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Westinghouse Astronuclear Laboratory

Test Program For Support
Of NR-1 Reactor Development
(Title Unclassified)

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TEST PROGRAM FOR SUPPORT OF

NR-1 REACTOR DEVELOPMENT

(Title Unclassified)

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by authority of DOC
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SEP 10 1973

~~INFORMATION CATEGORY~~

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Authorized Classifier

6/8/66

~~GROUP I - Excluded From Automatic
Downgrading and Declassification~~

SPECIAL REREVIEW FINAL DETERMINATION	Reviewer	Class.	Date
	IRP	U	4-22-82
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ABSTRACT

This document describes the development and component tests required in support of the NR-1 reactor. The description is directed mainly toward defining the test data requirements for all the tests that can be foreseen at this time. This covers both tests in connection with specific reactor components, as well as those intended to increase the general level of knowledge in such areas as friction, corrosion, thermal contact resistance, material information, etc. An estimated schedule is included with all the tests falling within the period from July 1966 to the presently scheduled reactor test date of July 1969.

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I. INTRODUCTION

At present the NR-1 reactor design features a 5000 MW power level, high-void fuel, counterflow cluster support, beryllium ring combined reflector, a flight-type shield, and a flight-type actuator system with ganged drum drive mechanism. The object of the tests described herein is to complete the development and reactor qualification of these and other features for NR-1.

As a quick reference, Table 1 is a summary listing of the tests with estimated completion dates (contract year and quarter). In some instances, two dates are given. The first indicates completion of preliminary testing for the purpose of guiding the detail design. The second date indicates completion of final or proof testing. These dates are the result of a joint effort on the part of the effected WANL groups (Reactor Engineering, Instrumentation and Control, and Materials Departments). The scheduling used in evaluating information need dates included the building of a cold flow reactor for vibration testing during 1968 and for systems tests prior to engine testing, but did not include a cold flow unit for reactor testing.

The need for some of the tests (such as those in connection with a fueled element thermocouple installation) is dependent on a number of contingencies, and they may be deleted at a later date. On the other hand, the requirements for additional tests will probably arise as the design progresses, and as the results of future NRX-1 reactor tests become available. Table 1 is complete in that it contains all the necessary tests that can be foreseen at this time. However, it will require periodic review to update it and to maintain proper priorities as additional test requirements develop.

TABLE 1
NR-1 TEST SUMMARY

SECTION	MECHANICAL TESTS	FLOW TESTS	MATERIALS TESTS
A. REFLECTOR SYSTEM			
1. SEALS	LEAKAGE WITH SIMULATED LOADING (2nd '68)	NOZZLE-INTERFACE SEAL LEAKAGE (1st '68)	
2. FLOW DISTRI- BUTION AND CONTROL	RING THERMAL DISTORTION (3rd '68)	ORIFICE LOSS COEFFICIENT (1st '68) NARROW CHANNEL DATA (3rd '68) PLUNGER SLEEVE IMPEDANCE (2nd '67)	
3. TRANSIENT MODELS		MULTI-CHANNEL FLOW MODELS (1st '69)	
4. O.D. BAFFLE		LEAKAGE AND THERMAL CONDUCTANCE (4th '67)	304 CRES CRYOGENIC STRESS- STRAIN (1st '67)
5. BERYLLIUM PROPERTIES	LOW-CYCLE FATIGUE (4th '67) STRESS CONCENTRATIONS (2nd '67)		STRESS-STRAIN DATA (TO2000°R, 1st '67 to 2500°R, 2nd '68) CREEP RUPTURE (2nd '68) BRITTLE FRACTURE (2nd '67) RADIATION EFFECTS (2nd '67) (CONTINUOUS)
6. AFT SUPPORT RING	LOAD VERSUS DEFLECTION (1st '68)		
R. CONTROL DRUMS			
1. GANGED DRIVE MECHANISM	FATIGUE - TENSILE PROPERTIES (3rd '67) (3rd '68) VACUUM TESTS (3rd '69)		

NOTE: DATES BASED ON CONTRACT
YEAR AND QUARTER

TABLE 1
NR-1 TEST SUMMARY (CONT'D)

