1 Composite Core

7.5.1.1 Steady State

Two steady state operational modes are examined in this section. The first relates to the ambient reactor. Full power design operation is the second mode.

7.5.1.1.1 Ambient Conditions

The ANISN calculations for the NSS at ambient conditions are listed in Table 6-2. These have been based on the fuel atom density model appearing in Table 6-6 and the non-fuel atom density model in Table 6-9. As was noted in Chapter 6, the computed effect of the IS and, in general, any non-fuel region, is obtained in a cumulative manner. Thus, the computed effects of the Internal Shield is taken to be the difference between 1) a simulation which includes all regions in Table 6-2 through the Support Plate Plenum and 2) a simulation identical to 1) but with the Internal Shield added.

Reactivity effects for all non-fuel regions are summarized in Table 6-10. Effects have been computed on both a cumulative and on a differential basis. The combined reactivity worth of the Aft Central Shield Plate, BATH Central Shield and Lead Central Shield is seen to be $+19.7\% \pm 10\%$.

The effect of the addition of the Aft Central Shield Plate and the BATH and Lead Central Shields on the core axial fission density is found by comparison of the fission distributions from ANISN cases E and F noted on Table 6-10. The two fission density profiles are shown in Figure 7-1. The computed core inlet value is seen to be 0.32 ± 0.03 an increase of 13% over the inlet value without the IS. This value of 0.32 is relative to an axial average of unity.

7.5.1.1.2 Design Operation

In a manner analogous to that described in Section 7.5.1.1.1, the reactivity worth and the effect of the IS on the reactor's fission density distribution at design operations has been calculated.

Reactivity effects for all non-fuel regions are summarized in Table 6-11. Effects have been computed on both a cumulative and on a differential basis. The combined reactivity worth of the Aft Central Shield Plate, the BATH Central Shield and the Lead Central Shield is seen to be $+5.8\% \pm 10\%$.

The effect of the addition of Aft Central Shield Plate and the BATH and Lead Central Shields on the core axial fission density is found by comparison of the fission distributions from ANISN cases J and K noted on Table 6-11. The two fission density profiles are shown in Figure 7-2. The inlet value is seen to be 0.51 ± 0.05 , an increase of 5% over the inlet value without the IS.

In order to determine the effect of the homogenizing technique used in the one-dimensional ANISN calculations and neutron streaming through the holes in the poison plates, R-Z cell calculations were performed with the DOT code for a single fuel cluster. The average inlet power distribution is found to be approximately 15 percent higher than the one-dimensional ANISN calculations. Thus, the normalized* computed inlet value for case K, 0.51, increases to 0.59 ± 0.06 . This is an average inlet fission density. Reference 5 shows that a variation of ± 7.5 percent can be expected about this nominal value. This local effect has not been shown on Figure 7-2.

* Normalized to a core average value of 1.0.

7.5.2 Graphite Core

7.5.2.1 Steady State

Two steady state operational modes are examined in this section. The first relates to the ambient reactor. Full power design operation is the second mode. The treatment is analogous to that appearing in Section 7.5.1.

7.5.2.1.1 Ambient Conditions

The ANISN calculations for the NSS at ambient conditions are listed in Table 6-2. These have been based on the fuel atom density model appearing in Table 6-6 and the non-fuel atom density model in Table 6-9. As was noted above, the computed effect of the IS and, in general, any non-fuel region, is obtained in a cumulative manner. Thus, the computed effects of the Aft Central Shield Plate and the BATH and Lead Central Shield are taken to be the difference between 1) a simulation which includes all regions in Table 6-2 through the Support Plate Plenum and 2) a simulation identical to 1) but with the "Shield" added.

Reactivity effects for all non-fuel regions are summarized in Table 6-13. Effects have been computed on both a cumulative and on a differential basis. The combined reactivity worth of the Aft Central Shield Plate, BATH Central Shield and Lead Central Shield is seen to be $+16.0\% \pm 10\%$.

The effect of the addition of the Aft Central Shield Plate and BATH and Lead Central Shields on the core axial fission density is found by comparison of the fission distributions from ANISN cases EE and FF noted on Table 6-12. The two fission density profiles are shown in Figure 7-3. The inlet value relative to an axial average of unity is seen to be 0.29 ± 0.03 , an increase of 12% over the inlet value without the IS.

