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NRX-A6 DESIGN REPORT

VOLUME II

THERMAL, FLUID FLOW AND NUCLEAR ANALYSIS OF THE NRX-A6 REACTOR

Prepared By:

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(Title Unclassified)

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SUMMARY

Volume II of the NRX-A6 Design Report covers the results of the additional thermal, fluid flow, and nuclear analyses performed since the 29 June 1965 review meeting to support the NRX-A6 design effort. These analyses were of the "design release" type similar to those performed for previous reactor designs which permit hardware definition and indicate areas requiring more detailed analyses and/or development testing.

The salient results of these analyses are as follows:

- 1) The lateral support seal system can function satisfactorily with any of the three designs being considered. The Minimum Leakage Methane Generating Periphery design imposes the most stringent design requirements. However, the analyses show that the desired temperature distributions can be attained by appropriate sizing of pyrographite insulating foil and tile.
- 2) Use of the Minimum Leakage Methane Generating Periphery indicates an advantage in terms of the corrosion damage anticipated for the peripheral fuel element. However, attainment of this improvement is dependent upon the demonstration of low peripheral leakage being maintained over the operating life of the reactor.
- 3) The initial reflector flow balance and steady state full power temperature distributions indicate that in the mid-core region (peak heat flux region), the average reflector temperature is about 410°R with peak temperatures (less than 600°R) occurring in the region 1.5 - 2.0 inches from the inside diameter. Minimum temperatures occur at the sector webs (about 200°R in the outer web and 360°R in the inner web) and at the webs between the ID coolant holes (approximately 280°R). These temperature distributions are being utilized to determine steady state stresses in the reflector. Transient analyses, corresponding to a 150°/sec start-up chamber temperature ramp, are presently underway.

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- 4) The core reactivity available with present uranium fuel densities is sufficient to meet the anticipated test requirements with over a 35 reactivity margin, thereby allowing the use of up to 400 core elements of the high void fraction type (110 mil reamed channel diameter). Incorporation of increased uranium loading will permit use of a full core of elements of the 37 percent void fraction type (110 mils reamed channel diameter).
- 5) The required reactivity worth of the control drums is 7.05 and is within the capability of the control drum configuration.
- 6) The core weight loss due to fuel element bore corrosion for a 60 minute full power run will be approximately 32 kg. This combined with surface corrosion weight loss, is estimated to result in an outward drum rotation of approximately 40 degrees.
- 7) Concurrent analysis of the reactor configuration at higher power density and flow rate indicates feasibility with no additional major problem area precluding success.

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SECTION I INTRODUCTION

BACKGROUND

The objective of the NRX-A6 design has been to insure a 60 minute full power core lifetime and a test bed for the major core, periphery, and reflector design features of larger size reactors. The design has been derived from previous NRX-A and larger reactor studies (WANL-TME-1200, 1212, 1221, 1186) and the evolution of these studies to a configuration within the NRX-A operating and geometry envelope. This portion (Volume II) of the NRX-A6 Design report documents the additional thermal, hydraulic, and nuclear analyses performed to support the NRX-A6 design effort.* It should be emphasized that these results are the design release type analyses similar to what has been performed on previous reactor designs which allow definition of the hardware and indicate areas requiring more detailed analysis and development testing in parallel with the reactor hardware procurement.

The basic goals for the NRX-A6 design are as follows:

- 1) The reactor will serve as a core test bed for extending the core lifetime to insure 60 minutes of operation at full power and provide the capability to permit fuel development testing of the higher void fraction fuel planned for larger reactors.
- 2) The reactor design shall incorporate a basic reactor structure which can be applied to reactors of greater power density and size.

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* The reader is referred to Volume I for a detailed description and discussion of the mechanical design aspects of the NRX-A6.

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- 3) The reactor design should provide increased component design margins and eliminate problem areas revealed by previous designs to improve reactor reliability for proto-type engine development.
- 4) The reactor design should have sufficient flexibility to incorporate alternate core periphery and lateral support system configurations leading to the minimizing of core corrosion problems.

With these general design objectives in mind, a design effort was undertaken to evaluate the various design alternatives and modifications. The results of this effort were presented at the review meeting of 29 June 1965 and are summarized in WANL-TME-1221. Following this review, the design effort has been directed towards the more detailed design definition of a reactor meeting the stated objectives to allow power testing in January 1967.

DESIGN CRITERIA FOR NRX-A6 REACTOR

Particular design criteria which have evolved for the NRX-A6 are as follows:

- 1) The core interior and envelope will be essentially that of the NRX-A5 reactor with provisions for incorporation of high void fraction fuel elements.
- 2) The method of fuel element cluster support utilizing tie rods attached to the core support plate will be the same as the NRX-A5. The support block design shall evolve from the present development program.
- 3) The geometry of the periphery shall be such as to allow the incorporation of the Minimum Leakage Method Generating Periphery (MLMGP),* the Cooled Periphery with Pyrographite on the Inside of the Filler Strip (CPPIFS),* or the Cooled Periphery with Pyrographite on the Outside of the Filler Strips (CPPOFS).* Since the MLMGP imposes the most stringent design conditions, it has been studied the most extensively since the last design review in June 1965. The selection of the periphery design for use in the NRX-A6 will be based upon the NRX-A4 and NRX-A5 reactor tests, development tests and analyses.
- 4) The reflector system will eliminate use of the graphite inner reflector and will be constructed from three beryllium stacked rings. It will include the NRX-A5 design control drum.

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* Definition of periphery & terminology (See page 1-3)

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- 5) The reflector will be supported at the nozzle end.
- 6) The reactor shall be capable of startup and shutdown chamber temperature ramp rates of $150^{\circ}/\text{sec}$.
- 7) The fuel distribution shall provide sufficient excess reactivity to meet the anticipated operating reactivity loss due to fuel corrosion, and shall be as high as possible, to allow design latitude for possible future increase in NbC coating or channel diameter.
- 8) The control span shall be adequate relative to the excess reactivity, core operating time, and the shutdown reactivity requirements.

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* Minimum Leakage Methane Generating Periphery (MLMGP) - This consists of sacrificial graphite used in conjunction with seal segments designed to permit only a minimal axial leakage flow. The attempt in this design is to assure that any hydrogen available to leak into the core will contain a large fraction of methane as compared with the equilibrium fraction at local core temperatures. In previous reports and discussions this has been called a "hot periphery" since heat is removed from the filler strips and seal segments by the reflector and thus run hotter than in the present design.

* Cooled Periphery with Pyrographite on Inside of Filler Strips (CPPIFS) - This is the NRX-A2, A3, and A4 periphery where the heat generated in the filler strips and seal segments is removed for the most part by coolant flowing through the seal system.

* Cooled Periphery with Pyrographite on Outside of Filler Strips (CPPOFS) - This is the so-called "hot buffer" periphery and is essentially the same as the CPPIFS except that the pyrographite tile on the OD permits the filler strip to run at approximately local core temperature. The heat generated within the filler strips is removed primarily through the core and filler strip coolant holes. This periphery is of the same design as the two sectors being placed in NRX-A5.

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SECTION II

DESIGN CONFIGURATION

DESCRIPTION OF THE REACTOR

A detailed description of the reactor configuration is contained in the companion volume this report, Volume I, Configuration, Design and Mechanical Analysis of the NRX-A6 Reactor.

This section of the report is limited to those parts of the design which are germane to the thermal, fluid flow and nuclear analyses.

An overall schematic of the reactor is shown in figure 2-1. It shows the major components and flow paths for the base configuration of the reactor incorporating Minimum Leakage Methane Generating Periphery.

Reflector

The outer reflector consists of three beryllium rings stacked end to end and held together by 24 tie bolts, 12 along the ID and 12 along the OD located mid-way between the control drums. Six of the former bolts extend to the nozzle flange and provide the principal reactor support. The flow passages in the reflector consist of 0.368 inch diameter at the inside diameter, 96 channels, 0.250 inch diameter holes in the beryllium rings, and the cooling annuli about the tie bolts, control drums, and at the OD to cool the pressure vessel. The beryllium rings are held together by the tie bolts and sealed from the lateral support system by an aluminum O-ring. The seal segment plungers are loaded by springs buried in radial holes in the reflector, which are cooled by conduction to the beryllium across a very small hydrogen gap. A typical cross-section of the reflector is shown in figure 2-2. The coolant holes at the ID and in the rest of the reflector are suitably orificed to attain appropriate flow distributions to minimize thermal gradients for both the MLMGP, CPPIFS and CPPOFS design configurations.

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NRX-A6 REACTOR FLOW SCHEME

Figure 2-1. Overall Schematic of Reactor

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Figure 2-2. Reflector Configuration Showing Hole Pattern

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Lateral Support, Seal System, and Filler Strips

The lateral support, seal system, and filler strip configuration is shown in figure 2-3 and 2-4. The lateral support system consists of 24 rows of plungers (24 plungers per row) acting upon either seal segments or load segments, loaded by the plungers and support springs buried in the beryllium reflector. The first two dome end rows of plungers are acted upon by heavy springs to provide a simulated "core band" loading. As can be seen from figure 2-1 the load segment region is vented to core inlet pressure. The first load segment has the configuration of a seal. It is designed to act as an ineffective seal to permit easy flow communication between the core inlet and seal region and to prevent particles from the seal system from reaching the core inlet. The seal system, starting at core station 17.5 inches consists of rows of seal segments of the single ledge and seal design. These seals are designed to permit only a minimal flow resulting from leakage through scarf joints and interfaces to pass along the core periphery in the MLMGP design. Seal grooves, if required, can be readily machined in the seal segments for controlled coolant flow should either of the CPPIFS or the CPPOFS design concepts be selected. The load segments along the first third of the core provide sacrificial graphite for corrosion in order to furnish methane contaminated hydrogen for flow along the periphery and inflow into the core. These load segments can be maintained at a predetermined temperature level by placing a suitable thickness of pyro-graphite between the load ring and the filler strips. The filler strips, as shown in figure 2-3, are provided with coolant channels to maintain their temperature at approximately core temperature for the MLMGP. For either of the "cooled" periphery designs, this channel can be heavily orificed or the filler strips may be solid, as required.

Core

The core interior is the same as NRX-A5. In the periphery, all partial elements have been eliminated and replaced with unfueled graphite suitably cooled with a single channel to maintain them at approximately the same temperature as the local core temperature. A typical peripheral region is shown in figure 2-3.

CYCLE OPERATING CONDITIONS

The 100 percent power and flow rate operating conditions have been prescribed to be essentially the same as for the NRX-A2 through NRX-A5 reactors. Since the core and

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Figure 2-3. Lateral Support System & Core Periphery

Figure 2-4. Lateral Support System (Axial Section)

Figure 2-5. Estimated NRX-A6 Startup Profile

Figure 2-6. Estimated NRX-A6 Shutdown Profile

shield are essentially identical with the NRX-A5 reactor, the initial estimates of the cycle operating points have been made on this basis of the NRX-A3 operation. These are shown in table 2-1 for steady state full power. It should be noted that the tie rod tube system is a single pass instead of a counterflow arrangement.

As defined by the reactor test objectives (see section 1), the transient operation is governed by normal maximum chamber temperature ramps of 150°R/sec for both startup and shutdown, as shown in figures 2-5 and 2-6, respectively.

TABLE 2-1. ESTIMATED NRX-A6 STEADY STATE CYCLE DATA

Condition	Unit	Value
POWER	MW	1120
FLOW	LB/SEC	71.3
NOZZLE INLET		
Pressure	PSIA	882
Temperature	°R	50
REFLECTOR INLET		
Pressure	PSIA	736
Temperature	°R	150
CORE INLET		
Pressure	PSIA	685
Temperature	°R	247
NOZZLE CHAMBER		
Pressure	PSIA	569
Temperature	°R	4090
TIE ROD COOLANT		
Exit Temperature	°R	698

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HEATING RATES

Presented in figures 2-7 and 2-8, respectively, are the average axial heating rate distributions in the NRX-A6 lateral support and beryllium reflector. Figure 2-9 gives the relative radial distribution for the lateral support and beryllium reflector. Each radial distribution is normalized such that the area weighted average equals unity.

The axial distributions were derived from the NRX-A4 analysis given in the Reactor Analysis Data Book (WANL-TME-1210) because of the similarity between the A4 and A6 reactors in the axial direction. The derivation of heating holes from the A4 analyses are considered satisfactory at this stage of design. The data includes the correction factor to calculated values of 1.235 which was determined from the NRX-A2 and A3 hot test analyses. The radial distributions were based on a radial one-dimensional analysis performed on a preliminary NRX-A6 design.

The accuracy of this data is believed to be within ± 30 percent for region integral values and within ± 100 percent for maximum local deviations. More complete analyses are presently underway leading to improved heating rate values with reduced uncertainties.

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Figure 2-7. Lateral Support System Average Axial Heating Rate Distribution

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Figure 2-8. Beryllium Reflector Average Axial Heating Rate Distribution

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Figure 2-9. Radial Heating Rate Distribution in Lateral Support and Reflector Systems

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SECTION III

THERMAL AND FLUID FLOW ANALYSIS

REFLECTOR SYSTEM

Reflector Flow Balance

The cooling hole pattern and flow annuli configuration for the reflector, as shown in figure 2-2, differs somewhat from the NRX-A5 reactor. The principal differences are the absence of a graphite inner reflector and the extension of the beryllium outer reflector inward to the lateral support system boundary. To remove the heat generated in the MLMGP a row of large (0.368 inch diameter) cooling holes are located close to the ID in order to minimize thermal gradients in the beryllium. In addition, the number of cooling holes in the beryllium has been reduced in an effort to raise the temperature level in the reflector and thereby reduce the peak-to-average thermal gradients. As previously discussed, the control drums are essentially the same as in the NRX-A5 reactor and incorporate the extra cooling slot to alleviate drum bowing during transient operation. Thus, the coolant passages within the reflector system consist of the following:

ID Cooling Holes	- 192, 0.368 inch diameter
Ring Cooling Holes	- 84, 0.250 inch diameter
Tie Bolt Annuli	- 24, 0.047 inch clearance
Pressure Vessel Annulus	- 0.128 inch clearance
- (estimated at full power operation based upon 0.140 inch at assembly)

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Control Drum	- 228, 0.1875 inch diameter
Cooling Holes	
Control Vane	
Inner Annulus	- 120°, 0.125 inch clearance
Control Vane	
Outer Annulus	- 120°, 0.082 inch clearance
Control Drum Slot	- 110°, 0.115 inch clearance
Control Drum	- 0.076 inch clearance
Annulus	

The flow balance reported herein assumed that: 1) neither the ID cooling holes or the ring cooling holes are heavily impeded at entrance or exit, 2) the pressure vessel annulus was heavily impeded at the inlet and exhausted to the reflector exit plenum and 3) the control drum had the same impedance characteristics as the NRX-A5 drum.

The analysis used the MCAP-TOSS computer code to determine temperatures and pressures in the coolant and material temperature distributions in the components. Heating rates for the materials in this region were obtained from section 2. A distributed flux was imposed on the inner reflector wall to simulate heat transfer from the seal system. The outer surface of the pressure vessel was assumed to be adiabatic, forcing all heat generated in the pressure vessel to flow into the annulus coolant.

Table 3-1* lists the hydraulic diameter of each channel, the number of equivalent channels and the entrance and exit loss coefficients assumed in analysis. Figure 3-1 indicates flows, coolant temperatures, and pressures to be expected in the NRX-A6 reflector and control drum region.