SECTION	MECHANICAL TESTS	FLOW TESTS	MATERIALS TESTS
1. GANGED DRIVE MECHANISM (CONTINUED)	ENDURANCE & FEASIBILITY (3rd '67) (3rd '68) SYSTEMS INTEGRATION (3rd '68)		
2. CONTROL DRUM	THERMAL BOW (3rd '69)		
3. CONTROL SLEEVE	THERMAL BOW (1st '68)	IMPEDANCE (1st '68)	
C. CORE LATERAL SUPPORT			
1. LATERAL SUPPORT SPRINGS	CONDUCTION COOLING (1st '68) RELAXATION (4th '67)	CAVITY PURGE (1st '68)	
2. SEAL SEGMENT DEVELOPMENT	FLEXURAL STRENGTHS (3rd '67) (3rd '68) GRAPHITE POROSITY (3rd '67) SINGLE JOINT SEALS (2nd '67) (2nd '68) PARTIAL LENGTH CORE (3rd '67) SEAL ANGLE STUDIES (3rd '67) (3rd '68) SEAL RETENTION TESTS (3rd '67) SCALE MODEL REACTOR VIBRATION (3rd '67)		GRAPHITE CEMENT AND BRAZE (3rd '67)
3. PLUNGER PINS	FLEXURAL STRENGTH (1st '68)		
4. THERMAL AND MECHANICAL PROPERTIES OF CONTRACTS	THERMAL CONDUCTIVITY (CONTINUOUS) FRICTION COEFFICIENT (CONTINUOUS)		
5. FULL REACTOR (CFDTS)VIBRATION	LOW-LEVEL SINUSOIDAL (2nd '69)		

NOTE: DATES BASED ON CONTRACT YEAR AND QUARTER

TABLE 1
NR-1 TEST SUMMARY (CONT'D)

SECTION	MECHANICAL TESTS	FLOW TESTS	MATERIALS TESTS
D. SHIELD			
1. INSERT MATERIAL THERMAL SHOCK (4th '67) DEVELOPMENT			STRESS-STRAIN DATA - METALLOGRAPHIC INSPECTION - CHEMICAL ANALYSIS (4th '67) RADIATION EFFECTS (4th '68)
2. SHIELD ELEMENT	CRYOGENIC VIBRATION (4th '68)	IMPEDANCE (4th '67)	
3. SHIELD ASSEMBLY	CRYOGENIC VIBRATION (2nd '68) LOAD VERSUS DEFLECTION (2nd '68)		
E. COUNTERFLOW TIE TUBE			
1. TIE TUBE	TENSILE (1st '67) (1st '68) FATIGUE (2nd '67) (1st '68) VIBRATION (1st '68) BURST TEST (1st '68)	PRESSURE DROP (4th '67)	
2. CORROSION		FULL CLUSTER AND SINGLE ELEMENT (2nd '69)	
3. SUPPLY PLENUM	LEAKAGE - THERMAL GRADIENTS (1st '67)		
4. EFFLUENT MIXING		MIXING REQUIREMENTS (4th '67)	
5. SUPPLY PLENUM MANIFOLD	CRYOGENIC VIBRATION AND LEAKAGE (2nd '68)		

NOTE: DATES BASED ON CONTRACT
YEAR AND QUARTER

TABLE 1
NR-1 TEST SUMMARY (CONT'D)

SECTION	MECHANICAL TESTS	FLOW TESTS	MATERIALS TESTS
6. SUPPORT PLATE AND PLUTONIUM ASSEMBLY	LOAD VERSUS DEFLECTION - THERMAL DISTORTION (1st '68)		
F. FUEL ELEMENTS			
1. FUEL DEVELOPMENT	TIP STRENGTH (3rd '67) POROSITY (3rd '68) HIGH TEMPERATURE STRENGTH (2nd '67) CORRODED STRENGTH (3rd '68)-WEIBALL CONSTANTS (2nd '67) - Same WITH PRODUCTION ELEMENTS (2nd '69)	CORROSION TESTS (CONTINUOUS)	MOLY-OVERCOATING - TIP DEVELOPMENT (2nd '67)
2. CORE BUNDLING	ELEMENT FORCES (4th '68) INTERSTITIAL GAP MEASUREMENTS (3rd '68)	NARROW CHANNEL FLOW (3rd '68)	
3. HIGH TEMPERATURE ORIFICES	BRAZE STRENGTH (4th '67) (4th '69)	LOSS AND FILM COEFFICIENTS (2nd '68)	HIGH TEMPERATURE BRAZE (2nd '67)
4. CORE PERIPHERY	RADIAL INFLOW (2nd '67) (1st '69) COATING EFFECTS (4th '67) FILLER STRIP SUPPORT (2nd '68) ENDURANCE TEST (3rd '69)		
G. CLUSTER SUPPORT			
1. SUPPORT BLOCKS	CRACK INITIATION (3rd '68) (2nd '69) THERMAL CYCLING (2nd '68) (2nd '69)	CORROSION (2nd '67)	EUTECTICS AND CARBIDING (4th '67)