7.5.2.1.2 Design Operation

In a manner analogous to that described in Section 7.5.2.1.1, the reactivity worth and the effect of the IS on the reactor's fission density distribution at design operation has been calculated.

Reactivity effects for all non-fuel regions are summarized in Table 6-13. Effects have been computed on both a cumulative and a differential basis. The combined reactivity worth of the Aft Central Shield Plate, BATH Central Shield and Lead Central

Shield is seen to be +4.8%

The effect of the addition of the Aft Central Shield Plate and the BATH and Lead Central Shield on the core axial fission density is found by comparison of the fission distributions from ANISN cases JJ and KK noted on Table 6-13. The two fission density profiles are shown in Figure 7-4. The inlet value relative to an axial average of unity is seen to be 0.44 ± 0.04 , an increase of 4% over the inlet value without the IS.

As discussed in Section 7.5.1.1.2, the computed inlet value must be increased by a nominal 15% to account for neutron streaming through the holes in the poison plate and disc. Thus, the nominal value of the core inlet fission density increases to 0.51 ± 0.05 . A variation of ± 7.5 percent can be expected about this nominal value. This local inlet effect has not been shown in Figure 7-4.

7.6 REFERENCES

1. L. I. Ortenberg, "Evaluation of the WANL Integrated Nuclear, Radiation and Shielding Standard Design Method," WANL-TME-2689, September 1970
2. C. M. Brammer, "NERVA Reactor Component Alternates," DRM-51068 (N), "October 5, 1970
3. A. Y. Lee, Computer Run TNTM158, October 2, 1970
4. A. Y. Lee, Computer Run TNTG381, September 29, 1970
5. S. Salah and F. Torres, "Operating Reactivity Worth of Materials on the Aft and Forward Ends of the R-1 Reactor," DRM-51515, July 21, 1969
6. E. Iwinski, "Nuclear Computer Analysis Problem Logs and Tape Listing," DRM-53365, to be published

Figure 7-1. Effect of IS on Axial Fission Density Distribution (Ambient, Composite Core)

Figure 7-2. Effect of IS on Axial Fission Density Distribution (Hot, Composite Core)

Figure 7-3. Effect of IS on Axial Fission Density Distribution (Ambient, Graphite Core)

Figure 7-4. Effect of IS on Axial Fission Density Distribution (Hot, Graphite Core)

8.0 REFLECTOR ASSEMBLY

8.1 INTRODUCTION

8.1.1 Description of Aluminum Structure Reflector ⁽¹⁾

The Reflector Assembly consists of a hollow right circular cylinder which encompasses the core from its forward to aft surface. The basic load-bearing member is an aluminum support structure to which are attached beryllium segments required to fulfill the primary function of neutron reflection. The reactivity control function is satisfied by 18 control drums which are mounted within the reflector and rotate on bearings mounted on each end of the reflector. Rotation of the drums changes the position of the control plates to vary the reactivity.

The major components of the RA from nuclear considerations are 1) the aluminum structure, 2) reflector segments, 3) control drums, and 4) the lateral support system. These components are analyzed in detail in this RA evaluation. The forward and aft rings, locating cylinder and standoffs are not directly included in this analysis due to their axial location relative to the core.

8.1.2 Content of Chapter

This chapter summarizes the nuclear analysis performed on the Reflector Assembly. The major neutronic characteristics of the Nuclear Subsystem utilizing the Reflector Assembly were evaluated for both the composite ⁽²⁾ and the HCTE graphite ⁽³⁾ fuel element designs. The significant results of this analysis pertain to the total control drum span of the NSS reference design.

The Reflector Assembly requirements are reviewed and the reference reflector design is evaluated based on its ability to satisfy these requirements.

The method of analysis employed in performing this design evaluation as well as data on the calculational models used and the status/maturity of the input data are referenced.

The validity of the analytical method is ascertained through comparison with experimental results, including the development of the total control drum span experimental normalization factors.