Reflector Temperature Distributions

Steady State Full Power Operation

Reflector temperature distributions have been obtained from the MCAP-TOSS analysis previously mentioned. Typical material temperatures are shown in figures 3-2 and 3-3 for core diameters 30.4 inches and 49.8 inches, respectively. The material temperatures in the vicinity of the spring are those for a thin layer of beryllium about the spring cavity. At

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* See top of page 3-6

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Figure 3-1. NRX-A6 Reflector Flow Schematic

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Figure 3-2 Typical Temperature Distribution in NRX-A6 Reflector at Core Station 30.4 Inches

Figure 3-3. Typical Temperature Distribution in NRX-A6 Reflector at Core Station 49.8 Inches

TABLE 3-1. REFLECTOR HYDRAULIC DATA

Identification	Section	Equivalent Diameter (Inches)	Equivalent Number of Channels	Inlet Loss Coefficient	Exit Loss Coefficient*
Inner 4 Drum Holes	1	0.1875	48.0	2.50	-5.50
Middle 8 Drum Holes	2-3	0.1875	96.0	1.60	-6.30
Outer 7 Drum Holes	4	0.1875	84.0	4.28	-1.39
Inner Vane Annulus	5-8	0.2500	126.0	4.15	-1.09
Outer Vane Annulus	9-12	0.1640	189.0	4.56	-1.84
Drum Slot	13	0.2000	120.0	1.60	-7.80
Drum Annulus	14-16	0.1520	622.2	22.70	-144.60
Reflector Holes (0.368 inch dia.)	17-18	0.3680	192.0	1.50	0.20
Pressure Vessel Annulus	19-21	0.256	386.8	100.00	0.20
Reflector Holes (0.250 inch dia.)	22-25	0.250	84.0	1.50	0.20
Tie Bolt Annulus	26-27	0.0940	247.2	1.50	0.20

*Positive exit loss coefficient denote pressure recovery.

core station 30.4 (which is in the vicinity of the maximum heat fluxes), the average reflector temperature is about 410°R with the outer surface temperature varying from 184°R at the control drum web to a maximum of 296°R midway between the drum axis and the tie bolt axis. The inner wall temperatures vary from 348°R at the control drum web to 546°R at the plunger. At core station 49.8 inches both the temperature levels and differentials are reduced, as would be expected from the reduced coolant temperatures and heat generator rates. Temperature profiles for radial slices at the plunger centerline (section A-A on figure 3-2) and removed from the plunger (section B-B on figure 3-2) are plotted separately in figure 3-4 for core station 30.4 inches. As can be seen, peak temperatures occur at about 1.5 to 2.0 inches from the ID and peak at about 590°R for section A-A and 490°R for section B-B. It should be pointed out that the flow balance and

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Figure 3-4. Radial Temperature Profile at Core Station 30.4 Inches

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temperature distribution results presented here are merely the initial ones. From these results stress analysis will show the magnitude of the stresses which will exist in the reflector at steady state 100 percent power and flow operating conditions. In the event that the stress levels are excessive, the flow distribution may be changed to achieve acceptable thermal stress levels.

Transient Startup and Shutdown

The transient temperature distribution for the nominal maximum startup and shutdown chamber temperature ramp rate of $150^{\circ}/\text{sec}$ is presently being analyzed for the reflector geometry and orificing used above.

Detailed Study of Temperature in the Vicinity of the ID Cooling Holes

A single row of cooling holes has been placed in the reflector 19.21 inches from the core centerline to aid in the removal of heat generated in the seal region. These holes, as shown in figure 2-2, are 0.368 inches in diameter and are on 0.728 inch centers except in the vicinity of the plunger pins. In this region it was necessary to double this spacing distance to accommodate the spring cavity. Since the thermal gradients in this region have been of design concern, it was desirable to determine temperatures as accurately as possible. Thus, two TOSS models were developed, one considering the region where the holes are evenly spaced along 0.728 inch centers and the other considering the wide spacing in the vicinity of the plunger pins. The models extended from the inner surface of the reflector to a depth of 0.749 inches. For this investigation, it was assumed that an adiabatic boundary would occur at the seal-filler strip interface so that all heat generated in the seal region would flow into the reflector. Preliminary analysis revealed that maintaining an adiabatic rear boundary (at the 0.749 inch depth) would generate more severe thermal gradients than similar cases where the boundary was maintained at a constant temperature equivalent of the average reflector temperature. All cases studied, thereafter, incorporated the "worst case" adiabatic rear wall.

From the MCAP-TOSS analysis discussed previously coolant temperatures and heat transfer coefficients at core station 26 in the 0.368 inch diameter coolant holes were found to be

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1790R and 4000 Btu/hr ft² respectively. This, coupled with the maximum heat flux profile from the seal region were input as boundary conditions to the TOSS model.

Analysis of the nominally spaced hole region indicated temperature scalloping at the reflector ID surface of the order of 5 - 100R between hole centerline and the mid-point between holes. However, with the wide spacing near the plunger holes there was a noticeable rise in temperature. Figure 3-5 shows that a temperature difference of 1250R may be expected along the inner surface with the highest temperature occurring at the centerline of the pin cavity. This value is in reasonable agreement with those shown in figure 3-2. The temperature levels are different due to the variation in the location of the adiabatic line between that assumed in this small model and that calculated on the large reflector model.

Spring Temperatures

As indicated in section 2, the plunger pins are forced against the seal segments by a coil spring inserted into a cavity in the reflector. In the NRX-A6 design, cooling of these parts is provided only by conduction across the hydrogen gap, through the reflector, and into the cooling channels. The springs will, therefore, operate at a higher temperature level than the adjoining beryllium of the reflector. The spring geometry was included in the MCAP-TOSS analysis discussed in section 3. The following three dimensional effects were considered in the analysis: 1) heat transfer radially across a hydrogen gap to the reflector 2) heat transfer axially across a hydrogen gap to the adjoining coil and 3) heat conduction along the coil.

A typical radial profile in the spring is shown in figure 3-6 for core station 30.4 inches. As indicated the spring temperatures peak at approximately the same radial location as the beryllium temperatures. There is also apparently significant conduction along the spring both to the OD of the reflector and to the plunger. From the plot, it can be seen that the peak temperatures are just below the design limit of 12000R and that the volume weighted average would fall significantly below this value. This is shown in figure 3-7 where the volume weighted average, minimum, and maximum spring temperatures are plotted for each axial station. It should be pointed out that both the spring thickness and wall clearance have been reduced subsequent to this analysis. This will reduce the temperature levels in the spring. In addition any reduction in beryllium temperatures which occurs as the design evolves will result in a further reduction in spring temperatures.

Figure 3-5. Local Temperature Distribution at ID of Reflector at Core Station 26 Inches

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Figure 3-6. Spring Temperature Profile at Core Station 30.4 Inches

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Figure 3-7. Spring Temperature vs. Core Station

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Control Drum Transients

Test results of previous NRX-A1 and A3 reactors showed that high control drum torques were experienced during startup ramps. This indicates that the control drum bowed temporarily during the early stage of flow initiation. In trying to equalize the chilldown on both sides of the control drum cylinder, such that the temperature difference across the diameter of symmetry may be reduced, the NRX-A5 drum design incorporates a machined slot in the drum cylinder opposite the vane annulus. This modification of the drum cylinder is incorporated into the A6 reactor. Transient thermal analysis of this redesigned drum is presented in this section.

Figure 3-8 shows a $150^{\circ}\text{R}/\text{sec}$ (chamber temperature ramp) startup profile used for the analysis. The power schedule during this startup profile is also shown in figure 3-8. This profile was developed with the NRX-A3 reactor model. However, from steady state thermal analysis results of the NRX-A6 and the NRX-A3, it is seen that the flow rates within the control drums are similar. It is therefore, considered that this profile is also applicable to the NRX-A6 analysis. Figures 3-9 and 3-10 show the temperature difference across the diameters AA' and BB' with two different sizes of slots. With the smaller slot (90 degree arc length and 0.10 inch deep), the maximum ΔT across AA' is about 60 degrees R, but across BB' it is about 105 degrees R. The high temperature difference across BB' is mainly due to the difference in heating of the drum. The heating at point B is higher than that at B' because B is closer to the core center. With the larger slot (120 degree arc length and 0.125 inch deep) the A side is being overcooled. Calculations performed by Reactor Mechanical Design show that the thermal bow of the drum with the large slot is about 0.011 inches (A' side convex) and with the small slot about 0.025 inches (A side convex). The final slot size to be used in NRX-A5 and for the NRX-A6 is 110 degree arc length by 0.115 inch deep. With this size of slot, the thermal bow during a $150^{\circ}\text{R}/\text{sec}$ startup ramp should be very small. Further transient thermal analysis to verify this is now underway.

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Figure 3-8. A 150°R/Sec Startup with Flow Lag

Figure 3-9. 150°R/Sec Startup Control Drum Temperature Difference

Figure 3-10. 150°R/Sec Startup Control Drum Temperature Difference

LATERAL SUPPORT SYSTEM

As previously described in section 2, the base design configuration in the lateral support and seal region is the minimum methane generating periphery (MLMGP). No additional analyses are being presented in this section on either of the design alternates (cooled periphery with pyrographite on inside of filler strips or cooled periphery with pyrographite on outside of filler strips) since both of these have been thoroughly evaluated in the prior work performed on the NRX-A3, NRX-A4, and NRX-A5 reactors.

Inherent in the MLMGP design is the control of leakage flow and temperature in the load and seal segment regions. This section of the report primarily covers the additional analyses performed to evaluate these aspects of the design.

Seal Pressure Distribution and Leakage

The lateral support system is designed such that the forward 17.3 inches, the load segment area, has free access to the core inlet pressure*. The remainder of this region is filled with 16 pairs of seal and spacer segments designed to reduce axial hydrogen flow to a minimum. The resulting lateral support system pressure distribution is given in figure 3-11.

Axial flow in the seal area can follow three parallel paths, namely, between the inner seal segments and filler strips, between the outer seal segments and the reflector, and, finally, across the conical contact surface between the inner and outer seal segments. In a parametric study of leakage flow rate, the three paths are expressed as an equivalent annular passage with a given gap width. Figure 3-12 shows lateral support system axial leakage as a function of gap width for an equivalent single flow path. Testing to define the equivalent flow path for the actual NRX-A6 components, as assembled, is planned. It is described later in this section of the report.

The interelement pressure distribution and the radial flow of seal system hydrogen into the core interelement gaps which result from the lateral support system pressures of figure 3-11 are discussed later under Surface Corrosion Analysis.

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*Note that the first load segment has the configuration of a seal. However, it is modified to provide free flow access and still act as a particle catcher.

Figure 3-11. Lateral Support Region Axial Pressure Distribution

Figure 3-12. Estimated Seal System Leakage (Nozzle End) Versus Seal Segment Equivalent Gap Size

Temperature Distribution in Seal Segment, Spacers, and Filler Strips

The base configuration of the lateral support system incorporates the minimum leakage methane generating periphery with sacrificial graphite over the first third of its length. The purpose of the sacrificial graphite is to provide methane contaminated hydrogen for core in-flow thereby reducing the corrosion of the peripheral fuel elements. In order to meet the corrosion requirements of these load segments, it is important to be able to tailor the axial temperature profile in the load segments. Since the core will run at temperatures below that desired in the load segments pyrolytic graphite insulation between the filler strips and load segments must be used to maintain the desired temperature levels in the load segments.

Initial sizing of the required pyrolytic graphite thickness to attain a 3000°R temperature at the load segment pyrolytic graphite interface has been performed based upon a one-dimensional heat transfer model. The model includes appropriate core and reflector boundary temperatures with conduction into the core and both conduction and radiation into the reflector. The results are plotted in figure 3-13, and show that a thickness of 0.150 inches is required at core station 4.5 inches and this thickness decreases to approximately 0.020 inches at core station 17 inches. Above core station 4.5 inches use of 0.150 inches of pyrographite will permit these segments to run cool enough to prevent any significant corrosion. A typical radial temperature distribution is shown in figure 3-14 for core station 17 inches. As shown there, the maximum temperature within the load segment using 0.022 inches of pyrographite is 3170°R with the reflector side running at approximately 3150°R. Due to the greater thermal resistance on the reflector side most of the heat generated within the load segments will be transferred across the pyrographite and into the adjacent filler strip. However, since the heat generation rates are low in this region, the effect on filler strip temperatures will be small.

Similar one-dimensional analyses have been performed for the seal segment region along with more detailed finite difference methods using the TOSS computer code. The latter model consists of the region bounded by the fuel element coolant channels and an adiabatic line just beyond the inner row of coolant holes in the beryllium reflector. Figures 3-15 and 3-16 show the results of these calculations at core station 30 inches with 0.050 inches of pyrographite on the reflector side of the outer seal segment.

Figure 3-13. Equivalent Pyrographite Thickness Required in Load Segments to Attain 3000°R Temperature

Figure 3-14. Temperature Profile in Load Segment at Core Station 17

Figure 3-15. One-Dimensional Temperature Distribution in Seal Region at Core Station 30 Inches

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TOSS MODEL
 INCLUDES FUEL ELEMENT
 BOUNDARY, FILLER
 STRIPS, SEAL SEGMENTS,
 AND APPROXIMATELY
 1.5 INCHES OF THE
 BERYLLIUM REFLECTOR.
 BOUNDARY CONDITIONS
 ARE BULK TEMPERATURES
 AND FILM COEFFICIENTS
 IN FUEL ELEMENT
 CHANNELS AND IN
 COOLING HOLES AT ID
 OF REFLECTOR.

Figure 3-16. Seal Segment Temperature Distribution as Obtained from TOSS Inches for Core Station 30

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A typical azimuth temperature distribution in the filler strips as obtained from the TOSS model for core station 30 inches is shown in figure 3-17, the temperature levels on the seal are greater than between seals due to the transfer of heat from the seal segments. In general, the deviation from the average temperature is within $\pm 200^{\circ}\text{R}$, and this can be reduced by appropriate orifice sizing and hole location within the filler strips.

Also plotted on figure 3-15 is the temperature profile along path AA of figure 3-16. As can be seen, the one-dimensional analysis is in reasonable agreement with the TOSS model as to the average temperature level at any radial position and its profile. Thus, it is capable of predicting average radial temperature gradients. However, since it is a one-dimensional analysis it cannot show the axial variations in the seal segment temperatures which result from the conduction and radiation across the hydrogen and from the shape of the segments themselves.

The temperature levels desired in the seal region from a corrosion standpoint are shown in figure 3-22 of section 3. These levels are stated to be approximately 100°R lower than the maximum fuel temperature at the edge of the core at the start of life for the aft end of the core, and a constant 3000°R average temperature from core station 17.5 inches to core station 26.0 inches. Since the temperatures of the fuel elements vary considerably (as elsewhere described in this section) from start to end of life, the pyrotile thicknesses will only meet these requirements for one point during the operating life. The actual design point for the sizing will be selected upon the basis of parametric core and seal region corrosion studies which are presently being carried out. At this point, it appears that there would be no difficulty in sizing pyro tile to maintain seal segment temperatures above the core temperature. However, where seal temperatures lower than the core temperature are desired the use of additional pyrolytic graphite tile will shift the location of the adiabatic line toward and perhaps into the core, thereby, increasing the heat flux through the seal segments and the temperature gradients across the seal segments. A compromise between the tolerable temperature gradients resulting in seal segment distortion and temperature levels resulting in increased corrosion, will have to be made. Analyses are presently under way which will show the design direction to achieve a near optimum configuration.