NOTE: DATES BASED ON CONTRACT
YEAR AND QUARTER

TABLE 1
NR-1 TEST SUMMARY (CONT'D)

SECTION	MECHANICAL TESTS	FLOW TESTS	MATERIALS TESTS
1. SUPPORT BLOCKS (CONTINUED)	ULTIMATE LOAD (2nd '69) THERMAL GRADIENT (4th '67) (3rd '69) COATING WELDING (2nd '67) (1st '69)		
2. BLOCKLESS CLUSTER	KEY AND TIP STRENGTH - CRACK INITIATION - (4th '68) (3rd '69) CREEP RUPTURE (2nd '68) (2nd '69) ELEMENT BUNDLING (3rd '67)	CORROSION (4th '67)	
3. METALLIC AND CARBIDE COMPONENTS	THERMAL STRESS - LOW-Cycle FATIGUE (4th '67) (3rd '69)		MECHANICAL AND RECRYSTAL- LIZATION PROPERTIES (2nd '67) NEW INSULATING MATERIALS (3rd '68)

NOTE: DATES BASED ON CONTRACT
YEAR AND QUARTER

TABLE 1
NR-1 TEST SUMMARY (CONT'D)

SECTION	MECHANICAL TESTS	FLOW TESTS	INSTRUMENT DEVELOPMENT TESTS
H. INSTRUMENTATION			
1. TEMPERATURE MEASUREMENT			
a. MATERIAL TEMPERATURE	ELEMENT STRENGTH (4th '68) THERMAL GRADIENTS (4th '68)	CORROSION (2nd '69)	QUALIFICATION TO 5000°R (3rd '68)
b. EXIT GAS TEMPERATURE	VIBRATION (4th '68)	CLUSTER CORROSION (2nd '69)	QUALIFICATION TO 5000°R (3rd '68)
2. CALORIMETERS			QUALIFICATION (4th '67)
3. DISPLACEMENT TRANSDUCER	ENDURANCE - LINEARITY - DRIFT (2nd '69)		QUALIFICATION (1st '69)
4. DRUM POSITION INDICATOR	VIBRATION - CRYOGENIC ENDURANCE (2nd '69)		QUALIFICATION (1st '69)
5. STRAIN TRANSDUCERS			
a. BERYLLIUM			QUALIFICATION (1st '69)
b. TIE TUBES	TENSILE - FATIGUE - ENDURANCE (2nd '69)		QUALIFICATION (1st '69)
6. PRESSURE TRANSDUCERS			QUALIFICATION (3rd '68)
7. VIBRATION			QUALIFICATION (1st '69)

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TABLE 1
NR-1 TEST SUMMARY (CONT'D)

SECTION	MECHANICAL TESTS	FLOW TESTS	NUCLEAR TESTS
I. CORROSION		BASIC INFORMATION TESTS (CONTINUOUS)	
J. VACUUM	STRESS-STRAIN RELATIONS - CONDUCTIVITY - WELDING - FRICTION (CONTINUOUS)		
K. REACTOR CRITICAL EXPERIMENTS		H ₂ ABSORPTION IN GRAPHITE (3rd '67) (4th '68)	POWER DISTRIBUTION (2nd '67) (4th '68) DRUM SPAN (2nd '67) (4th '68) REACTIVITY COEFFICIENT (3rd '67) (1st '69) FRC (1st '69) H ₂ WORTH (3rd '67) DOSIMETRY (3rd '67)
L. ACTUATOR DEVELOPMENT			
1. CONTROL DRUM ACTUATOR	BREADBOARD TESTS (4th '67) PROOF TESTS - GANGING TESTS (2nd '68)		
2. TIE TUBE FLOW CONTROL	ACTUATOR TESTS (1st '68) COMBINATION TESTS (2nd '69)	VALVE TESTS (1st '68)	

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II. DEVELOPMENT TESTS

A. REFLECTOR SYSTEM

1. Seals

All seals that differ substantially in either size or configuration from those in NRX-A6 should be leak tested. This includes the following seals:

- a. Nozzle Interface Seal
- b. Beryllium Ring Radial Seal
- c. Reflector Forward-End Seal
- d. Reflector to Pressure Vessel Seal (if any)
- e. Drum Annulus Sleeve Seal
- f. Spring Retention Cap Seal
- g. Plunger Sleeve Seal

The leakage data should be generalized by plotting against the product of the test gas density (ρ) and the test pressure differential (Δp). In the event that the leakage flow is laminar, the product $\rho \Delta p$ should be divided by the test gas viscosity (μ). The range of the data should be sufficient to cover hydrogen with properties and pressure differentials as expected during reactor operation. For Δp values up to 300 psi, the maximum value of $\rho \Delta p$ is $0.75 \text{ lb}^2/\text{in}^5$, and of $\rho \Delta p / \mu$ is $20 \times 10^5 \text{ lb-sec/in}^4$.

This testing should include simulated seal loading and is intended to complement the NRX-A6 component test data. Additional testing may be needed to investigate the effects of seal loading, seal surface defects, vibration, and thermal transients.

The need for such testing would be indicated by excessive leakage rates in the above

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testing. For example, excessive leakage of the plunger sleeve seals would require further thermal transient tests. These tests should be preceded and followed by complete inspection of the sleeve and the seal to uncover any plastic deformation that may be caused by transient thermal stresses.

2. Flow Distribution and Control

- a. To assure the proper flow distribution in the reflector, the impedance characteristics of the channel orifices must be known. Tests should be conducted to determine the orifice loss coefficients over the range of Reynold's numbers (10^6 to 10^8) expected during reactor operation. In addition, choking of the orifices should be investigated by determining the flow characteristics as a function of pressure ratio for ratios down to 0.5. Orifice leakage rates should also be found in a manner similar to the seal leakage rates in Section A-1.
- b. The reflector flow distribution will also be affected by the amount of inter-channel flow that occurs at the beryllium ring interfaces as the rings distort due to thermal gradients. Testing is required to investigate two aspects of this problem as follows:
 - (1) Beryllium Ring Thermal Distortion - Analysis of the shape of the beryllium rings during reactor operation is quite complex. Testing of models similar to the reflector is required to check the analytical tools used for predicting the shape of the reflector. Tests should be conducted on a stacked ring configuration with heat inputs and cooling similar to the reactor. The similarity need not be to scale but only sufficient to test the accuracy of the

analytical tools used to predict the model's distortion. An accurate prediction of the model's distortion will result in confidence in the analysis. The required test hardware is similar to that required in Section 3 below (Transient Flow and Thermal Models). These tests may be combined if satisfactory results can be obtained.

- (2) Using the analysis confirmed in the thermal distortion test (Section A-2b(1)), the characteristics of the flow through small clearances between parallel plates should be determined. As in Section A-1 with seal leakage tests, the flow rates as a function of test gas density, viscosity, and pressure differential should be determined for a range that includes reactor conditions. If the flows are found to be excessive, various methods for reducing the flow may require testing. These methods might include anything from surface roughening to additional sealing devices depending on the test results.
- c. The plunger sleeve design is expected to change the NR-1 if detailed thermal analysis indicates that such a change is needed. If so, the flow impedance of alternate sleeve designs (if any) must be determined experimentally. This is required so that the proper flow distribution can be maintained in the region of the plungers. Loss coefficients should be determined for a range of Reynold's numbers as found in the reactor (10^4 to 10^6). The test results will be used to guide the re-design of the sleeves in order to minimize the impedance.

3. Transient Flow and Thermal Models

The complexity of the transient analysis of the beryllium reflector indicates that confirming tests should be conducted. As in the thermal distortion testing above, this testing is mainly a check on the analytical tools used in this area. Thus, the model requires reactor simulation only to the extent that it be a multiple flow channel unit with both external and internal heat inputs. Flow and temperature in the model will be compared to predicted transient and also steady-state values. Variation in test conditions should be made as required to further develop the analytical techniques.

4. O.D. Baffle

The magnitude of the thermal stresses in the beryllium is partly dependent on the insulating character of the outside baffle. The baffle's insulating character is, in turn, dependent on the leakage flow between the baffle and the beryllium. Thus, the heat conductance through the baffle must be determined under conditions that include this leakage.

The testing should determine the overall conductance of the path through the baffle, the hydrogen, and into the beryllium. This must be preceded by a leakage rate test similar to the seal tests of Section A-1 with leakage determined as a function of normal pressure between the baffle and the beryllium. This can then be followed by thermal conductance tests at various material temperatures and leak rates. (A similar test is presently being conducted as part of the NRX-A6 program.)