8.2 SUMMARY OF RESULTS AND CONCLUSIONS

(U) This section presents a summary of the results and conclusions regarding the nuclear analysis of the REFLECTOR ASSEMBLY.

8.2.1 Control Drum Span

(U) The predicted NERVA total control drum spans for the reference REFLECTOR ASSEMBLY are \$5.40 and \$5.35 for the composite and graphite fueled reactors. These predictions correspond to an 80 percent probability and a 95 percent confidence level. Both of these drum spans meet the \$4.5 minimum control drum requirement.

(U) It should be noted that the above drum spans were developed for the reference core, periphery, lateral support and reflector designs. The results of the Core Periphery Trade Study, No. 778, and the Lateral Support Trade Study, No. 782, may result in a design concept being chosen which has a significant effect on the control drum span.

The above predictions do not include the following considerations:

- a) The analytical-experimental comparisons were primarily based upon the PAX experiments which did not include all significant R-1 design characteristics,
- b) NRX experience demonstrated a continuing decrease in drum span between measured reactors having the same basic design. The net reduction in measured drum span between NRX-A2 and XE-PRIME was \$0.80.

(U) Inclusion of the latter considerations would result in drum spans of \$4.60 and \$4.55 for the composite and graphite fueled reactors in a flight engine.

8.2.2 Neutron Reflection

(CRD) The REFLECTOR ASSEMBLY is required to be compatible with a maximum graphite loading of 630 mg/cc, a maximum composite loading of TBD mg/cc and an initial \$1.5 positive reactivity at assembly prior to shimming. Both of the NERVA calculational models incorporated an initial \$1.5 shim content, and, as reported in Chapter 2.0, the maximum uranium loading for the graphite element having 43 grams/element of niobium is 620 mg/cc. This 620 mg/cc meets the 630 mg/cc loading requirement.

(U) It is noted that 4.7 inches of beryllium at 68 percent density are required.

8.3 REQUIREMENTS

8.3.1 Specification Identification

The governing specification for this component is Specification Number EC-677559, Part I, Performance/Design and Qualification Requirements for Reflector Assembly, Forming a Part of CP-677555, NERVA Nuclear Subsystem.

8.3.2 Applicable Requirements

The requirements listed in EC-677559 which directly affect the nuclear design analysis follow:

3.1.1.1.3.1.1 Neutron Reflection. - The RA shall provide the neutron reflection consistent with the following criteria:

- (a) The reactivity variation capability of the RA shall be as specified in 3.1.1.1.3.1.4 (Reactivity Control).
- (b) The RA shall be compatible with a maximum fuel loading in a graphite matrix core of 630 mg/cc and in a composite core of (TBD) mg/cc.
- (c) The RA shall be compatible with a Nuclear Subsystem Module plus the Nozzle at 540°R average core and reflector temperature having an initial positive reactivity of \$1.50 at assembly prior to shimming.

3.1.1.1.3.4.1 Control Drum Reactivity Span. - The RA shall incorporate 18 control drums which provide the capability of reactivity adjustment of 540°R and without propellant flow of \$4.50 minimum based on a cold critical shutdown margin of \$1.50. Further, this reactivity span shall be based on rotation of the drums through angular positions ranging from 5 degrees to 175 degrees, a total interval of 170 degrees.

8.4 METHOD OF ANALYSIS - DESIGN PRACTICE

The procedure employed in the Nuclear Analysis of the Reflector Assembly is specified in this section. The analysis is consistent with the overall design engineering procedure presented in Section 2.4. The calculational models for both the composite and

graphite fuel designs are referenced. The validity of the analytical approach used to predict the R-1 total control drum span is presented by means of an uncertainty analysis based on PAX/R-1 experimental results. Finally, the basic assumptions concerning the input data used in this analysis and the sources of neutron cross sections employed in the calculations are referenced.

8.4.1 Analytical Procedure

The analytical procedure employed in evaluating the Reflector Assembly is specified in NSS Section 2.4.1.3.

8.4.2 Analytical Model

The analytical models employed for the two dimensional analysis of the Reflector Assembly for both composite and graphite fueled cores are described in detail in Section 2.4.2.