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UPPER LEVEL
(CORRESPONDING TO THE SAME ELEVATION IS
THE INNER SEAL SEGMENT)

LOWER LEVEL
(CORRESPONDING TO THAT PORTION OF THE FILLER
STRIP BETWEEN THE INNER SEAL SEGMENT AND THE
AFT END OF THE OUTER SEAL SEGMENT)

Figure 3-17. Filler Strip Temperature Distribution at Core Station 30 Inches

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Spring Cavity Venting Requirements

As shown in figure 2-2, the lateral support springs are buried in a dead-end (i.e. plugged) radial hole in the beryllium ring. During transient operation additional pressure forces will act upon the plunger if pressures are not balanced across it. Due to the necessity for maintaining small clearances about the plunger during operation it was deemed necessary to provide a central vent hole in the plunger. The basis for this decision is shown in figure 3-18 which shows the time lag in balancing pressures across the plunger, assuming isothermal flow conditions.

SURFACE CORROSION ANALYSIS

Background

One important aspect of the NRX-A6 design is the improvement in the outside surface corrosion of the peripheral fuel elements which can occur by means of the Minimum Leakage Methane Generating Periphery (MLMGP). Essential to the design philosophy is that the graphite lateral support system components be kept at a high enough temperature that the axial flow in this region chemically reacts with the graphite present. This results in corrosion of the seal region components and subsequent contamination of the hydrogen stream with hydrocarbons. (In this particular case, the material temperature level is such that the only significant hydrocarbon is methane.) With the seal cavity filled with hydrogen carrying an important fraction of methane as compared with the equilibrium fraction at the local core temperature, radial inflow from the seal cavity is expected to bring about lower amounts of peripheral element corrosion on the average than can be attained with a periphery such as that used in NERVA reactors through NRX-A5.

This portion of the report will include:

- a) A general description of the method of corrosion analysis and its limitations
- b) Information on lateral support system and interelement flows and pressures, which are intimately tied in with the corrosion analysis
- c) Lateral support system component corrosion estimates
- d) Results of NRX-A6 core corrosion calculations to include comparisons among the NRX-A6 design alternates and with an NRX-A2 through A5 cooled seal system for an equivalent operating time.

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Figure 3-18. Time Lag to Pressurize or Depressurize Spring Cavity
 as a Function of Plunger Clearance

Method of Analysis

The calculation of corrosion in the NRX-A6 core consists of hydraulic analyses, corrosion calculations in the lateral support region, and, finally, the calculations of corrosion in the core proper. The calculational sequence is essentially the same as that described more completely in section 1.4.b of WANL-TME-1210, Reactor Analysis Data Book.

The hydraulic analysis is performed by means of the THT-B computer code which calculates core interelement pressures based upon a given interelement geometry, fixed core inlet pressure, core exit pressure, and the seal system axial pressure distribution.

The corrosion analysis is begun by calculation of corrosion of the seal system components and the attendant buildup of methane in the hydrogen flowing axially through the seal cavity. Fuel element external surface corrosion is then calculated by means of the TDC-2A code. TDC-2A requires as input: 1) core interelement flows as deduced from the pressure distribution calculated by THT-B, 2) the core material temperatures, and 3) the methane content (as a function of axial station) of the hydrogen in the seal region. The final corrosion results are in the form of a two dimensional (R-Z geometry) map of fuel element surface corrosion.

For further details on the calculational tools, to include the chemical constants used in the corrosion calculations, see section 1.4.e.1.1 of WANL-TME-1210.

It must be emphasized that the entire analysis is based on average conditions, in particular, average interelement gaps (leading to average interelement flow) and average material temperatures. Locally more severe corrosion due to higher than average flows or temperatures is not taken into account. (See section 1.4.e.1.2, sheets 5 and 6 of WANL-TME-1210 for the comparative effects of locally worse corrosion conditions.)

Hydraulic Analysis

Figure 3-19 shows the fuel element and filler strip gaps used in the hydraulic analysis. It should be noted that where thermal and bundling force considerations predict no physical gap between filler strips, a 0.25 mil hydraulic leakage gap was assumed. The average interelement gap is seen to be about 0.5 mils. Figure 3-20 shows the boundary conditions of inlet, outlet, and lateral support system pressures as well as two interelement axial pressure distributions within the core resulting from the THT-B calculations. The calculated

Figure 3-19. Interelement and Filler Strip Gap Width used in NRX-A6 Core Hydraulic Analysis, Minimum Leakage Methane Generating Periphery (MLMGP)

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Figure 3-20. NRX-A6 Interelement Gap Axial Pressure Distribution for 116 PSI Core ΔP , Minimum Leakage Methane Generating Periphery

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core interelement pressure distribution results in a core interelement flow distribution as given on figure 3-21.

Corrosion Analysis

The first portion of the corrosion analysis involves the seal system components. The corrosion depth of graphite components in the seal region is shown in figure 3-22. For component temperatures and the ratio of seal system flow to load segment area as indicated on the figure, the maximum total corrosion depth of 21 mils for 60 minutes of full power operation is predicted for the forwardmost load segments. Temperatures in the seal segments are specifically set (curve on right edge of figure 3-23) according to two criteria to take advantage of the fact that the equilibrium concentration of methane in hydrogen decreases with increasing temperature. In the first place, the segment temperatures are maintained at a constant 3000°R between stations 17 and 26, then progressively increased for all but the last few inches of the core. By this arrangement, the flow which is almost saturated with methane relative to the 3000°R level, carries enough of the hydrocarbon to be completely saturated at the higher temperatures to which the downstream segments have been raised. Consequently, the segments downstream of about 26 inches should suffer virtually no corrosion. The second criterion for the seal segment temperatures is that they should be lower than core temperatures. The reasoning is the same as was outlined above; with cavity flow at methane saturation at a lower temperature, radial inflow cannot cause corrosive damage of the hotter core material. Consequently, the temperature profile shown on figure 3-23 is lower than minimum expected peripheral fuel element temperature for approximately the aft half of the core. With the material temperatures described and explained, the hydrogen flow rate remains to be defined. Assuming that the outside surface of the load segments constitutes the area for corrosion, a leakage flow rate out of the lateral support system of 0.5×10^{-3} lb/sec is assumed in the corrosion calculations presented in the next few figures.

The seal cavity temperatures and flow just described lead to a progressive change of methane mole fraction in the seal region hydrogen as depicted on figure 3-23. On the same curve the equilibrium mole fraction of methane in the interelement hydrogen flow appears by way of comparison. For purposes of defining interelement temperature, an axial core temperature profile was estimated from figure 3-34. This represents, of course, the maximum

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Figure 3-21. Interelement Velocity Map - NRX-A6 Minimum Leakage Methane Generating Periphery (MLMGP)

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Figure 3-22. Corrosion of NRX-A6 Lateral Support System Components, Minimum Leakage Methane Generation Periphery

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Figure 3-23. Comparison of Lateral Support System Methane Mole Fraction with Core Equilibrium Hole Fraction

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amount of methane which can be carried in the hydrogen flowing between elements in the core. For reasons given in the previous paragraph, wherever seal region hydrogen carries more methane than interelement hydrogen can thermodynamically contain, radial inflow will suppress element corrosion in the core. The estimated corrosion depth of the outermost edge of fuel material and the depth for an element at the core centerline are shown on figure 3-24. The analysis is for 60 minutes of full power operation. Maximum depths are 16 mils at the core periphery and about 1 mil at the core center. Figure 3-25 shows the radial distribution of maximum corrosion depth. The corrosion decreases to core centerline corrosion levels within a short distance from the periphery. Estimated filler strip corrosion is plotted on figure 3-26. No filler strip corrosion is expected past about Station 23, which is important in terms of the ability of the periphery to restrict leakage flow to a minimum.

The corrosion calculations were repeated for an estimate of the most rapid possible buildup of methane in the seal region. This would be the most advantageous form of the NRX-A6 Minimum Leakage Methane Generating Periphery (MLMGP). An accelerated buildup of methane in the seal region would be associated with extremely small seal segment leakage flows and/or a design providing extensive corrosion reacting graphite surface within the space limitations of the lateral support system region. An estimate of this fastest possible buildup is given in figure 3-27, along with the core equilibrium methane mole fraction. The resulting corrosion of the outermost fueled material is compared in figure 3-28 with the previously calculated A6 corrosion, as a demonstration of the maximum possible improvement in the peripheral picture. The maximum corrosion depth is reduced from 16 to 11 mils. In this "best possible" case, the temperature selected for the load segments was 3000°R. Use of a lower temperature would allow a greater amount of methane to be carried in the seal system hydrogen (equilibrium concentration of methane in hydrogen increases with decreasing fluid temperature), but the reaction rate (methane production rate) becomes so low at temperatures materially less than 3000°R that only the most minute flow rate could pass through the seal system and still be adequately contaminated. The flow rate to reacting area ratio used to calculate the "best possible" curves in figures 3-27 and 3-28 is 0.624×10^{-6} lbs/sec. in², or using the outside surface of the load segments as the corroding area a leakage flow rate of 0.2×10^{-3} lb/sec.

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Figure 3-24. Corrosion of NRX-A6 Outermost Fueled Material, Minimum Leakage Methane Generating Periphery

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Figure 3-25. Radial Distribution of Maximum Corrosion Depth

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Figure 3-26. Corrosion of NRX-A6 Filler Strips, Minimum Leakage Methane Generating Periphery

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Figure 3-27. Comparison of Most Rapid Lateral Support System Methane Buildup with Core Equilibrium Mole Fraction

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Figure 3-28. Corrosion of MRX-A6 Outermost Fueled Material with Estimated Best Possible Methane Buildup, Minimum Leakage Methane Generating Periphery

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Figure 3-29 is an interesting comparison of core peripheral corrosion for the present NERVA reactor seal system design and the three NRX-A6 alternates. The corrosion depth of the outermost edge of uncoated fuel material is plotted as a function of axial location for

- 1) NRX-A6, Cooled Periphery with Pyrographite Tile on the Inside of the Filler Strips (CPPIFS)
- 2) NRX-A4
- 3) NRX-A6, Cooled Periphery with Pyrographite Tile on the Outside of the Filler Strips (CPPOFS)
- 4) NRX-A6, Minimum Leakage Methane Generating Periphery (MLMGP), 0.5×10^{-3} lb/sec leakage flow
- 5) NRX-A6, MLMGP, 0.2×10^{-3} lb/sec leakage flow.

All designs are analyzed for 60 minutes of full power operation. The maximum corrosion depths are, in the same order as the designs are listed above, 46, 33, 30, 16, and 11 mils. This figure demonstrates the peripheral corrosion reduction possible with the hot periphery concept. A comparison of filler strip corrosion for the 3rd and 4th designs (CPPOFS and MLMGP) appears on figure 3-30. The advantages of the corroding periphery design are again apparent.

The methane buildup in the seal system hydrogen (and, therefore, fuel element corrosion at the core periphery) is an important function of seal system axial flow and load segment reaction surface. Load segment configuration should be modified to offer as much surface possible for corrosion without compromising structural requirements on the segments. It is even more important to minimize leakage flow by the necessary combination of sound design and close tolerances. Designing the seal system in such a way as to restrict axial leakage now to about 10^{-3} lb/sec seems at this point necessary to the success of the hot periphery (MLMGP) design.

Summary

- 1) The NRX-A6 hot periphery design philosophy appears to be valid. That is, by a proper choice of the seal system axial flow rate and seal system component material temperatures, corrosion of the outermost uncoated fueled material can be significantly suppressed.

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Figure 3-29. Corrosion of NRX-A6 Outermost Fueled Material, Comparison Among NRX-A6 Alternates

Figure 3-30. Corrosion of NRX-A6 Filler Strips, Comparison Between Periphery Designs

- 2) The same mechanisms which reduce maximum core peripheral corrosion have the bonus advantage, by proper choice of seal segment temperatures, of leaving the seal segments uncorroded in approximately the aft half of the core.
- 3) Radial inflow should not induce core peripheral corrosion even if locally extreme inflow should occur. This is true for the presented seal segment temperature distribution (with seal segment temperatures constantly increasing in the axial direction as described in the previous paragraph). The segment temperatures are maintained somewhat lower than the core temperature over the aft end of the core.

CORE

The core design changes and their implication to the reactor with respect to thermal and fluid flow performance of the reactor have been reported in WANL-TME-1221. Of prime consideration is the fuel element bore corrosion anticipated for a 60 minute run. This corrosion is related to changes in reactivity which result in drum position changes that will be experienced with time at power. As a consequence, different temperature patterns for the central and peripheral regions of the reactor occur. A change in radial power distribution will be experienced as the drums move out which, in turn, will cause the center of the core to become progressively cooler and the core periphery to become progressively hotter. Since corrosion of carbon, which is believed to be the main contributor to this loss of reactivity, is directly related to temperature, the anticipated corrosion within the core will vary according to the drum swing from the start of life to the end of life.

In order to determine the anticipated NRX-A6 corrosion, a parametric study was performed whereby assumed drum swings with their associated core temperatures were obtained. In this analysis the NRX-A6 reactor was assumed to be orificed at a drum angle that yielded equal exit gas temperature in the center and periphery of the core at the beginning and end of life, respectively. With these temperature patterns, the total corrosion was then calculated and the corresponding reactivity losses determined. These calculated reactivity losses were compared with the assumed reactivity losses. The conditions at which the corrosion, and therefore the calculated reactivity loss, equaled the assumed reactivity loss is the solution of the anticipated corrosion and drum swing.

Analyses were performed for three drum swings: 14 degrees, 28 degrees, and 44 degrees. In order to calculate the temperatures for these drum swings, the effects of operating reactivity changes on exit gas temperature in the central and peripheral fuel element are shown in figure 3-31. Temperature deviations from the average core exit gas temperature are given for three orificed angles, 90 degrees, 105 degrees, and 115 degrees. The small spread in the slopes of the corresponding central and peripheral temperature perturbations indicates that within calculational error the absolute position of the orificed angle has no effect on the temperature perturbations caused by a given reactivity change. Based on the criterion that the increase in exit temperature above the orificed condition at the core center at the start of life equals that at the periphery for the end of life, the expected 1\$ net reactivity loss, for example, gives a change of temperature of approximately 60°R, see figure 3-31. The expected deviation for a 2\$ reactivity loss is 135°R and 200°R for the 3\$ reactivity loss. Figure 3-32 shows the variation in radial temperature patterns experienced for each respective drum swing. Figures 3-33, 3-34 and 3-35 illustrate the axial temperature distributions resulting from these drum swings at the core center and core edge for the start and end of life.

The most recent relationship for the mid-band corrosion was used which assumes cracks to exist in the NbC liner between the temperatures of 2000°R and 3371°R. This relationship is expressed quantitatively by the expression:

$$\Delta W = K (T_o - T) (\Delta \bar{\alpha}) \exp (-E/RT) \quad (1)$$

where

T_o = the approximate coating deposition temperature, °R

T = the temperature being considered from the reactor run, °R

$\Delta \bar{\alpha}$ = the difference in the coefficient of contraction of NbC and graphite,
in./in. °R

E = the activation energy for the hydrogen graphite reaction, Btu/lb mole

K = an area factor, gm/min in.