Since the baffle will tend to shrink on the beryllium during reactor operation, detailed cryogenic stress-strain data should be obtained for butt-welded 304 stainless steel sheet material.

5. Beryllium Properties (See also Section J, Vacuum Testing)

- a. Stress-Strain Relations - In the past, accurate stress analysis of the reflector has been limited by the lack of stress-strain data on beryllium. Tensile data (stress vs. strain relations) are required for temperatures from 140°R to 2500°R and compressive data from 140°R to room temperature. Creep rupture data should be obtained at the higher temperatures.
- b. Radiation Effects on Strength - The present program includes testing to determine the effect of radiation on the notched and unnotched (both tension and compression) strengths of beryllium from 140°R to room temperatures. This testing also includes an investigation of the annealing temperatures and time requirements. This same data is required for the higher dosages encountered in NR-1 operation. Also, the effect of stress level during irradiation should be determined.*
- c. Low-Cycle Fatigue - The present test program includes low-cycle cryogenic fatigue testing of beryllium. To date, relatively few test specimens have been tested. There is a need for continued testing of the same type to determine the differences between pressings, locations within each pressing, and orientation relative to the direction of pressing for both cryogenic and room temperatures.
- d. Stress Concentrations - There are a number of holes and corner radii in the reflector (i.e., coolant holes, pilot groove, radial seal groove, and outer seal segment grooves). Little data is available on the effect of

*This is part of a more general program aimed at obtaining radiation effects information on other reactor materials (i.e. aluminum, titanium, Inconel, etc.)

stress concentrations in beryllium. More basic data is required. For example, basic brittle fracture properties (G_c or K_{Ic}) of beryllium are needed in order to better understand the material and to permit stress analysis of allowable crack size and crack propagation. Data on these properties are needed for temperatures between room temperature and 140°R.

6. Aft End Support Ring

The analysis of the aft end support ring should be confirmed by static load tests similar to those presently planned for NRX-A6. The tests should be conducted with reactor grade hardware (CFDTS components).

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P. CONTROL DRUMS

1. Ganged Drive Mechanism

Prior to the final design of the ganged drive mechanism, certain basic design information must be available from testing:

- a. Bending Fatigue - A non-reversing cyclic bending stress occurs in the titanium bands at the point of tangency on the drum drive pulley. Fatigue testing of this configuration is required to guide the final design. Sufficient data should be obtained to plot a modified Goodman diagram based on a minimum life requirement of 20,000 cycles at a torque level of 1100 in-lb. This data is not available at present for very thin material (0.005 in to 0.015 in).
- b. Tensile Properties - Basic stress-strain relations should be determined for the thin titanium band material. This information is required for temperatures from 140°R to room temperature to establish the design margin for this design.
- c. Decoupling Mechanism - Endurance tests of the drive mechanism are required for conditions simulating a frozen drum. The tests should be conducted at cryogenic temperatures with low-speed (20 degree/sec) small angle (10 degree) cycling. Overall feasibility of the system should be demonstrated at this time by conducting these tests with complete drive mechanism hardware for at least three ganged drums.
- d. Vacuum Testing - Vacuum tests (10^{-9} in Hg.) are required for both the bearings and the bands. Endurance tests similar to Section c should be conducted in a vacuum for the bearings. The breakaway torque of the bands and bearings should be determined after long periods at a fixed position in a vacuum.

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e. System Integration - Ultimately, there will be a need to mock-up and test the entire drum, drive, and actuator system to demonstrate that it performs satisfactorily. This will require a full reactor's complement of simulated control drum with simulated frictional torques. The entire ganging mechanism should be fabricated and assembled. While being driven by the proper number of actuators, the system's response to step (including scram), ramp, and cyclic changes in command position should be determined. The tests should be conducted at both cryogenic and room temperatures.

2. Control Drum

At present, the NR-1 control drum is merely a scaled-up version of the NRX-A6 drum. However, due to the changes in dimensions, it is desirable to conduct a transient thermal bow test to confirm the analysis. As is presently planned for the NRX-A6 drum, the amount of distortion should be determined as a function of initial drum temperature and reactor startup flow ramp.

3. Control Sleeve Development

Significant advantages to the actuator system might result from the low friction torque of the control sleeve design. Thus, it may be advisable to continue the development of this design as a back-up to the prime NRX-A6 type drum design.