8.4.2.1 Problem and Tape Log

A list of the various program tapes, the problem log, and corresponding output tapes pertinent to this analysis can be found in Reference (14). Source references to these problems are found in the lower right corner of all tables and figures where applicable.

8.4.3 Status/Maturity of Input Data

The status/maturity of the input data is presented in Section 2.4.3 of the NSS chapter.

8.4.4 Validity of Analysis

In order to verify that a particular reactor design does provide the required control drum span, Nuclear Design performs extensive neutronic calculations. Although the procedure followed in these calculations is rather detailed, ⁽⁴⁾ experience has shown that there is a bias between the calculated drum span values and that measured for the reactor simulated by the calculation. The magnitude of the bias for a number of reactors tested recently is summarized in Table 8-1. The ratio of the measured to calculated drum span-- those values appearing in the right-hand column of the table--is the information that will be used in the following discussion to develop 1) an average value for the experimental/analytical bias (Experimental Normalization Factor) and 2) the safety margin that must be allowed in order to

assure that the drum span of the reactor design being evaluated will satisfy the design requirement without unnecessary penalty to the NSS design.

8.4.4.1.1 Development of the Total Control Drum Span Experimental Normalization Factor

To allow use of a common statistical method for the calculation of the average experimental normalization factor and its variability, a first step is the determination of the underlying population frequency distribution. Toward this end, Figure 8-1 has been prepared. This figure shows the five data points plotted on a probability scale of Cumulative Frequency.⁽⁵⁾ When data satisfying a "normal" distribution is plotted in such a fashion, it will fall on a straight line. The extent to which this is not the case will be an indicator of the non-normality of the sample. For the five data points here, it appears that the non-normality is not great enough to be detected by the available sample. This allows, then, the statistical treatment of the data to proceed using the equations from "normal" distribution theory to calculate the parameters that characterize the population from which the five data points have been taken. These calculations are summarized in Table 8-2. The estimate of the population mean (M) and standard deviation of a single measurement (S) is found to be 0.861 and 0.032*, respectively. In a previous analysis, reported in Trade Study 776 (See Reference 13), the values of M and S were found to be 0.873 and 0.02*, respectively. The revised values are a result of inclusion of the experience in the prediction of the worth of Drum B in the test sector of the PAX-G3 reactor. The PAX-G3 data was not available for the earlier analysis. The Cumulative Frequency distribution appearing in the Trade Study is compared in Figure 8-1 with that resulting from the present analysis.

The lower portion of Table 8-2 shows the manner in which the parameters M and S are used to develop a measured to calculated ratio consistent with the desired probability and confidence levels. The ratio to be applied to the calculated drum spans in order to assure that the drum span of the reactor design being evaluated will satisfy the requirement

*The standard deviation on the ratio accounts for the experimental uncertainty and is consistent with the value obtained by repeat measurements of the same configuration.

is shown in the last column in this lower portion of the table. It has been calculated as $(M - KS)$ where K is a factor to account for 1) the size of the sample used to estimate the population parameters, 2) the desired confidence level, and 3) the fraction of the population that is to be included. These values are plotted in Figure 8-2.

From the specified Nuclear Criteria in the NSS Detail Specification (CP-677555A), the "Control Drum Reactivity Worth" (Section 3.3.1.3.2) for preliminary design is to be selected on the basis of an 80% probability of meeting the control drum reactivity span requirement with a confidence level of 95%. Thus, it can be seen from Figure 8-2 that the appropriate value for the correction factor to be applied to calculated drum worth values is 0.782. However, the best estimate or predicted drum span is to be based on a correction factor equal to the mean value, 0.861.

8.5 RESULTS, ANALYSIS AND INTERPRETATION

8.5.1 Composite Core

8.5.1.1 Steady State

Total Control Drum Span

The total control drum span was developed using the Method of Analysis described in Section 2.4.1.3. Utilizing the composite core calculational model in conjunction with the 16-group regionwise spectrally weighted cross sections (See Section 2.4.1.3), R-0 EXTERMINATOR problems were set up for the 0° and 180° drum bank positions. The difference in reactivity between these two problems ($\rho_{180^\circ} - \rho_{0^\circ}$) corresponds to the R-1 calculated control drum span for the reference composite fuel and the reference Reflector Assembly. As summarized previously, the reference R-1 control system consists of 18 drums with a control vane outer diameter of 4.0 inches. The control vane is made of $B^{10}Cu$ with a B^{10} surface density of 0.100 grams/cm^2 . The results of these two EXTERMINATOR calculations indicate the calculated control drum span to be \$6.90.