The temperature dependence of the complete expression is shown in figure 3-36. The relationship for hot end corrosion is given on figure 3-37. The inception of hot end

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Figure 3-31. Effect of Reactivity Changes on Central and Peripheral Core Temperatures for Different Orificed Drum Angles

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Figure 3-32. Fuel Element Gas Exit Temperature Change from Average-Shimmed to Achieve Equal Maximum Temperatures at Start and End of Life

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Figure 3-33. Maximum Fuel Temperature at Start and End of Life for 14° Control Drum Swing

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Figure 3-34. Maximum Fuel Temperature at Start and End of Life for 28° Control Drum Swing

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Figure 3-35. Maximum Fuel Temperature at Start and End of Life for 44° Control Drum Swing

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Figure 3-36. Effect of Temperature on Theoretical Corrosion Through Cracks for Reference Temperatures of 3371°R

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Figure 3-37. Weight Loss - Hot End Diffusion

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diffusion is extrapolated to 3600°R. Both of the preceding expressions are used to calculate the fuel element bore weight loss in a code being developed to predict weight loss as a function of core station, core radius, drum angle and reactor operating time. The weight loss is required for use in calculating reactivity losses. Presently the code is used only to determine the total bore weight loss with test temperature patterns and time. The code calculates the total weight loss distribution for reactor test conditions and is being developed to calculate total weight loss distributions for performance predictions. The differentiation is made since the former calculations for test conditions generate axial weight loss distribution as a function of running time, core radius and measured temperatures. The calculations for the latter case, performance predictions, will generate axial weight loss and the feedback for reactivity loss due to this corrosion. The final output generated for this case will be the weight loss and the associated drum change for the run profile.

The bore corrosion weight loss for the reactor is based on the corrosion testing of the fuel elements. The calculated temperature distributions from the fuel element 1B Quality Control corrosion test was analyzed to obtain the integrated area for the curve of $\Delta W/K$ versus length in the corrosion test. This associated area corresponds to a mid-band corrosion of 0.2 gms/min - element which is a constant arrived at from many tests run. Equation 1 cannot be used directly to calculate weight loss since the constant K has not been determined. This constant, K, accounts for the variation in the coating crack width along the length of the element at temperature. It cannot be presently determined since only the overall weight loss in the mid-band region is obtained in the corrosion tests. The mid-band corrosion for reactor conditions is then calculated by:

$$\Delta W_{\text{mid-band}} = \gamma \beta \quad (2)$$

where γ is the integrated mid-band weight loss (gms/min - element) for the reactor conditions and:

$$\beta_{1B \text{ Q.C. test}} = \frac{W_{1B}}{\left[\int_{Z_1}^{Z_2} \frac{\Delta W}{K} (Z) dZ \right]_{1B}} \quad (3)$$

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where $\Delta W_{1B} = 0.2$ gms/min - element. The integrated mid-band weight loss γ is evaluated and β is an input to the code. Figure 3-37 is used directly by the program to calculate the hot end corrosion due to diffusion of carbon through the coating.

Table 3-2 presents the resulting mid-band and hot end bore corrosion losses for the estimated 14 degree, 28 degree and 44 degree drum swings. The start of life and end of life values were averaged for the two core radii, and the number of fuel elements associated with these radii were used to obtain the total core weight loss. The weight loss per hour as indicated by table 3-2 is seen to be essentially independent of the temperature pattern utilized in these calculations. In this particular analysis, the indication is that there is a trade-off between mid-band and hot-end corrosion. Figure 3-38 shows the average total corrosion weight losses based on 30,096 bores in the NRX-A6 reactor. Applying the corrosion distribution calculated by the code for the bore corrosion, and correcting for additional loss due to surface corrosion, the reactivity loss was estimated to be 2.3\$ as described in section 4. From this result then it can be concluded that the drum swing of NRX-A6 will be approximately 40 degrees based on full power operation for one hour.

COMPONENT TEST PROGRAM

The NRX-A6 design configuration, as discussed in this and previous reports, e.g. WANL-TME-1221, requires component test support in several areas. This section briefly discusses these requirements and their relative priority to meet the thermal and fluid flow information needs peculiar to this design.

Reflector

- a) Impedance Testing (such as the determination of loss coefficients for the pressure vessel annulus impedance device, coolant hole inlet impedance, etc.)
(Priority-2)
- b) Seal Effectiveness Testing (including such items as the beryllium ring seal, nozzle interface seal, and the shield seal) (Priority-2)
- c) Heat Transfer Testing - lateral support spring conduction cooling (Priority-1)
Testing has already been initiated and initial results indicate feasibility of the conduction cooling concept.

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Figure 3-38. Total Weight Loss Versus Time for the NRX-A6 Reactor

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TABLE 3-2. BORE CORROSION ESTIMATES; 60 MIN. FULL POWER OPERATION AT DIFFERENT
 DRUM SWINGS

Core Radius	Drum Swing (degrees)	ΔW gm/element SOL	ΔW gm/element EOI	Average gm/element ΔW	Number of Elements In Region	ΔW kg	Total ΔW kg
Center - 0.754"	14	23.56	19.81	21.68	1014	21.98	32.93
Edge - 17.04"		14.86	23.56	19.21	570	10.95	
Center	28	23.86	17.19	20.52	1014	20.81	31.48
Edge		13.58	23.86	18.72	570	10.67	
Center	44	27.21	16.85	22.03	1014	22.34	33.96
Edge		13.57	27.21	20.39	570	11.62	

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Lateral Support and Seal System

The testing discussed here is pointed primarily toward the Minimum Leakage Methane Generating Periphery.

- a. Flow Characteristics (Priority-1)
  - 1) Sealing surface leakage
  - 2) Scarf joint leakage
  - 3) Radial stepping gap leakage
  - 4) Effect of thermal distortion on sealing effectiveness - i.e., cocking and bow
- b. Heat Transfer Characteristics (Priority-2)
  - 1) Thermal conductivity of ZTA-evaluation of anisotropy
  - 2) Thermal conductivity of pyrographite foil
  - 3) Thermal contact resistance
    - (a) ZTA - ZTA
    - (b) pyrographite tile - ZTA
    - (c) pyrographite foil - ZTA
    - (d) pyrographite foil - beryllium
- c. Corrosion (Priority-1)
  - 1) Pyrographite foil corrosion rate
  - 2) Effect of seal segment thermal gradients on location and severity of corrosion - Initial test results on a simulated seal cavity indicate that for a cavity sufficiently isolated from a source of fresh hydrogen, minimum corrosion will occur in the cavity.
  - 3) Effect of sealing surface corrosion on sealing effectiveness

Core Periphery

- a. Flow characteristics - Determine the direction and magnitude of the peripheral flow (Priority-1)
- b. Corrosion - Comparative evaluation of peripheral corrosion with that obtained in the NRX-A2 through NRX-A4 designs (Priority-1)

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Core (Priority-1)

Continued effort must be expended leading to an understanding and alleviation of the various core considerations which might limit the scalability of the reactor and its operating time. Areas requiring investigation include such items as pinholing, support block cracking and corrosion, bore coatings, isolated interelement channel corrosion, and interelement flow distribution and magnitude.

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SECTION IV NUCLEAR ANALYSIS

The nuclear design of the NRX-A6 reactor is based upon analytical techniques which have been previously developed and continuously improved by comparison with extensive critical experimentation and the NRX-A2 and NRX-A3 hot test data. These analytical methods will be supported by special NRX-A6 critical experiments to insure attainment of nuclear control span requirements, reactivity and power distribution goals. The following sections cover reactivity at ambient and at full power operating conditions, control drum span estimates, selection of shim content, fuel loading, power distribution, the nuclear experimental programs, and nuclear design flexibility.

COMPOSITION AND GEOMETRICAL REPRESENTATION OF THE NRX-A6 REACTOR

The geometrical representation of the NRX-A6 reactor employed in all of the drum span and power calculations presented herein is shown in figure 4-1. The control vane (region 14) is shown as an explicit region in order to allow its representation at various drum angles. The lateral support spring mechanism occupies three regions (regions 9, 10, and 11) since the spring itself extends only halfway into the beryllium reflector, the inner half being occupied by the graphite plunger and a locking screw or pin. The major portion of the beryllium reflector (regions 8, 12, and 13) is represented as three separate regions due to the non-homogeneous nature of its coolant channels. The titanium tie bolts are also represented explicitly, see Region 15.

Tables 4-1 through 4-5 show the masses and atom densities corresponding to the regions represented in figure 4-1. The fuel element compositions given in table 4-1 represent the present best estimate of the required element loadings. For convenience these weights were assigned the corresponding "as-built" weights in NRX-A3, except the code 89 through 39

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A. DIMENSIONS SHOWN ARE FOR THE OPERATING CONDITION
 B. CONTROL DRUM REPRESENTED AT 115 DEGREES
 C. REGION NOMENCLATURE GIVEN IN TABLES 4-2 AND 4-3.

Figure 4-1. Geometric Representation of the NRX-A6 Reactor for Nuclear Design Calculations

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TABLE 4-1. NRX-A6 FUELED ELEMENT WEIGHTS

Loading Code	Number of Elements	Average Uranium Weight (gm)	Total Uranium Weight (kg)	Average Carbon Weight (gm)	Total Carbon Weight (kg)	Average Nb Weight (gm)	Average Nb Effective Weight (gm)	Total Nb Effective Weight (kg)
99	1134	122.876	139.342	530.437	601.516	55	60	68.040
89	78	93.608	7.301	538.989	42.041	55	60	4.680
79	78	81.104	6.326	539.199	42.057	55	60	4.680
69	84	66.470	5.583	544.655	45.751	55	60	5.040
59	36	57.584	2.073	544.349	19.597	64.5	69.5	2.502
49	60	48.742	2.925	545.642	32.739	64.5	69.5	4.170
39	114	34.723	3.958	547.117	62.371	64.5	69.5	7.923

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TABLE 4-2. NRX-A6 CORE WEIGHTS (kg)

Radius Outer Region of Region, CM	cold hot	Region 1 & 2 ⁽¹⁾ 36.408 36.674	Region 3 40.279 40.573	Region 4 41.938 42.244	Region 5 43.500 43.818	Total
Carbon Weight						
a) Fuel Element		601.516 kg	129.849	52.336	62.371	846.072
b) Unfuelled single hole element		115.645	27.829	24.118	11.132	178.724
c) Pyrographite sleeves		17.343	4.173	3.617	1.670	26.803
total		734.504	161.851	80.071	75.173	1051.599
Uranium Weight		139.342	19.210	4.998	3.958	167.508
Niobium Weight						
a) Fuel Element bore coating ⁽²⁾		68.040	14.40	5.760	6.840	95.040
b) Fuel Element Outer coating		00.0	00.0	0.913	1.083	1.996
c) Unfuelled elements		1.496	0.360	0.312	0.144	2.312
total		69.536	14.760	6.985	8.067	99.348
Stainless Steel		5.116	1.231	1.067	0.492	7.906
Inconel-X ⁽³⁾		13.178	3.170	2.748	1.269	20.365
Hydrogen		0.149	0.033	0.015	0.015	0.212

(1) The location of each region is shown in figure 4-1

(2) Effective weight (sine squared)

(3) Hydrogen weights correspond to axial midplane densities used in extermiator problems, not representative of total hydrogen mass in core

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TABLE 4-3. NRX-A6 REFLECTOR WEIGHTS (kg)

Region ⁽¹⁾	Volume ⁽²⁾	Material ⁽³⁾	Weight
6, Filler Strip	62,344.1 cm ³	Carbon Niobium Hydrogen	115.143 kg 0.378 0.000024
7, Lateral Support	91,920.7	Carbon Hydrogen	108.484 0.227
Beryllium Reflector			
8, a) Adjacent to lateral support	57,461.2	Beryllium Hydrogen	74.692 0.193
9, b) Molybdenum Screws	4,382.8	Beryllium Carbon Molybdenum	6.645 1.117 5.003
10, c) Graphite plunger	32,432.8	Beryllium Carbon Hydrogen	47.496 8.507 0.008
11, d) Inconel Springs	54,132.7	Beryllium Aluminum Nickel Iron Chromium Hydrogen	63.288 14.424 31.200 13.620 11.916 0.056
12, e) Reflector region adjacent to tie bolts	262,912.2	Beryllium Hydrogen	484.192 0.039
13, f) Reflector region encompassing control drum	262,811.6	Beryllium Aluminum Hydrogen	424.0369 18.170 0.407
15, g) Tie bolts	3,938.9	Titanium Aluminum	18.070 1.65
14, h) Control drum	4,039.548	B ¹⁰	5.16
Beryllium Reflector (total)	678,408.8 cm ³	Beryllium Carbon Molybdenum Aluminum Nickel Iron Chromium Titanium B ¹⁰ Hydrogen	1,000.35 9.612 5.003 34.244 31.200 13.620 11.916 18.070 5.16 0.703

- (1) The location of each region is shown in figure 4-1.
 (2) Expanded for operating conditions.
 (3) Hydrogen weights correspond to axial midplane densities used in extermiator problems, not representative of total mass of hydrogen in reactor.

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TABLE 4-4. NRX-A6 AXIAL MIDPLANE CORE ATOM DENSITIES $\times 10^{24}$
FOR 100 PERCENT FLOW AND POWER CONDITIONS

Element	Region 1+2 ⁽¹⁾ 36.674 cm	Region 3 40.573 cm	Region 4 42.244 cm	Region 5 43.818 cm
Hydrogen	0.000159	0.000159	0.000159	0.000159
Carbon	0.06665	0.064602	0.069558	0.066713
Iron	0.000089	0.000099	0.000187	0.0000885
Nickel	0.000189	0.000204	0.000385	0.0001814
Chromium	0.000063	0.000068	0.000129	0.000061
U ₂₃₅	0.000591	0.000364	0.000206	0.000166
U ₂₃₈	0.000044	0.000027	0.000015	0.000012
Nb	0.000786	0.000743	0.000784	0.000925
Volume; hot	561,212.733	125,674.12756	57,743.993	56,523.254

(1) The location of each region is shown in figure 4-1.

uranium weights have been reduced to compensate for the higher fission density at the periphery of the NRX-A6 core.

The NRX-A6 composition and geometry as defined above differs from that presented in WANL-TME-1221 "NRX-A6 Design Modifications" as follows:

1. Niobium weight per element has been increased from 43 to 55 grams per element.
2. Core diameter has decreased to reflect a reduction in the number of fueled elements from 1656 to 1584.
3. The total weight of the beryllium reflector has been reduced to 1121 Kg as compared with 1100 Kg.
4. The design shim content is 3.80 S as compared with 4.50 S.
5. The uranium loading is 167.5 Kg compared with 150 Kg.

A more detailed discussion of these differences is given in Appendix A.

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TABLE 4-5. NRX-A6 AXIAL MIDPLANE REFLECTOR ATOM DENSITIES
 FOR 100 PERCENT FLOW AND POWER CONDITIONS

Region ⁽¹⁾	Material	Atom Density $\times 10^{24}$
6) Filler Strip	C	0.092645
	Nb	0.000039
	H	0.0000002
7) Lateral Support	C	0.059201
	H	0.00147
8) Ref. adjacent to Lateral Support	Be	0.086893
	H	0.002006
9) Molybdenum Screws	Be	0.101352
	C	0.012784
	Mo	0.007168
10) Graphite Plunger	Be	0.097895
	C	0.013157
	H	0.000147
11) Inconel Springs	Be	0.078161
	Al	0.005950
	Ni	0.005915
	Fe	0.002714
	Cr	0.002550
	H	0.000618
12) Reflector Region adjacent to Tie Bolts	Be	0.123111
	H	0.000088
13) Reflector Region Encom- passing Control Drum	Be	0.1078566
	Al	0.001544
	H	0.00092
15) Tie Bolts	Tie	0.057704
	Al	0.009355

(1) The location of each region is shown in figure 4-1.