The effectiveness of the spiral coolant flow in the control sleeve design should be confirmed by experiment. First, the impedance of the spiral flow path should be determined as a function of Reynold's number (10^4 to 10^6). Then, the temperature distributions during transients should be found. This would include a thermal bow determination.

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C. CORE LATERAL SUPPORT

1. Lateral Support Springs

The following aspects of lateral support spring design should be investigated experimentally:

- a. Conduction Cooling - The adequacy of conduction cooling of both constant and variable section springs should be confirmed. The testing should cover manufacturing variations, tolerance stack-ups, and eccentric positioning. Complete simulation of the reactor is required except that uniform internal (electrical) heating of the spring is acceptable. The maximum spring temperatures for different combinations of the above variables should be found. The test should be conducted in a stagnant hydrogen environment with approximate gas temperatures of 1000°R and up to 750 psia pressure.
- b. Spring Relaxation - The loss of load capability during decay heat or flight accident conditions should also be investigated. It is necessary to know the change in free length of a loaded spring after being held at high temperatures (1200°R to 2200°R) for periods of time up to 50 hours. The 50-hour time period is arbitrary but consistent with requirements for previous reactors. For the lower temperatures (1200°R to 1300°R), a low amplitude (± 5 percent of load) vibration (to 2000 cps) should be applied to the spring for the first hour of testing.
- c. Spring Pocket Purging - The presence of trapped air or other gases in the spring pockets will affect the conduction cooling of the springs during reactor operation. Further, the presence of trapped hydrogen could

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result in a potentially dangerous situation during disassembly if a cutting tool is used near the pockets. The purge characteristics of these pockets can be found only by testing. The gas composition in the pockets should be determined subsequent to simulated purging as done during reactor testing.

2. Seal Segment Development

A number of alternate seal segment designs are now being investigated for possible use in NR-1. The objects of these new designs are to reduce leakage past the seals and to reduce the internal heat generated in the seal materials. The designs include both alternate configurations with fewer scarf joints and alternate materials such as P-O3, RVD, and Carbitex. The selection of the proper design will require data from development tests.

Reduction in internal heat generation can be attained by either a lower density material (Carbitex) or a reduced graphite volume. The volume might be reduced by fabricating a hollow seal using either P-514 cement or a high temperature braze. The porosity of these materials must be determined experimentally at temperatures up to 3500°R. If found to be excessive, the improvement possible with metal coating (molybdenum) or other means should also be determined. The flexural strength of the hollow seals should also be determined at various temperatures from cryogenic to 3500°R. The present high temperature braze development program must be completed.

The reduction of the number of scarf joints requires a change in material from the present ZTA graphite. Full scale tests of single joint P-O3 and Carbitex seals should be conducted to determine their mechanical characteristics. Their sealing effectiveness should be determined under conditions simulating lateral

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core shift. Further, the effects of radial and axial thermal gradients should be found. In all of these tests the operating limits should be identified.

Other operational limits should be identified by testing similar to the NRX-A6 partial length-full diameter core tests with NR-1 size components. Lateral "g" load capabilities should be determined. The wear properties of the plungers and seals are needed. The possibility of loss of sealing due to permanent core shift caused by friction lock-up of the lateral support system should be investigated. Preliminary data using the NRX-A6 test hardware would be useful for developmental purposes.

Additional testing effort is required to study the seal segment angles. Testing should determine relationships between seal angles and wear life, sealing effectiveness and plunger load. Further, it is expected that a new method for retaining the outer seal segments will be required for NR-1. Each candidate design should be structurally tested to determine its load carrying capability. These retention devices are normally loaded only during shutdown or during lateral core shift. Thus, the tests should be conducted after no-load thermal cycling to steady state temperature.

A lateral vibration test of a scale model of the reactor (highly simplified i.e., one-third scale with solid graphite core and three rows of seal segments) is being considered to determine the effects of the forward end restraint of the tie tubes and lateral support seals. With such a model, resonant frequencies can also be compared with analyses early in the design. Later, a full scale reactor will be vibration tested (see Section C-5, Reactor Vibration Tests). A similar response of these two units will substantially contribute to confidence in future reactor model work.

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3. Plunger Pins

The diameter of the plunger pin is enlarged at the end in contact with the spring. This permits the plunger hole in the reflector to be as small as possible to improve the cooling of the beryllium. A flexural strength test should be conducted on this stepped configuration using reactor grade hardware. Both the axial and the lateral bending loads from the seal segments should be simulated.