In Section 8.4.4, the drum span experimental normalization factors were developed. These were generated utilizing the following reactors: NRX-A6, PAX-GO, PAX-G1, PAX-G2, and PAX-G3. The experimental normalization factor developed statistically from these

individual values for an 80% probability and a 95% confidence level, to insure a \$4.5 drum span, is 0.782. Using this 0.782 coefficient and the EXTERMINATOR drum span calculation, the final total control drum span for the composite fuel is \$5.40 ($0.782 \times \6.90). Table 8-3 summarizes the drum span results.

In the NSS Specification, the specified requirement for the drum span in preliminary design is at least \$4.5 based on an 80% probability and a 95% confidence level. Based on this requirement, the drum span value of \$5.40 meets this requirement.

As developed in Section 8.4.4, the same experimental data results in an average drum span normalization factor of 0.861. Utilization of this 0.861 factor corresponds approximately to a 50% probability and a 95% confidence level. Using this value in conjunction with the EXTERMINATOR drum span calculation results in a predicted nominal R-1 drum span of \$5.95 ($0.861 \times \6.90).

In reviewing the above predicted drum spans, it should be noted that no allowance was made for future design changes which could reduce drum span. However, NRX reactor experience has clearly demonstrated that the drum span had decreased due to design revisions from reactor to reactor. Table 8-4 presents a summary of the measured drum spans for each of the NRX reactors. As shown, the drum spans decrease from \$9.0 for NRX-A2 to \$7.6 for the NRX-A6 reactor. In view of the above discussion it is recommended that the present NSS drum span be retained to allow a margin for future design changes which would affect the drum span.

Furthermore, the current predicted NSS drum spans were generated without the presence of instrumentation in the reactor. Inclusion of instrumentation will probably result in a reduction in total drum span.

8.5.1.2 Transient

Effect on Total Control Drum Span

The neutronic cold-to-hot effects which were included in this analysis of the REFLECTOR ASSEMBLY are consistent with the results of the TNT model used in the S-001 (Systems Analysis Summary Report) and the S-031 (Thermal and Fluid Flow Analysis Report) Data Items. In this model there is a reduction in the control drum span in going from ambient

to the hot operating conditions of 6%. In TNT, this drum span reduction is caused by the insertion of peripheral hydrogen flow in the operating conditions. This 6% drum span reduction is applied to both the composite and graphite fuel element designs.

8.5.2 Graphite Core

8.5.2.1 Steady State

8.5.2.1.1 Total Control Drum Span

The R-1 total control drum span was developed using the Method of Analysis described in Section 2.4.1.3. Utilizing the graphite core calculational model in conjunction with the 16-group regionwise spectrally weighted cross sections (See Section 2.4.1.3), R-0 EXTERMINATOR problems were set up for the 0° and 180° drum bank positions. The difference in reactivity between these two problems ($\rho_{180^\circ} - \rho_{0^\circ}$) corresponds to the R-1 calculated control drum span for the graphite fuel and the reference Reflector Assembly. As summarized previously, the reference R-1 control system consists of 18 drums with a control vane outer diameter of 4.0 inches. The control vane is made of $B^{10}Cu$ with a B^{10} surface density of 0.100 grams/cm^2 . The results of these two EXTERMINATOR calculations indicate the calculated control drum span to be \$6.84.

In Section 8.4.4, the drum span experimental normalization factors were developed. These were generated utilizing the following reactors: NRX-A6, PAX-G0, PAX-G1, PAX-G2, and PAX-G3. The experimental normalization factor developed statistically from these individual values for an 80% probability and 95% confidence level, to insure a \$4.5 drum span, is 0.782. Using this 0.782 coefficient and the EXTERMINATOR drum span calculation, the final total control drum span for the graphite fuel is \$5.35 ($0.782 \times \6.84).