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AMBIENT REACTIVITY

Two methods were utilized to estimate the excess reactivity available in NRX-A6 with the fuel distribution described above. Both methods utilize the NRX-A3 reactor as a base case, since the reactivity available in this reactor has been measured.

In the first method an experimental reactivity coefficient for each chemical element was multiplied by the appropriate mass difference between the two reactors to find the associated reactivity change. The calculation, summarized in tables 4-6 and 4-7 indicated the excess reactivity available in NRX-A6 to be 2.3\$ at 89 degrees. Utilizing the 7.53\$ drum span developed in a later section, the excess reactivity at 180 degrees would be 6.1\$.

It should be noted that in table 4-2 a small reactivity loss, $-.90\%$, is derived as a difference between comparatively large numbers. Since the experimental reactivity coefficients used were derived from small mass changes in the KIWI and PAX reactors, the extrapolation to the large changes in NRX-A6 is subject to considerable error. For these reasons the excess reactivity derived by this method is considered much less reliable than by the method to be described.

The second reactivity estimate was derived by comparing the eigenvalues of consistent R- θ EXTERMINATOR problems for the NRX-A3 and NRX-A6 reactors. In both problems the reactors were represented in an operating condition. After adjustment for the difference between the NRX-A3 result and the actual reactivity, the excess reactivity available in NRX-A6 at 90 degrees was calculated to be 3.8\$, in the unshimmed, cold condition as contrasted with 2.3\$ developed above. For the reasons described above, this number is believed to be more reliable than the result of the previous method and was adopted as the basis for the present report. The critical experiment program discussed later in this section will verify these numbers.

As will be shown in a later section, a cold critical position in the range of 80 to 90 degrees is suitable for the NRX-A6 test profile. The 3.80\$ calculated excess reactivity available at 90 degrees is thus a realistic value for the excess actually required for NRX-A6.

This margin, which may be compared to the 3.4\$ used in NRX-A3 allows the use of high void fraction fuel or heavy Nb coating with some margin for uncertainty. It will be shown later that high void fraction fuel can be used even on a core-wide basis if uranium loading is increased.

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TABLE 4-6. MATERIAL AND REACTIVITY DIFFERENCES IN THE CORE AS A FUNCTION OF RADIUS

MASS DIFFERENCES (kg), NRX-A6 MINUS NRX-A3					
Material	I	II	III	IV	Total
Carbon	5.551	0.528	0.566	-14.080	-7.435
Uranium	6.0	-2.136	-0.555	-1.759	-4.450
Niobium	18.055	3.787	2.762	-0.262	24.342
Stainless Steel	0.0	0.0	0.0	0.0	0.0
Inconel-X	0.0	0.0	0.0	0.0	0.0
Reactivity Differences (¢) NRX-A6 Minus NRX-A3					
Material	I	II	III	IV	Total
Carbon	+34.03	+2.41	+2.35	-53.50	-14.71
Uranium	0.0	-22.00	-6.14	-21.42	-49.56
Niobium	-134.69	-14.12	-8.37	+0.65	-156.53

Reactivity coefficients taken from TME-878, "NRX-A2 Reactivity and Shimming", D. Hill.

NRX-A6 OPERATING REACTIVITY

Cold-to-Hot Reactivity Swing

In table 4-8 the components of the cold-to-hot reactivity change of the NRX-A6 reactor are compared with those for the NRX-A3. The NRX-A6 core in the present reference design is very similar to the NRX-A3, in contrast to the study reported in WANL-TME-1221. The only major difference noted in table 4-8 is the increased reflector temperature in the NRX-A6 design. The net effect of the changes shown is to decrease the cold-to-hot swing by 13 cents. The NRX-A3 prediction of -53 cents loss at full power may be compared with a

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TABLE 4-7. MATERIAL AND REACTIVITY DIFFERENCES BETWEEN NRX-A6 AND NRX-A3

Region	Material	NRX-A6 Mass(kg)	NRX-A3 Mass(kg)	Difference (kg)	Reactivity Coefficient ρ /kg	Reactivity Change(ρ), NRX-A6 Minus NRX-A3
Core	Uranium	167.508	171.958	- 4.450		- 49.56
	Carbon	1052.599	1059.601	- 6.435		- 14.71
	Niobium	99.348	73.572	24.342		-156.53
	Stainless Steel	7.906	7.906	0.0		0.0
	Inconel X	20.365	20.365	0.0		0.0
	Hydrogen	0.011	0.011	0.0		0.0
Filler Strip	Carbon	115.143	94.75	20.383	3.5	71.38
	Niobium	0.378	0.0	0.378	-2.48	- 0.94
Lateral Support	Carbon	108.484	298.96	-190.476	2.0	-380.95
	Steel	0.0	11.82	-11.82	-2.3	27.19
	Aluminum	0.0	26.48	-26.48	0.26	- 6.88
	SiO ₂	0.0	8.102	- 8.102	1.2	- 9.72
Beryllium Reflector Including Control Drum	Beryllium	1100.35	920.424	179.926	3.2	575.76
	Carbon	9.624	0.0	9.624	1.9	18.28
	Aluminum	34.244	19.82	14.424	.26	3.75
	Nickel	31.200	0.0	31.200		
	Iron	13.620	0.0	13.620	-3.0	-170.21
	Chromium	11.916	0.0	11.916		
	Titanium B ₁₀	18.070	27.14	- 8.07	-0.45	3.63
		5.16	5.16	0.0		
TOTAL						- 89.51
A3 excess reactivity 68°F, unshimmed at 89 degrees						\$3.20
A6 - A3 reactivity difference						- 0.90
A6 excess reactivity at 89 degrees						2.30

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TABLE 4-8. COLD-TO-HOT REACTIVITY CHANGES DETERMINED BY
 COHOT COEFFICIENT, 100 PERCENT POWER

<u>NRX-A6 Hot Conditions</u>		
Beryllium Average Temperature	400°R	
Core Temperature	3044°R	
Ambient	528°R	
Operating Power	100%	
Propellant Flow Rate	71.3 lb/sec	
Total Core H ₂ Content	0.359 kg	
Lateral Support H ₂	0.227 kg	
Be Reflector H ₂	0.703 kg	
<u>Temperature Effects</u>	<u>NRX-A3</u>	<u>NRX-A6</u>
Due to core expansion	-1.58	-1.58
Due to thermal bore shift	-0.62	-0.28
Due to Doppler Effect - for U	-0.45	-0.44
Due to Doppler Effect - for Nb	-0.21	-0.28
Due to Doppler Effect - for Ta	-0.19	-0.19
Due to Be Contraction	0.07	0.02
Subtotal of temperature effects	-2.98	-2.75
<u>Hydrogen Effects</u>		
Due to Core Hydrogen	1.70	1.67
Due to Lateral Support	0.38	0.29
Due to Be Reflector	0.03	0.05
Due to change in worth of H ₂	0.00	0.00
Due to release of adsorbed H ₂	0.00	0.00
Due to diffused hydrogen	0.26	0.26
Subtotal of Hydrogen effects	2.45	2.35
TOTAL of Cold-to-Hot Swing (predicted)	-0.53	-0.40
Corrected to NRX-A3 experiment	-0.82	-0.69

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value of -82 cents inferred from EP-IV of the NRX-A3 test. For the purposes of the present review, the calculated 13 cent difference in cold-to-hot swing between the two reactors will be applied to this -82 cent number, giving a -69 cent loss as a tentative estimate for NRX-A6.

The preceding discussion does not include any provision for counterflow tie tubes. The additional hydrogen contained in these would act to increase operating reactivity.

Reactivity Effects of Corrosion

In Section III the bore component of the weight loss of NRX-A6 elements during operation is calculated. This analysis indicated that the total core weight loss due to bore corrosion is insensitive to the drum movement input to the calculations to establish the radial temperature profile. The total weight developed in each of the three cases summarized in table 3-2 is approximately 32 kg. Since the results of the three cases are essentially equal within the calculational error involved, it is not necessary to iterate the procedure to develop a weight loss whose reactivity effect would produce the drum movement initially assumed. As will be shown, the second case, derived on the assumption of a 2.0\$ operating loss, yields a reactivity loss of about 2.3\$.

Since the worth of the carbon loss depends on its axial position, explicit account was taken of the axial distribution of weight loss in each of the cases. Table 4-9 shows this calculation, using the weight loss derived by assuming an initial 2.0\$ reactivity swing and calculating the start-of-life and end-of-life temperature distributions corresponding to this swing. Using the axial reactivity worth profile for carbon, the element at core center at start-of-life was found to have a reactivity-equivalent weight loss in 60 minutes of 23.7 grams, as compared with the actual 23.86 grams shown previously. That is, 23.7 grams of carbon distributed uniformly would have the same reactivity effect as the calculated axial distribution of 23.86 grams carbon. Since the start-of-life and end-of-life weight loss rates are similar (23.7 and 19.6 grams respectively) it is adequate to assume that the average rate of weight loss during life is simply the mean between the values, or 21.7 grams in 60 minutes for the central element. Further, the total weight loss in 60 minutes for the central and edge elements is nearly the same so that the mean between these values is an adequate core average value. Multiplying this average weight of 21.3 grams per

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TABLE 4-9. NRX-A6. OPERATING REACTIVITY LOSS

Element Location	Reactivity - Equivalent Corrosion Per Element in 60 minutes			Core Average Effective Corrosion gm/element	Effective Total Loss in 60 minutes kg	Carbon Worth ¢/kg	Total Reactivity Loss, Dollars
	Start-of-Life	End-of-Life	Average				
Bore Corrosion	23.7	19.6	21.7	21.3	33.7	5.62	1.90
Center							
Edge	18.3	23.7	21.0				
Surface Corrosion (estimated)							
All Elements	Total Reactivity Loss Due to Corrosion						0.40 2.30

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element by 1,584 elements, the effective total weight loss in the core due to bore corrosion is calculated to be 33.7 kg, worth 1.90\$, assuming the material loss to be carbon worth 5.62¢ per kg.

As discussed in section III, additional material loss occurs due to surface corrosion of the fuel elements by interstitial hydrogen. The external corrosion was estimated roughly as contributing a reactivity loss of 40¢. Combining this with the 1.9\$ bore corrosion effect in minute run yields a total 2.3\$ reactivity loss. For comparison, LASL has quoted 2.0\$ as an acceptable reactivity loss during operation for the Phoebus-II reactors, and associates this with a weight loss of 24 grams per element, indicating that this goal is compatible with the expected performance of Phoebus fuel elements.

For the purpose of estimating the control span required in the NRX-A6 reactor, an upper limit to the reactivity loss due to corrosion is required. As a fairly conservative number for this purpose, the value 2.8\$ was chosen. Somewhat higher numbers could be derived if pessimistic assumptions are made to cover corrosion losses during preliminary low power testing and the loss of uranium that may occur during the test. For perspective, in the extreme case that the uranium and carbon were assumed to be lost in proportion to their original inventories in the reactor, the calculated reactivity loss would increase from 2.3\$ to 2.8\$.

Development of Control Span Requirement for the NRX-A6 Reactor

In this section the control span is developed necessary to perform the planned 60 minute power run while satisfying the stated 2.0\$ minimum shutdown requirement throughout the test will be developed. In figure 4-2 the excess reactivity and shutdown reactivity requirements are shown schematically as they vary during the test. The upper line represents the excess reactivity at the 180 degree drum position. The dashed lines on either side of the solid line represent the minimum and maximum reactivities at 180° which might result as a combination of the uncertainties in cold critical reactivity, cold-to-hot swing, and predicted operating reactivity loss.

The excess reactivity initially available in the cold clean reactor at 180 degrees, shown here as 3.8\$, must be sufficient so that at any time during the test the minimum reactivity line at 180 degrees is greater than zero. The ± 53 cent uncertainty attached to the initial cold

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Figure 4-2. NRX-A6 Control Requirement

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reactivity estimate is derived from experience in shiming NRX reactors with essentially complete fuel inventory information. The -0.75 cold-to-hot swing previously developed is assigned an uncertainty of ± 30 cents. At the start of full power operation, Point A, the reactivity uncertainty is taken as the square root of the sum of the squares of 53 and 30¢ reactivity, giving a statistical estimate of 60 cents. During operation, loss of carbon by corrosion leads to 2.35 reactivity loss as previously described; an additional reactivity uncertainty of 50¢ is associated with this loss estimate. At the end of the 60 minutes operation, Point B on the graph, the uncertainty is taken to be the square root of the sum of squares of the 53¢, 30¢, and 50¢ uncertainties enumerated above, leading to ± 80 ¢ total uncertainty. The net reactivity, even with the minimum 80¢ uncertainty figure is above or equal to zero during the latter part of the test. Thus, the test would not be terminated because of reactivity loss, although the operating drum position in this minimum reactivity situation would have reached 180 degrees. Although admittedly somewhat undesirable, it is entirely feasible to orifice the core so that operation at 180 degrees does not produce undue distortion of the radial temperature profile.

It will be noted that the steady state minimum reactivity of zero implies that the excess reactivity available for maneuvering power upward must be obtained from some other source than the control drums. The NRX-A2 and NRX-A3 tests have given ample evidence that power maneuvering can be accomplished by variation in propellant flow, without stability problems. Sustained operation in a flow-rich mode would produce a prohibitive reduction in exit coolant temperature and specific impulse, and is therefore, not contemplated in establishing the minimum excess reactivity requirement.

Even at the point of highest reactivity in the test, in this instance the cold, clean condition, the control drums must be able to shut the reactor down by a calculated 2.05 shutdown margin. Since this margin must be available even if the initial cold reactivity is higher than the predicted reactivity, the 53¢ uncertainty in cold reactivity has been added to 2.05 to derive a total shutdown requirement of 2.55.

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Adding the excess and shutdown reactivity intervals of 3.8\$ and 2.5\$, respectively, we obtain 6.3\$ as a minimum control reactivity compatible with the NRX-A6 power test. Some further observations may be made regarding control. In present NRX-A reactor tests, it has been customary to allow as much as 1\$ reactivity addition due to hydrogen in cold flow tests, while retaining the 2\$ shutdown margin. It is, therefore, desirable that the control drums be capable of providing 3.5\$ shutdown in the cold reactor, so that 1\$ hydrogen feedback can be allowed without terminating the flow test due to drum movement inward to the angle corresponding to 2\$ shutdown. It could be argued that since the flow test is conducted after initial criticality, only 0.5\$ must be added, yielding a shutdown requirement of 3.0\$. In this case there would exist a possibility that the LH₂ flow test would be terminated at, say 10 lb/sec flow. The above considerations indicate that a control reactivity span of about 7.0\$ is a conservative minimum for the NRX-A6 test.

If the shim content of the reactor were established on the basis of a prior "proof-test" experimental criticality measurement on the actual test reactor, as originally contemplated for the NRX-A2 reactor, the 53 cent uncertainty would be eliminated. In other words the initial cold reactivity, and critical position at NRDS would be exactly as specified. The initial excess reactivity in this case would be set at 3.6\$, (combining the 0.3\$ and 0.5\$ uncertainties in operating reactivity changes one obtains 0.6\$ as the uncertainty in reactivity at the end-of-life; the sum of 0.6\$, 2.3\$, and 0.7\$ then yields 3.6\$).