4. Thermal and Mechanical Properties of Contacting Surfaces

At present, the thermal analysis of the core periphery and the reflector requires that assumptions be made regarding the thermal resistances at the interfaces between the various components. Additional test data is required to improve the accuracy of the analysis in this area. The data should provide the thermal conductivity of the interfaces between the components (i.e., graphite on graphite, pyrofoil on beryllium) as a function of temperature, pressure, normal forces, surface hardness, and finish.

Additional data on the friction coefficients at these interfaces as a function of the same variables is required. This information will result in greater accuracy in predicting the mechanical loads on the core periphery components.

5. Reactor Vibration Tests

During the period covered by this document, preparations will be made and some vibration testing will be done with the objective of fully qualifying the NR-1 design for the vibration and shock environments of both ground testing and flight (including booster operation). Near the end of the period, the CFDTs reactor will undergo a low level sinusoidal vibration test to identify the major resonances of the reactor components. This information plus additional information that will

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be gathered on the ground test and flight environment will be used to plan and prepare for complete sinusoidal and random vibration testing and shock testing of the CFDTs reactor. Studies will be made on the possibility of subjecting a later NR reactor to the vibration and shock inputs that are expected during booster operation and then shipping the reactor to NRDS for power tests.

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D. SHIELD

The development of a flight type shield for NR-1 involves two tasks. The first is screening and manufacturing development of candidate shield insert materials such as TiH and B_4C . The second task covers the design of the shield elements which mechanically hold the inserts within the reactor. There are two design approaches for the shield elements. The first is to mount them in a separate shield plate forward of the core support plate (as planned during the NRX-A program). The second is to mount the elements on or within the support plate. The details of the development testing in conjunction with each task are as follows:

1. Insert Material Development

The thermophysical properties of each material must be carefully evaluated under operating conditions. It is well established that there exists a transition of TiH to $TiH_{1.25}$ under hydrogenation conditions. This is attendant with a change in the lattice strain and about a 15 percent change in the theoretical density. The change in lattice conditions imply a change in the thermal conductivity. There are additional manners in which the thermal conductivity may change. One such is the result of radiation damage of the lattice at cryogenic temperatures. The magnitude of this change is dependent upon the total dosage, temperature, and specific material. All of this indicates a change in the thermal diffusivity and it is therefore necessary to evaluate these before transient thermal conditions of the shield can be described.

Samples of candidate insert composites should be produced with varying amounts of TiH and B_4C . The samples should be produced in groups with slight differences in production methods between groups to permit investigation of the processing variables. The samples should be used to obtain the following information:

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- a. Stress-strain data at both cryogenic and room temperature. Room temperature tensile yield and ultimate strengths and elongations. Thermal shock data.
- b. Metallographic sections including microphotographs.
- c. Chemical analysis of hydrogen content.
- d. Irradiation (3×10^{19} nvt) effects on:^{*}
 - (1) density, thermal conductivity, and specific heat.
 - (2) room temperature and cryogenic stress-strain relations.
 - (3) metallographic sections.

This testing is intended to complement the data obtained from experimental inserts in XE-2.

2. Shield Element Testing

Flow tests are required to determine pressure drop as a function of Reynold's number for the range of Reynold's numbers encountered in the reactor.

Cryogenic vibration tests are also required to find resonances of the shield elements for inputs from 10 cps to 2000 cps.

3. Shield Assembly

A full scale cryogenic vibration and pressure drop load-deflection test should be conducted on the final shield assembly design (CFDTS hardware). The exact vibration inputs and loads will be determined after the element tests of Section D-2 have been completed. The testing should include the shield support ring (if any)

*This is part of a more general program aimed at obtaining radiation effects on other reactor materials (i.e. aluminum, titanium, Inconel, etc.)

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on the core support plate. If the shield is integral with the core support plate, this testing can be combined with the testing in Section E6 below.

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E. COUNTERFLOW TIE TUBE SYSTEM

1. Tie Tube Development

Further structural testing is required for both the tie tube and the counter-flow tube. In addition to elongation and tensile tests on the full assembly, lateral flexibility and bending fatigue tests should be conducted on the forward end of the tie tube assembly. Tests to determine the resonance and dampening characteristics of the tie tube assembly should be completed to define the tie tube centering requirements, if any. This latter testing should be conducted both with and without high pressure gas flow. The integrity of the aft end welds should be checked by pressure testing to 1000 psid at cryogenic temperatures and 250 psid at steady state operating material temperatures (approximately 1200°R).