In the NSS Specification, the specified requirement for the drum span in preliminary design is at least \$4.5 based on an 80% probability and a 95% confidence level. Based on this requirement, the drum span value of \$5.35 meets this requirement.

As developed in Section 8.4.4, the same experimental data result in an average drum span normalization factor of 0.861. Utilization of this 0.861 factor corresponds approximately to a 50% probability and a 95% confidence level. Using this value in conjunction with the EXTERMINATOR drum span calculation results in predicted nominal R-1 drum span of \$5.90 ($0.861 \times \6.84). Table 8-3 summarizes these drum span results.

8.5.2.1.2 Reactivity Coefficients

(U) Table 8-5 summarizes the predicted R-1 material reactivity coefficients in the various radial regions. The measurements referred to on Table 8-5 were obtained from the PAX-R1 series of critical reactor configurations.

(CRD) In addition to the material worths summarized in Table 8-5, the neutronic worth of an LiF impregnation material was determined. The use of an impregnant in the lateral support seals and plunger pins was considered due to its possible friction reduction capability. R-θ EXTERMINATOR calculations were performed at the 0° and 180° drum positions (Drawing No. 982D273 reflector design) to determine the effects on reactivity and drum span of the LiF Impregnation in the lateral support seal segments and plunger pins. Approximately 5.2 kg of lithium were added in these calculations which resulted in a reduction in reactivity (90°) of approximately -\$3.1 and a reduction in calculated total control drum span of approximately -\$3.3. These results demonstrate the nuclear unacceptability of such a large amount of Li added into the lateral support region. Other impregnation material such as the CaF should be more acceptable based on reactivity considerations.

8.5.2.2 Transient

Effect on Total Control Drum Span

(U) (See Section 8.5.1.2)

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TABLE 8-1
SUMMARY OF NRX-A6 AND PAX/R-1
MEASURED AND CALCULATED DRUM SPANS

Measurement Made In	Reactor Simulated in Analytical Model	EXTERMINATOR-2 Calculated Drum Span, \$	Drum Span Inferred from Measurement, \$	Ratio Measured/ Calculated
NRX-A6	NRX-A6	a 8.50	e 7.60	0.894
PAX-G0	Test Fuel Sector in PAX-G0	d 7.04*	b 6.26	0.889
PAX-G1	R-1 Sector in PAX-G1	d 8.00*	c 6.89	0.861
PAX-G2A	R-1 Sector in PAX-G2A	c 7.84	d 6.64	0.847
PAX-G3	R-1 Sector in PAX-G3	g 5.93	h 4.84	0.816

*Adjusted for $\beta = 0.0070$ (Ref. f).

- a Ref. 14
- b Ref. 6
- c Ref. 7
- d Ref. 8
- e Ref. 9
- f Ref. 10
- g Ref. 11
- h Ref. 12

TABLE 8-2
DRUM SPAN RATIO STATISTICS

Estimate of Population Parameters

Point i	R _i	Absolute Value of (R _i -M)	Cumulative Frequency
1	0.894	0.033	0.90
2	0.889	0.028	0.70
3	0.861	0.0	0.50
4	0.847	0.014	0.30
5	0.816	0.045	0.10

M = Estimate of the population Mean = 1/5 (Sum of the R_i values) = 0.861

S = Estimate of the population standard deviation calculated from the above sample:

$$\sqrt{\frac{\sum (R_i - M)^2}{(5-1)}} = \sqrt{\frac{0.004094}{4}} = 0.032$$

Population Fraction and Confidence Level

Population Fraction	Confidence Level, %	K*	KS	(M-KS)
0.99	95	5.741	0.184	0.677
0.95	95	4.202	0.134	0.727
0.90	95	3.407	0.109	0.752
0.75	95	2.149	0.069	0.792
0.99	90	4.666	0.149	0.712
0.95	90	3.400	0.109	0.752
0.90	90	2.742	0.088	0.773
0.75	90	1.698	0.054	0.807

* From Table V (Tolerance Factors for Normal Distributions - One Sided), SNPO-C-1, "NERVA Program Structural Design Requirements," 12/19/68.