The 0.53\$ cold reactivity uncertainty enters more importantly into the shutdown requirement as shown in figure 4-2. This requirement is reduced by 0.5 from the 3.0-3.5\$ developed above, giving 2.5-3.0\$. Combining the excess and shutdown requirements, one obtains 6.6\$ as a practical control span requirement for the situation in which criticality is established experimentally prior to shimming.

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CONTROL DRUM SPAN ESTIMATE

The NRX-A6 control drum span was developed from 0 to 180 degree EXTERMINATOR eigenvalues. A control drum span predicted for the NRX-A3 design by the same analytical model gave a 9.45 span as compared to its measured span of 8.85. Table 4-10 gives a comparison of the hot eigenvalues and drum spans calculated by the EXTERMINATOR code for the NRX-A3 and NRX-A6 designs. The core inventory and geometry shown in tables 4-1 through 4-3 and figure 4-1 were used in the NRX-A6 calculations.

Normalization of the calculated NRX-A6 drum span of 8.025 by the ratio of the observed to predicted NRX-A3 drum spans gives 7.535. The validity of this normalization is questionable since the geometry of the reflector regions of each design are quite different. This problem will be resolved after the NRX-A6 30 degree mockup experiment described later in this section has been performed. The contributing factors which reduced the normalized NRX-A6 drum span from 8.755 as reported in WANL-TME-1221 to 7.535 are discussed in Appendix A. For the present design the 7.535 span is adopted as a conservatively low estimate. Figure 4-3 presents the reactivity as a function of drum position for an assured 90 degree critical angle.

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Figure 4-3. NRX-A6 INTEGRAL CONTROL DRUM WORTH

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TABLE 4-10. NRX-A3 AND NRX-A6 EXTERMINATOR EIGENVALUES
 AND DRUM SPANS

Drum Position (degrees)	Eigenvalues	
	NRX-A3 Design	NRX-A6 Design
0	0.9793	0.9855
115	1.0205	1.0252
180	1.0458	1.0424
	Drum Spans	
	NRX-A3 Design	NRX-A6 Design
Exterminator Prediction	9.375	8.025
Measured Span	3.805	----
Normalized Span	----	7.535

SELECTION OF AIM SHIM CONTENT

In the preceding sections, the following parameters were derived for the reference design:

Parameter	Value
Ambient reactivity (unshimmed; with 55 gms Nb/element and 98 mil reamed channel diameter.)	3.85 at 90°
Control drum worth	7.53
Cold-to-hot swing (reactivity swing from ambient conditions to 1120 Mw, 71 lb/sec flow).	-0.69
Operating reactivity loss (33 kg bore carbon loss or 21 gm/element)	2.3 ± 0.5

* Calculated drum spans not normalized to experiment.

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In shimming the reactor the aim cold reactivity initially available must be high enough to offset operating reactivity losses, but cannot be so high that less than 2.0\$ shutdown is available at any point in the test series.

Table 4-11 compares the expected range of operating angles for several cold critical positions. A cold reactivity uncertainty of ± 0.53 is utilized, assuming that no prior proof critical experiment has been performed. For each operating angle range an orifice angle is shown which minimizes fuel maximum temperatures.

The top line of Table 4-11 presents several important NRX-A6 reactivity values, for the reference design with an assumed negative shim inventory of 2.4\$, giving 2.4\$ shutdown and 5.2\$ excess reactivity in the pre-run, cold condition. The excess reactivity is high enough so that even in the hot, post-run condition, 2.2\$ excess remains. The initial shutdown in this case, while adequate for the power test, would restrict the amount of feedback reactivity permitted during flow testing.

The middle line represents a compromise in that reactivity has been decreased by 0.6\$ by inserting additional TaC-loaded central shim elements. In this case, as much as 0.5\$ feedback reactivity can be inserted while still maintaining a calculated 2.5\$ nominal shutdown. Even with ± 0.5 uncertainty the 2.0\$ minimum shutdown can be met. The second case also has sufficient excess reactivity that a nominal 1.6\$ remains at the end of power operation.

In the third case, the excess reactivity remaining at the end of the power run is only marginally acceptable, with the stated uncertainty. From this table it appears that an aim cold angle in the range 80 to 90 degrees is appropriate for the NRX-A6 test conditions with the quoted 7.5\$ control span. About 3.0\$ worth of Tac shims would normally be required in this configuration.

NRX-A6 ZONE LOADING AND POWER DISTRIBUTION

Zone Loading

The zone loading in the present NRX-A6 design was derived from a comparison of the fission distributions of the NRX-A6 reference design reported in WANL-TME-1221 and the PAX reactor. This comparison indicated loadings in two of the three outer loading zones in

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TABLE 4-11. SELECTION OF SHIMMED REACTIVITY IN NRX-A6

Critical Drum Positions, degrees				Reactivities ⁽¹⁾ , dollars		
Critical Angle, Cold	Start-of-life Drum Angle, Full Power	Orifice Angle	End-of-life Angle Full Power	Negative Shim Worth	Shutdown Reactivity, Cold	Excess Reactivity Hot, E.O.L.
68^{+8}_{-9}	79^{+9}_{-10}	105	114^{+15}_{-12}	-2.4	$-2.4 \pm .5$	$2.2 \pm .8$
77^{+9}_{-8}	89^{+9}_{-10}	115	125^{+17}_{-13}	-3.0	$-3.0 \pm .5$	$1.6 \pm .8$
90^{+9}_{-8}	101^{+9}_{-9}	130	143^{+37}_{-17}	-3.8	$-3.8 \pm .5$	$0.8 \pm .8$

(1) As a basis for this table, the excess reactivity available at 90 degrees before shimming is taken to be +3.85

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the NRX-A6 design should be reduced by 10 percent and the extreme outer zone by 15 percent in order to have similar power distributions in both designs.* Since this NRX-A6 design is less reactive than the reference design reported in WANL-TME-1221, it was not advisable to lower the fuel loading in the central core region as was previously anticipated. The core weights and atom densities tabulated in tables 4-1 and 4-5 were calculated from the NRX-A3 "as-built" core weights, the fuel loadings in the outer three zones being reduced by 10, 10 and 15 percent, respectively.

Power Distribution

Calculations performed on this new NRX-A6 reference design used the Nuclear Standard Design Method, NSDM-II^{**}. Salient design modifications from this reference design were analyzed by two dimensional EXTERMINATOR problems. Power factors were calculated for each coolant channel by the use of the PFENCE^{**} computer program. Normalization coefficients obtained by comparing a PAX-A2 power prediction and that derived from wire activation data obtained from the PAX reactor were used to correct these power predictions. The NSDM-II method and normalization coefficients used in these calculations have been shown to give good overall agreement for the NRX-A2 power predictions when compared to the measured hot fission distribution, as shown in figure 4-4.

NSDM-II problems were run to obtain nuclear constants for the reactor regions shown in figure 4-1. These nuclear constants were used in two-dimensional EXTERMINATOR problems for the 0, 115, and 180 degree control drum angles to obtain 9 group neutron flux distributions in the reactor. As previously mentioned, the majority of the power calculations were performed at the 115 degree drum angle which has tentatively been selected as the NRX-A6 orifice angle. These flux distributions, in conjunction with fission cross sections, were incorporated into the PFENCE program to obtain a predicted fission distribution in each coolant channel.

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* The "zones" referred to here are the four regions of Figure 4-1. Code 9 fuel occupies the central region

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*WANL-TME-1091, Improved Nuclear Standard Design Method for NERVA Calculations - NSDM-II, J. M. Ravets, L. I. Kopp, A. L. Mowery, and P. J. Sipush

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Figure 4-4. Comparison of the NRX-A2 Measured and Predicted Power Distributions through Traverse to J4M, Fissions Per Gram

The fission per gram distribution generated in these calculations was corrected by the PAX-A2 normalization coefficients; corrections as large as 20 percent were made at the edge of the core. Using this corrected operating fission distribution and the power contribution from gamma and neutron kinetic energy heating, a complete set of channel power factors was calculated for the NRX-A6 reactor at the 115 degree control drum angle, (See table 4-12).

Figures 4-5 and 4-6 show a comparison of these NRX-A6 channel power factors with those calculated by the same method for the NRX-A3 design. This comparison indicates that the major portion of the loading scheme used in these NRX-A6 calculations will probably be very close to that selected for the final design. The channel power factors in the perimeter of the central core region which are above the central values will be lowered by a reduction in loading. Figures 4-7 and 4-8 show a comparison of the operating fissions per gram distributions derived in the same calculations.

Figure 4-9 shows a comparison of the azimuth scalloping in the NRX-A3 and NRX-A6 reactors. As previously noted in WANL-TME-1221, the peak-to-minimum scalloping ratio has been reduced in the NRX-A6 design; for the 115 degree control drum bank angle the peak-to-minimum scalloping ratio was reduced from 1.094 to 1.068.

Figure 4-10 shows the effect of the control drum bank position on the channel power factors in a central and peripheral element for the NRX-A3 and NRX-A6 designs. The NRX-A6 power perturbation per degree of drum movement is much less than in the NRX-A3 so that temperature variation with drum angle is reduced.

WANL EXPERIMENTS PLANNED IN SUPPORT OF NRX-A6

Background

Two separate series of experiments are planned to verify the calculated nuclear design parameters presented in this report. The first, termed the NRX-A6 30 degree Sector Experiment, consists of a mockup of only one 30 degree NRX-A6 reflector sector, and is scheduled for October. This timing allows an initial measurement of NRX-A6 drum worth and reactivity prior to fuel specification.

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NAX-A6 POWER FACTORS CALCULATED BY NSOM II											
115 DEGREE DRUMS											
99	10	0F	236	1178	1178	1167	1167	1167	1156	1156	000
99	99	001	1156	1156	1145	1145	1145	1134	1134	1134	000
99	1A	1F	236	1148	1138	1128	1156	1147	1137	1165	001
99	99	001	1136	1126	1164	1154	1145	1135	1162	1153	002
99	1A	1G	236	1133	1133	1134	1142	1143	1143	1150	003
99	99	001	1153	1153	1160	1161	1162	1162	1169	1170	004
99	1A	1B	236	1129	1141	1153	1128	1140	1152	1163	005
99	99	001	1162	1174	1137	1149	1161	1172	1147	1159	006
99	1A	1C	236	1143	1155	1166	1131	1143	1155	1166	007
99	99	001	1155	1166	1119	1131	1143	1155	1119	1131	008
99	1A	1D	236	1158	1160	1162	1146	1148	1149	1151	009
99	99	001	1139	1140	1123	1125	1127	1128	1112	1114	010
99	1A	1E	236	1161	1152	1143	1159	1150	1141	1132	011
99	99	001	1129	1120	1145	1136	1127	1118	1134	1125	012
99	1B	1F	236	1121	1113	1104	1128	1120	1111	1103	013
99	99	001	1110	1101	1133	1125	1116	1108	1132	1123	014
99	1B	1G	236	1111	1112	1114	1118	1120	1121	1123	015
99	99	001	1130	1132	1134	1136	1138	1139	1143	1145	016
99	1B	1B	236	1107	1119	1131	1104	1117	1129	1141	017
99	99	001	1138	1150	1111	1123	1136	1148	1121	1133	018
99	1B	1C	236	1114	1126	1139	1101	1114	1126	1139	019
99	99	001	1125	1138	1088	1100	1113	1125	1087	1100	020
99	1B	1D	236	1125	1127	1130	1112	1114	1117	1119	021
99	99	001	1106	1103	1088	1091	1093	1095	1078	1080	022
99	1B	1E	236	1128	1120	1112	1125	1117	1109	1101	023
99	99	001	1098	1090	1111	1103	1095	1087	1099	1091	024
99	1B	2F	236	1126	1116	1106	1133	1123	1113	1103	025
99	99	001	1111	1101	1138	1128	1118	1108	1135	1125	026
99	1B	2G	236	1117	1117	1117	1125	1125	1125	1133	027
99	99	001	1133	1133	1141	1141	1141	1141	1148	1148	028
99	1B	2B	236	1120	1131	1141	1118	1129	1139	1150	029
99	99	001	1148	1158	1124	1135	1146	1156	1132	1143	030
99	1B	2C	236	1132	1143	1154	1119	1130	1142	1153	031
99	99	001	1140	1151	1105	1116	1127	1138	1103	1114	032

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18	2D	236	1141	1142	1143	1128	1129	1130	1131	1115	1116	1117	033
99	001	1118	1119	1103	1104	1105	1106	1091	1092	1093			034
18	2E	236	1138	1129	1119	1135	1125	1116	1106	1132	1123	1113	035
99	001	1104	1094	1119	1110	1101	1091	1087	1098	1088			036
1C	1F	236	1083	1075	1068	1059	1081	1074	1066	1094	1087	1080	037
99	001	1072	1064	1093	1085	1078	1070	1091	1084	1076			038
1C	1G	236	1075	1078	1080	1082	1084	1086	1088	1088	1090	1097	039
99	001	1095	1097	1096	1099	1101	1103	1105	1107	1109			040
1C	1H	236	1071	1084	1096	1098	1080	1093	1105	1065	1077	1089	041
99	001	1101	1114	1073	1086	1098	1110	1082	1094	1106			042
1C	1C	236	1075	1088	1100	1062	1074	1087	1099	1048	1061	1073	043
99	001	1085	1098	1048	1060	1073	1085	1047	1059	1072			044
1C	1D	236	1082	1085	1088	1069	1072	1074	1077	1055	1058	1061	045
99	001	1064	1066	1045	1048	1051	1053	1035	1037	1040			046
1C	1E	236	1086	1079	1072	1062	1075	1068	1061	1078	1071	1064	047
99	001	1057	1050	1067	1060	1053	1046	1056	1049	1042			048
1C	2F	236	1094	1085	1076	1100	1092	1083	1074	1106	1098	1089	049
99	001	1080	1071	1104	1095	1086	1078	1101	1093	1084			050
1C	2G	236	1088	1089	1090	1095	1096	1097	1098	1102	1103	1104	051
99	001	1105	1105	1109	1111	1111	1112	1117	1118	1119			052
1C	2H	236	1090	1101	1112	1087	1098	1109	1120	1083	1095	1106	053
99	001	1117	1128	1091	1103	1114	1125	1099	1111	1122			054
1C	2C	236	1097	1109	1120	1084	1095	1107	1118	1070	1082	1094	055
99	001	1105	1116	1069	1090	1092	1103	1067	1078	1090			056
1C	2D	236	1103	1105	1106	1090	1091	1093	1054	1076	1078	1080	057
99	001	1081	1083	1064	1066	1068	1070	1053	1055	1058			058
1C	2E	236	1102	1093	1085	1098	1090	1081	1073	1094	1085	1078	059
99	001	1069	1061	1082	1074	1065	1057	1070	1062	1053			060
1C	3F	236	1090	1080	1070	1097	1087	1077	1067	1104	1094	1084	061
99	001	1074	1064	1100	1091	1081	1070	1097	1087	1077			062
1C	3G	236	1086	1085	1085	1093	1092	1092	1091	1100	1100	1099	063
99	001	1099	1098	1107	1106	1106	1105	1113	1113	1112			064
1C	3H	236	1093	1103	1113	1090	1100	1110	1120	1087	1097	1107	065
99	001	1117	1127	1094	1104	1114	1124	1101	1111	1121			066
1C	3C	236	1104	1115	1125	1091	1102	1112	1122	1079	1089	1099	067
99	001	1109	1115	1076	1086	1096	1107	1073	1083	1094			068
1C	3D	236	1109	1110	1096	1096	1097	1097	1097	1083	1083	1084	069
99	001	1084	1084	1070	1071	1071	1071	1072	1058	1058	1059		070