TABLE 8-3
SUMMARY OF R-1 DRUM SPAN RESULTS *

	Exterminator Calculated Drum Span (A)	Final Drum Span 80% Probability 95% Confidence Level (A x 0.782)	Predicted Nominal Drum Span 50% Probability 95% Confidence Level (A x 0.861)
I <u>Composite Fuel</u>	\$6.90	\$5.40	\$5.95
II <u>Graphite Fuel</u>	6.84	5.35	5.90

*An approximate 5 cent reduction was included in these drum spans due to the aft end beryllium removal. No instrumentation was included in these drum span predictions. Based on NRX experience, inclusion of subsequent design changes will tend to reduce the above R-1 reactivity control.

TABLE 8-4

SUMMARY OF MEASURED DRUM SPANS FOR THE NRX REACTOR SERIES

Reactor	Measured Drum Span, \$
NRX-A2	9.0
NRX-A3	8.8
NRX-A4	8.8
NRX-A5	8.2
XE-PRIME	8.2
NRX-A6	7.6

TABLE 8-5
R-1 REACTIVITY COEFFICIENT PREDICTIONS

1. CORE

URANIUM	+12.10 ¢/kg	(measurement)
CARBON	+ 3.90 ¢/kg	(measurement)
NIOBIUM	- 4.68 ¢/kg	(measurement)
NIOBIUM (WIRES)	- 3.84 ¢/kg	(measurement)
"X" MATERIAL	- 9.30 ¢/kg	(measurement)
"X" MATERIAL (WIRES)	- 5.60 ¢/kg	(measurement)
IRON	- 0.02 ¢/kg	(calculation)
NICKEL	- 1.60 ¢/kg	(calculation)
"Q" MATERIAL (WIRES)	+ 0.05 ¢/kg	(measurement)

2. LATERAL SUPPORT

CARBON	+ 2.00 ¢/kg	(calculation)
"X" MATERIAL	- 4.80 ¢/kg	(calculation)
"Q" MATERIAL	+ 0.16 ¢/kg	(calculation)
HYDROGEN	+163.00 ¢/kg	(calculation)

3. ALUMINUM CYLINDER

CARBON	+ 1.70 ¢/kg	(calculation)
ALUMINUM	+ 0.73 ¢/kg	(calculation)
HYDROGEN	+156.00 ¢/kg	(calculation)

4. BERYLLIUM SECTOR

BERYLLIUM	+ 1.30 ¢/kg	(calculation)
INCONEL	+ 0.30 ¢/kg	(calculation)
HYDROGEN	+ 13.20 ¢/kg	(calculation)

Figure 8-1. Examination of Drum Span Ratio Data for Normality

Figure 8-2 Drum Span Correction Factor for Various Proportions (P) and Confidence Levels

9.0 CONTROL DRUM AND DRIVE ASSEMBLY

9.1 INTRODUCTION

Stuck control drum analysis will be included in a subsequent revision of the S-101 Nuclear Analysis Report.

10.0 STRUCTURAL SUPPORT COOLANT ASSEMBLY

10.1 INTRODUCTION

The location of the Structural Support Coolant Assembly to the core is such that it has no significant nuclear effect.

11.0 NSS INSTRUMENTATION

11.1 INTRODUCTION

The NSS Instrumentation has not been selected. Therefore, information will be presented in S-054 TS 792 - NSS Instrumentation.

The neutronic effect of instrumentation has been assumed to be 20¢ based on previous experience. This reactivity effect has been included in the fuel loading analysis.

TITLE DATA ITEM S-101 Nuclear Analysis Report, Nuclear Subsystem	
	DOCUMENT NO. WANL-TME-2758	DATE
ABSTRACT <p>The purpose of this report is to document the Nuclear Analysis of the NUCLEAR SUBSYSTEM (NSS) and its Engineering Critical Components (ECC). This report constitutes the primary source of all steady state and transient nuclear data. The detailed Nuclear Analysis consisted of the following:</p> <p>Support for the derivation and establishment of appropriate performance and design requirements for the NSS and its ECCs, Establishment and substantiation of the appropriate parameters of the selected design.</p>		
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