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10	4F	236	1046	1036	1026	1052	1042	1032	1022	1058	1048	1038	109
99	99	001	1028	1018	1054	1044	1034	1024	1050	1040	1030		110
99	99	001	1045	1044	1043	1051	1051	1050	1049	1058	1057	1056	111
99	99	001	1055	1055	1063	1062	1061	1069	1069	1068	1067		112
99	99	001	1055	1064	1074	1052	1061	1070	1079	1049	1058	1067	113
99	99	001	1076	1085	1055	1064	1073	1082	1061	1070	1079		114
99	99	001	1069	1075	1085	1053	1063	1072	1081	1040	1050	1059	115
99	99	001	1069	1078	1037	1046	1056	1065	1033	1043	1052		116
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	117
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	118
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	119
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	120
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	121
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	122
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	123
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	124
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	125
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	126
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	127
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	128
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	129
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	130
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	131
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	132
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	133
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	134
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	135
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	136
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	137
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	138
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	139
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	140
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	141
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	142
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	143
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	144
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	145
99	99	001	1041	1041	1028	1028	1028	1028	1015	1041	1041	1041	146

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99	1E 3C 236 1013 1014 1015 1016 1017 1018 1019 1020 1021 1022 1023 1024 1025	147
99	99 001 1026 1027 1028 1029 1030 1031 1032 1033 1034 1035 1036 1037 1038	148
99	99 001 1039 1040 1041 1042 1043 1044 1045 1046 1047 1048 1049 1050	149
99	99 001 1051 1052 1053 1054 1055 1056 1057 1058 1059 1060 1061 1062	150
99	99 001 1063 1064 1065 1066 1067 1068 1069 1070 1071 1072 1073 1074	151
99	99 001 1075 1076 1077 1078 1079 1080 1081 1082 1083 1084 1085 1086	152
99	99 001 1087 1088 1089 1090 1091 1092 1093 1094 1095 1096 1097 1098	153
99	99 001 1099 1100 1101 1102 1103 1104 1105 1106 1107 1108 1109 1110	154
99	99 001 1111 1112 1113 1114 1115 1116 1117 1118 1119 1120 1121 1122	155
99	99 001 1123 1124 1125 1126 1127 1128 1129 1130 1131 1132 1133 1134	156
99	99 001 1135 1136 1137 1138 1139 1140 1141 1142 1143 1144 1145 1146	157
99	99 001 1147 1148 1149 1150 1151 1152 1153 1154 1155 1156 1157 1158	158
99	99 001 1159 1160 1161 1162 1163 1164 1165 1166 1167 1168 1169 1170	159
99	99 001 1171 1172 1173 1174 1175 1176 1177 1178 1179 1180 1181 1182	160
99	99 001 1183 1184 1185 1186 1187 1188 1189 1190 1191 1192 1193 1194	161
99	99 001 1195 1196 1197 1198 1199 1200 1201 1202 1203 1204 1205 1206	162
99	99 001 1207 1208 1209 1210 1211 1212 1213 1214 1215 1216 1217 1218	163
99	99 001 1219 1220 1221 1222 1223 1224 1225 1226 1227 1228 1229 1230	164
99	99 001 1231 1232 1233 1234 1235 1236 1237 1238 1239 1240 1241 1242	165
99	99 001 1243 1244 1245 1246 1247 1248 1249 1250 1251 1252 1253 1254	166
99	99 001 1255 1256 1257 1258 1259 1260 1261 1262 1263 1264 1265 1266	167
99	99 001 1267 1268 1269 1270 1271 1272 1273 1274 1275 1276 1277 1278	168
99	99 001 1279 1280 1281 1282 1283 1284 1285 1286 1287 1288 1289 1290	169
99	99 001 1291 1292 1293 1294 1295 1296 1297 1298 1299 1300 1301 1302	170
99	99 001 1303 1304 1305 1306 1307 1308 1309 1310 1311 1312 1313 1314	171
99	99 001 1315 1316 1317 1318 1319 1320 1321 1322 1323 1324 1325 1326	172
99	99 001 1327 1328 1329 1330 1331 1332 1333 1334 1335 1336 1337 1338	173
99	99 001 1339 1340 1341 1342 1343 1344 1345 1346 1347 1348 1349 1350	174
99	99 001 1351 1352 1353 1354 1355 1356 1357 1358 1359 1360 1361 1362	175
99	99 001 1363 1364 1365 1366 1367 1368 1369 1370 1371 1372 1373 1374	176
99	99 001 1375 1376 1377 1378 1379 1380 1381 1382 1383 1384 1385 1386	177
99	99 001 1387 1388 1389 1390 1391 1392 1393 1394 1395 1396 1397 1398	178
99	99 001 1399 1400 1401 1402 1403 1404 1405 1406 1407 1408 1409 1410	179
99	99 001 1411 1412 1413 1414 1415 1416 1417 1418 1419 1420 1421 1422	180
99	99 001 1423 1424 1425 1426 1427 1428 1429 1430 1431 1432 1433 1434	181
99	99 001 1435 1436 1437 1438 1439 1440 1441 1442 1443 1444 1445 1446	182
99	99 001 1447 1448 1449 1450 1451 1452 1453 1454 1455 1456 1457 1458	183
99	99 001 1459 1460 1461 1462 1463 1464 1465 1466 1467 1468 1469 1470	184

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1F 4C 236	979	989	999	966	976	986	996	953	963	973	223
1F 4D 236	983	992	950	960	970	979	946	956	966	976	224
1F 4E 236	977	978	979	964	965	966	967	953	952	953	225
1F 4F 236	951	956	965	958	951	944	932	933	931	931	226
1F 4G 236	971	964	956	978	964	952	944	967	949	978	227
1F 4H 236	964	951	968	941	940	932	945	934	941	941	228
1F 4I 236	959	951	942	964	955	947	938	969	960	951	229
1F 4J 236	945	939	993	985	958	942	965	956	938	938	230
1F 4K 236	962	961	967	967	967	966	966	972	972	971	231
1F 4L 236	971	971	977	977	976	976	980	980	980	980	232
1F 4M 236	972	981	990	969	978	986	995	964	973	982	233
1F 4N 236	991	988	997	966	975	984	993	954	963	971	234
1F 4O 236	980	989	950	959	968	977	946	954	963	971	235
1F 4P 236	976	976	976	963	964	964	964	956	953	951	236
1F 4Q 236	952	942	949	968	966	921	931	931	931	931	237
1F 4R 236	979	970	961	967	959	951	942	981	972	964	238
1F 4S 236	953	946	956	947	939	930	944	935	927	958	239
1F 4T 236	945	955	942	951	946	940	935	970	964	958	240
1F 4U 236	931	918	950	964	935	940	961	949	938	942	241
1F 4V 236	945	944	943	950	949	948	947	956	955	953	242
1F 4W 236	952	953	961	968	975	983	965	967	966	966	243
1F 4X 236	958	966	974	955	963	971	978	952	960	967	244
1F 4Y 236	975	982	955	963	971	978	960	968	975	975	245
1F 4Z 236	969	977	985	956	964	972	980	943	951	959	246
1F 5A 236	989	976	940	948	956	963	935	943	952	957	247
1F 5B 236	989	992	974	958	957	956	949	960	958	957	248
1F 5C 236	943	945	942	943	945	947	921	919	917	917	249
1F 5D 236	969	963	957	955	943	960	943	964	957	951	250
1F 5E 236	934	923	945	937	932	927	935	934	927	932	251
1F 5F 236	1072	1048	1025	1101	1066	1042	1029	1131	1095	1060	252
1F 5G 236	1046	1033	1133	1096	1071	1056	1135	1107	1081	1053	253
1F 5H 236	990	984	978	1031	990	984	977	1049	1031	991	254
1F 5I 236	983	975	1063	1035	1013	990	1077	1044	1020	1016	255
1F 5J 236	1016	989	977	1021	1020	993	990	1037	1026	1016	256
1F 5K 236	1006	1002	1060	1050	1038	1042	1085	1072	1060	1057	257
1F 5L 236	1095	1079	1063	1150	1111	1095	1081	1235	1166	1127	258
1F 5M 236	1113	1100	1248	1181	1152	1137	1260	1205	1176	1166	259
1F 5N 236	1113	1100	1248	1181	1152	1137	1260	1205	1176	1166	260

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32	1J 3K 067 830	855	878	899	910	932	968	964	976	989	413
32	32 001 1027 1065	0	0	0	0	0	0	0	0	0	414
39	1J 3N 067 853	817	781	922	875	839	804	995	944	895	415
39	39 001 862	827	1044	991	935	901	1094	1035	977	904	416
49	1J 3V 094 868	828	788	929	886	846	820	981	947	904	417
49	49 001 878	852	1023	980	940	915	1054	1017	980	926	418
69	1J 3W 128 883	861	840	931	904	882	861	1022	974	926	419
69	69 001 905	884	1060	1011	960	938	1097	1046	996	931	420
69	1J 3R 128 879	859	840	934	904	884	861	992	961	931	421
69	69 001 906	882	1041	1009	976	951	1089	1056	1023	908	422
49	1J 3S 094 872	835	795	939	890	853	831	1007	957	908	423
49	49 001 886	864	1032	979	947	925	1055	1022	987	908	424
39	1J 3T 067 824	800	775	904	858	833	805	985	936	889	425
39	39 001 864	835	1036	984	935	909	1086	1035	984	889	426
39	1J 3U 067 835	804	774	904	863	832	758	976	931	888	427
39	39 001 856	822	1032	985	932	899	1089	1034	979	929	428
59	1J 4F 111 874	845	815	945	901	871	849	1016	972	929	429
59	59 001 907	884	1049	1006	969	946	1083	1046	1009	944	430
79	1J 4G 156 941	922	903	1008	969	952	939	1072	1032	993	431
79	79 001 983	973	1106	1066	1025	1014	1142	1100	1058	959	432
79	1J 4B 156 906	889	872	951	933	917	900	985	972	959	433
79	79 001 944	927	1039	1024	1002	989	1095	1070	1046	944	434
69	1J 4C 128 888	870	853	940	515	898	895	995	969	944	435
69	69 001 940	937	1034	1009	999	995	1074	1065	1056	944	436
49	1J 4D 094 860	853	846	919	901	894	894	981	960	941	437
49	49 001 942	942	1032	1013	996	994	1084	1066	1049	941	438
39	1J 4H 067 840	827	814	919	883	872	859	1006	967	930	439
39	39 001 915	900	1068	1030	993	976	1135	1096	1057	940	440
32	1J 4J 067 913	896	880	994	966	948	930	1081	1052	1024	441
32	32 001 1004	983	0	0	0	0	0	0	0	0	442
39	1J 4K 067 698	689	679	763	742	731	715	827	806	786	443
39	39 001 769	753	878	857	833	816	926	903	881	861	444
39	1J 4M 067 825	793	763	900	859	830	799	980	936	893	445
39	39 001 864	837	1038	992	946	916	1098	1050	1003	989	446
69	1J 5F 128 934	908	882	1003	961	934	920	1076	1032	989	447
69	69 001 974	960	1109	1066	1034	1019	1143	1111	1080	941	448
79	1J 5G 156 879	864	869	938	910	895	883	994	968	941	449
79	79 001 929	917	1030	1004	972	962	1068	1035	1001	941	450

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69	1J 9E 128	891	910	928	921	936	955	984	951	967	982	527
69	69 001 1C12	1043	1013	1029	1059	1089	1075	1106	1136			528
59	1J 9V 111	891	860	791	943	893	842	808	995	945	895	529
59	59 001 860	826	1008	557	917	881	1021	979	938			530
69	1J 9W 128	837	805	772	889	846	818	786	944	899	855	531
69	69 001 829	799	799	957	911	864	837	570	921	873		532
69	1J 9K 120	854	843	822	906	876	854	829	950	918	888	533
69	69 001 860	835	835	989	956	918	890	1027	987	948		534
49	1J 9S 094	846	804	762	859	857	815	785	993	910	868	535
49	49 001 837	808	574	931	893	862	994	956	917			536
39	1J 9T 067	775	740	707	846	795	760	727	919	867	816	537
39	39 001 781	746	746	951	900	848	813	983	933	879		538
39	1J 9U 067	741	710	679	803	753	722	687	863	815	768	539
39	39 001 730	695	695	885	837	783	745	908	852	798		540

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Figure 4-5. Comparison of the NRX-A3 and NRX-A6 Predicted Power Distributions through Traverse 00B to J4M, Fissions/cc

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Figure 4-6. Comparison of the NRX-A3 and NRX-A6 Predicted Power Distributions through Traverse 00B to J5L, Fissions/cc

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Figure 4-7. Comparison of the NRX-A3 and NRX-A6 Predicted Power Distributions through Traverse 00B to J4M, Fissions Per Gram

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Figure 4-8. Comparison of the NRX-A3 and NRX-A6 Predicted Power Distributions through
 00B to J5L, Fission Per Gram

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Figure 4-9. Comparison of the NRX-A3 and NRX-A6 Power Scalloping at the Periphery of the Cone, Extremator Predictions

Figure 4-10. Comparison of NRX-A3 and NRX-A6 Coolant Channel Ten Power Factors vs Drum Bank Position

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The second experiment, termed the NRX-A6 PAX (Pre-assembly Experiment) is to be performed in June or July of 1966. This test will include measurements of drum worth, reactivity, and power distribution in a reflector geometry very closely simulating the NRX-A6 reactor. The simulation is to be more nearly exact in one quadrant of the reactor, and for this reason the measurement has been termed a "90 degree NRX-A6 PAX Experiment", although as may be seen, the reflector will be modified throughout the remaining 270 degrees to simulate the NRX-A6 reflector insofar as the mass of beryllium and graphite are concerned.

The following sections will amplify the objectives and details of each experiment. The present schedule for these experiments is shown in figure 4-11.

NRX-A6 30 Degree Sector Experiment

The purpose of this experiment is to provide an initial definition of the changes in reactivity, drum span, and edge fission distribution produced by the reflector and lateral support modification in NRX-A6. The analysis of these measurements must be complete in time to affect the NRX-A6 fuel manufacture scheduled to begin in December 1965 or January 1966.

This experiment will use two experimental tools, the so-called "permanent PAX graphite barrel" and a mockup NRX-A6 reflector section. The permanent PAX barrel differs from the NRX-A5 barrel chiefly in the fact that a 30 degree sector is removable, as shown in figure 4-12. This barrel was expressly designed to allow investigations of modified reflector geometries and is already in house. This barrel will be utilized in the rebuild of PAX in mid-October.

The second experimental tool is the modified NRX reflector sector shown in figure 4-13. In order to simulate the additional beryllium inside the control drum in NRX-A6, half-inch beryllium rods have been added. Since the mockup sector was released before final definition of the NRX-A6 reflector geometry, the number of the half-inch rods may be varied to establish parametrically the effect of the additional beryllium mass.

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Figure 4-11. NRX-A6 Nuclear Experimental Schedule

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Figure 4-12. Permanent PAX Reactor

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Figure 4-13. Sector 30° Mockup Assembly, NRX-A6 PAX (From WANL Dwg. No. 710J701)

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The control drum to be used in the mockup is 4.01 inch diameter as in the present NRX-A6 design, but will not contain the 0.10 inch deep slot machined in the NRX-A5 and NRX-A6 drums to minimize the temperature gradient across the drum during startup. This cutout extends through a 110 degree arc on the side of the drum opposite the vane, removing 0.6 kg beryllium per drum. Modification of the available PAX drum to this configuration would delay the PAX experiment, which must precede fuel specification. The difference in drum worth is estimated to be about one percent. The effect of this small difference may be corrected analytically with small error.

The PAX fuel configuration will be unchanged, except for the replacement of the fueled partial elements with unfueled graphite elements, which will be made in support of the NRX-A5. This configuration simulates the NRX-A6 except that the outer rows of elements in NRX-A6 may be 10-15 percent lighter in uranium loading than the PAX fuel.

Full NRX-A6 PAX Experiment

The purpose of this experiment is to define the reactivity, drum worth and power accurately for the NRX-A6 reactor. The reactivity estimate will be used to shim the NRX-A6 reactor, and the power measurement will be used as a basis for orificing the NRX-A6 fuel elements. The completion of this experiment is therefore timed to allow specification of shimming and orifice size by the time the first fuel elements reach Reactor Assembly. For this experiment, the "permanent PAX barrel" described above will be removed.

In this experiment the permanent PAX reactor will be modified to produce an exact nuclear simulation of the NRX-A6 core and reflector in one quadrant, and a good simulation of the NRX-A6 beryllium reflector as far as mass is concerned throughout the remaining quadrants. In the mockup quadrant, three NRX spare sectors will be modified as shown in figure 4-14a to simulate NRX-A6. Throughout 270 degrees, the reflector consists of 9 KIWI sectors augmented on the inner diameter with additional beryllium liners, as shown in figure 4-14b.

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A. MODIFIED NERVA SECTOR - THREE REQUIRED FOR 90° QUADRANT OF NRX-A6 PAX

B. MODIFIED KIWI SECTOR - NINE REQUIRED FOR 270° PORTION OF NRX-A6 PAX

Figure 4-14. Sector Design for NRX-A6 PAX Experiment

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Throughout the 90 degree sector, the fuel will be changed to mock up the NRX-A6 configuration. As noted earlier, however, the present PAX loading of 415 mg uranium per cc of fueled graphite in the central fuel zone is essentially the same as that developed herein. Thus, the PAX core will actually simulate NRX-A6 in all sectors, except for the outer elements. In the 90 degree sector these elements will be replaced by elements made to the NRX-A6 fuel specification.

Although the reflector geometry is described as "exact" in the 90 degree region, it will be noted that the actual NRX-A6 outer reflector rings have been replaced by staves, and the actual seal, retainer, and load segments are simulated by a cylindrical shell of graphite about 2.4 inches thick. These changes have simplified the design at negligible loss of nuclear equivalence.

Since the KIWI sectors have adjustable beryllium shims, the effect of varying beryllium mass on reactivity and drum span may be studied experimentally. If small changes to the NRX-A6 reflector design are made, this flexibility will be used to interpolate to the actual beryllium mass. Similarly, a reduction in graphite content of the seals can be handled by drilling holes in the graphite shells if the NRX-A6 seal, retainer, and load segments are reduced in size.

NRX-A6 - NUCLEAR DESIGN FLEXIBILITY

Increased NbC Coating Weights

The niobium mass utilized in the reactivity calculations is equivalent in reactivity to the actual distribution of about 55 grams niobium per fuel element. A physical weight of 55 grams niobium as actually distributed along the fuel length is taken to be as effective as 60 grams niobium distributed uniformly. Thus the assumed axial distribution is more effective by about 9 percent. A recent estimate by the Materials Department as to the coating profile to be used in NRX-A fuel elements was found to be about 7 percent more effective than a uniform distribution. Using this weighting function, the uniform mass of niobium utilized in the calculations may be reinterpreted as being equivalent to about 56 grams of niobium in the expected NRX-A6 profile; the difference in weighting function results in about only 11 cents change in reactivity.

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In Section 5 it is shown that an 80° to 90° cold angle is appropriate for the NRX-A6 test conditions, requiring that 2.4 to -3.0\$ excess reactivity be available in the unshimmed reactor at the drum angles. This reactivity, which will have been verified prior to the time that niobium carbide coating of NRX-A6 elements is initiated, would allow about 55 kg additional niobium to be added to the core, or 35 grams per fuel element. Not all of the available excess reactivity can safely be used as a basis for increasing niobium coating mass, however. A part of this reserve must be retained as a margin against any future changes which may be required in the NRX-A6 design, and against error in the calculated reactivity. Each of the planned PAX experiments should reduce the uncertainty in this reactivity estimate; it is likely that prior to coating sufficient evidence will exist that an increase of, say 10 grams per element can be safely made.

One-dimensional studies have shown that the fuel niobium content has small effect on control drum worth. This has not as yet been verified by experiment or by the more realistic 2-dimensional calculations.

Use of High Void Fraction Fuel in NRX-A6

Although considerable numbers of high void fraction fuel elements can be incorporated in the NRX-A6 without serious reactivity loss, an overall increase in coolant channel diameter in the NRX-A6 design beyond 104 mils would require an increase in uranium loading. For example, if the void fraction is increased from 29 percent to 35 percent, corresponding to an increase in coolant channel diameter from 98 mils to 107 mils, a reactivity loss of about 6\$ ensues, if the uranium loading per unit of volume of fuel material remains constant. Most of the reactivity loss is produced by the reduction in both uranium and carbon content in the ratio 65:71, namely, the ratio of fuel volumes in the two cases. Since the coolant channel surface increases, the niobium carbide content of the core is increased roughly in the proportion of 107 to 98, contributing a further 0.50\$ reactivity loss. Figure 4-15 shows the available excess reactivity in the NRX-A6 at the 90 degree drum position as a function of niobium mass per fuel element and also channel diameter.

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If the fuel distribution in the reference design is raised uniformly in the proportion 500 to 417, so that the central loading becomes 500 milligrams per cc, the excess reactivity available at 90 degrees is 2.0\$ at 35 percent void fraction (107 mils channel diameter) or zero at about 37 percent void fraction (110 mils channel diameter). The corresponding excess at 98 mils channel diameter is 8.4\$. It also follows that a core composed of equal numbers of the 98 and 115 mil-reamed channel diameter elements would have an excess reactivity intermediate between the 8.4\$ and -4.0\$ values, or 2.2\$.

The use of unit uranium loadings higher than 417 mg/cc to offset increased void fraction should not cause a loss of drum span. Figure 4.16 shows that the drum span is essentially unchanged if the excess reactivity is maintained constant by adjusting uranium loading upward as void fraction increases. The two effects, increasing uranium loading and increasing void fraction tend to cancel, the one tending to reduce leakage and hence drum worth, the other acting to increase neutron leakage and drum worth.

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Figure 4-15. Effects of Channel Diameter on Available Excess Reactivity at 90° Drum Position

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Figure 4-16. Effect of Uranium Fuel Loading and Void Fraction on Drum Span

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APPENDIX A

NRX-A6 REACTIVITY AND DRUM SPAN COMPARISON

NRX-A6 REACTIVITY COMPARISONS

The Reference NRX-A6 nuclear design described at the previous review meeting* required 150 kg uranium to provide a calculated 4.5\$ excess reactivity at the 90 degree drum position, and with 74 kg Nb (about 43 gm per fuel element). With the increase in bore Nb mass expected in NRX-A6 to about 55 gm per element, and with external coating of peripheral elements (cf. Table 4-1), the excess reactivity is expected to be reduced by 1.6\$.

The present NRX-A6 configuration has a slightly smaller core, 1584 elements versus the 1656 assumed previously, and with the present reflector sector and control drum dimensions, contains somewhat less beryllium.

As a result of these differences the excess available in the present NRX-A6 design, but with the WANL-TME-1221 fuel loadings and Nb content, is now estimated to be about 2.6\$ at 90 degrees. Using the Nb masses of Table 4-1, the reactivity would be about 1.0\$.

Although this reactivity is believed adequate for the NRX-A6 test as presently defined, the objectives of the NRX-A6 and follow-on reactors imply that greater reactivity is desirable to allow for possible design changes for increased use of niobium, or wider use of high void fraction fuel. About 2.5\$ excess reactivity has therefore been added in the present design by increasing the uranium loading to the value used in the present NRX-A2 through NRX-A5 reactors, requiring about 0.42 gm uranium per cc of fueled graphite elements in the central fuel zone.

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*WANL-TME-1221, NRX-A6 Design Modifications: Table III-5, p.3-11.

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As shown in section 4, an entire core of fuel with a void fraction of about 33 percent is within the capability of the NRX-A6 design with present 0.42 gm/cc loading. Alternatively, clusters of higher void fractions, i.e. - up to 40 percent may be alternated with 29 percent void fraction elements in the approach to be utilized in the Phoebus 1 reactor.

In the NRX-A6 design, it is also feasible by raising the uranium density in central elements to 0.50 gm/cc to test an entire core of fuel elements with reamed channel diameters of at least 110 mils diameter (37 percent void fraction), with no other changes to the reactor. This is the current planned void fraction for the Phoebus 2 reactor.

An important advantage derived from the change from the WANL-TME-1221 loadings to those specified herein is that the existing PAX core may be considered to be a mockup of the reference NRX-A6 core. The calculated excess reactivity and drum span presented in this document may therefore be compared directly with the measurements to be made in the NRX-A6 30 degree sector experiment to be performed prior to fuel specification.

NRX-A6 DRUM SPAN COMPARISONS

Effect on Drum Span of an Additional Coolant Slot on the Control Drum

A coolant slot similar to that which is required for the boral plates is now machined on the opposite side of the drum to relieve thermal bowing during reflector cooldown. The loss in drum span was evaluated by perturbing the mass of beryllium in region 13 of figure 4-1 and noting the change in drum span predicted by EXTERMINATOR problems at 0 and 180 degrees. The resulting drum span coefficient of +1.34 cents per kilogram of beryllium gave approximately an 8¢ loss in drum span.

Effect on Drum Span of an Additional Loss of 15 kgs. of Beryllium in the Reflector

The distribution of an additional 15 kgs. of beryllium lost in the reflector from the reference design reported in WANL-TME-1221 is important in evaluating its net effect on drum span. Since the coolant channels and spring holes are slightly larger in the current NRX-A6 design and their number and position have changed, it is extremely difficult to assess the exact change in drum span which can be attributed to these design modifications. Using the drum span coefficient of +1.34¢/kg of beryllium, which was developed for the control drum region, gives a net loss in drum span of approximately 20¢.

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The effect on drum span of changes in mass of beryllium in the region near the graphite lateral support system was evaluated by perturbing the beryllium mass in region of figure 4-1 of EXTERMINATOR problems at 0 and 180 degrees. A negative drum span coefficient of $-.68\text{¢/kg}$ was noted versus the above positive 1.34¢/kg for the drum region. Any loss in reactivity due to a reduction in mass of beryllium in this region to increase drum span would be prohibitive due to reactivity considerations.

Effect on Drum Span of the Use of Higher Loaded Central Fueled Elements

The increased niobium concentration in the core due to thicker bore coatings* and the loss of beryllium mass in the reflector from the design reported in WANL-TME-1221 will require the central code 99 fuel elements to have loadings similar to those used in the NRX-A3 design. The effect of this higher loaded central core region has been estimated by drum span coefficients to have reduced the total drum span by approximately 50 to 60 cents.

Effect on Drum Span of a Reduction in the Number of Fueled Elements

In addition to the changes noted above, the number of fueled elements in the NRX-A6 design has been reduced from 1656 to 1584 as in NRX-A4 and A5. An upper limit of this reduction on drum span has been estimated to be approximately 10 cents.

Effect on Drum Span of the Use of Molybdenum Screws

The use of molybdenum screws to fasten the retainer segments to the beryllium reflector is being evaluated by the Mechanical Design Group. The effect of molybdenum on drum span was obtained by running EXTERMINATOR problems at 0 and 180 degrees with and without molybdenum in region 9 of Figure 4-1. This resulted in a negative drum span coefficient of $-.6\text{¢/kg}$. One of the design concepts which shows approximately 1.4 kgs. of molybdenum between the core and reflector would lower the drum span by approximately 8 cents.

Numerical addition of the above drum span losses gives a net change in drum span of approximately 1\$ which is in agreement with the difference between the 8.98\$** calculated

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*The NRX-A6 Reference Design in WANL-TME-1221 used 43 grams Nb per element similar to NRX-A3 and A4 while this NRX-A6 analysis uses 55 grams per element similar to NRX-A5.

**Calculated drum spans not normalized to experiment.

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drum span presented in WANL-TME-1221 and the 8.03\$* drum span calculated for the present NRX-A6 design. Loss of beryllium in the reflector and the increase in uranium loadings were the major contributing factors which reduced drum span.

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*Calculated drum spans not normalized to experiment.

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APPENDIX B

AS BUILT NRX-A3 COMPOSITION

The actual composition of the NRX-A3 reactor is given in table 4-13. This inventory was used as the basis for the NRX-A3/NRX-A6 reactivity comparison previously discussed. These numbers are slightly different from the inventories previously published (WANL-TME-840A, WANL-TME-878) since the earlier tabulations were extrapolated from fuel element data for only part of the total core inventory.

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TABLE B-1. NRX-A3 "AS-BUILT" CORE WEIGHTS

Material	Region 1 & 2	Region 3	Region 4	Region 4	Total
Outer Radius of Cold Region, cm ³ Hot	36.408 36.674	40.279 40.573	41.938 42.244	43.804 43.818	
Carbon Weight, kg	601.516	129.441	52.230	76.710	859.897
Fuel Elements	34.565	2.367	9.943	5.208	52.083
Unfueled 13-Hole Elements Single Hole	70.500	24.736	11.133	4.329	110.698
Pyrographite Sleeves*	17.343	4.173	3.617	1.670	27.370
Positive Shims	5.029	0.606	2.582	1.336	9.553
Total	728.953	161.323	79.505	89.253	1059.601
Uranium Weight kg	139.342	21.346	5.553	5.717	171.958
Niobium Weight kg ²					
Fueled Elements	51.471	10.970	4.221	8.328	73.572
Unfueled Elements	0.010	0.003	0.002	0.001	0.016
Total	51.481	10.973	4.223	8.329	73.588
Negative Shims, Weight kg	12.490	0	0	0	12.490
Stainless Steel, Weight kg	5.116	1.231	1.067	0.492	7.906
Inconel X*-Weight kg	13.178	3.170	2.748	1.269	23.616
Hydrogen Weight kg ¹	0.149	0.033	0.015	0.018	0.215

1. Hydrogen weights correspond to axial midplane densities used in exterminator problems not representative of total hydrogen mass in core.
  2. These are correct weights to be used in radial calculations. The niobium weight has been corrected for the axial distribution of the coating and includes external coating on peripheral elements.
- *NOTE: Inconel-X and pyrographite sleeve weights were adjusted from actual lengths of 60.3 inches and 53.1 inches, respectively; all weights are now for 52 inch fueled core length.