Flow testing is required to determine the pressure drop versus Reynold's number relationships for both the entire tie tube assembly and the aft end turn-around. Information is also required on the effects of heat addition on pressure drop. This information should be obtainable from the central element and cluster corrosion tests described below.

2. Corrosion

The forward end orificing of the NRX-A tie rod channel resulted in a relatively low axial pressure drop on the pyro sleeves in the central element. However, in the counterflow cluster this pressure drop is increased to full core pressure drop. Thus, higher leakage flow rates can be expected past the pyro sleeve seal at the forward end. Furthermore, the changes in dimensions of the pyro sleeves will probably result in new optimum seal loading forces which must be found by experiment.

In view of the above, both full cluster and central element corrosion tests should be conducted. The weight loss of all graphite components should be found

for the equivalent of 60 minutes full power reactor operation.

3. Supply Plenum

A portion of the supply plenum should be mocked-up for leak testing.

Leakage rates between plenums, along tie tube threads, and along the counterflow tube should be found. The leakage data should be plotted as a function of pressure drop (up to 1000 psi), density (0.05 to 5.0 lb/ft³), and viscosity (0.5×10^{-5} to 1.0×10^{-5} lb/ft-sec). Various seal designs might require additional leakage testing depending on the results of the above tests. The effects on leakage of temperature differences (1000°R) between the forward, the aft plenum plates and the core support plate, and temperature gradients within each plate (400°R) should be determined.

4. Effluent Mixing

The design requirements for satisfactory mixing of the effluent gas with the other core flow must be determined by testing. The turbulence of the flow and the length or configuration of the flow path required to assure a completely uniform flow temperature must be found. Both the main effluent flow into the reflector outlet plenum and the leakage flow through the core support plate should be considered.

5. Supply Plenum Manifold

A cryogenic vibration and leak test of the final supply plenum manifold and piping design should be conducted to be certain that the system is properly supported within the reactor. A pressure test to 1000 psid is also needed.

6. Support Plate and Plenum Assembly

The load-deflection characteristics of the final plenum and support plate system design should be determined by full scale testing (CFDTS hardware). This is needed to confirm the analysis of the stiffening effect of the plenum plates.

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Cryogenic gases should be passed through the system to confirm the analysis of the effects of thermal distortion on the seals in the plenum (Section E-3).

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F. FUEL ELEMENTS

1. Fuel Development

A complete development program is now underway to develop the high void fraction fuel for NR-1. Testing is being done to determine the basic properties of these fuel elements. Continued testing will be needed to completely determine the Weibull constants, elevated temperature strength, and corroded strength of the new elements. If some properties are found to be unacceptable by these tests, additional development effort will be required. In addition to these tests, a complete corrosion test program is needed. This corrosion testing should determine weight loss as a function of time and temperature. Metallographic inspection should also follow the corrosion tests.

Other programs are now underway as part of the NRX-A fuel development program. These include: moly overcoating, impregnation with graphite to reduce permeability, brazed tips, graded tips, and tab tips. These programs should result in suitable production techniques and qualification for the NRX-A reactors. The ability to extend these designs to high void fraction elements should be confirmed by high temperature testing. This testing should include corrosion tests with metallographic inspection for all the designs. For the tip designs, the effect of reduced cross section on the flexural strength at both elevated (500°R) and room temperature should be found. The improved porosity of the impregnated element should be confirmed for the thin-webbed high void elements. It is expected that the larger channel diameters in the high void fuel will result in a different crack pattern in the niobium coating. Thus, the effects of the moly overcoating might be different and should be determined by testing.

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2. Core Bundling

The design of alternate cluster support schemes and the stress analysis of the high void fraction fuel elements requires additional information on the variations of frictional forces between fuel elements. Thus, continued testing is required to find the force necessary to move an element axially as a function of radial position and bundling pressure for the NR-1 size core. Also, additional testing is needed to determine interstitial gap dimensions for the larger core. This is required for cores with both coated and uncoated peripheral elements.

The above information can be used in conjunction with flow data on narrow channels between graphite components to more accurately predict interstitial flows and corrosion. Testing should determine flow rates as a function of pressure drop, viscosity, surface conditions, and channel L/D ratio for gaps up to 0.015 inch.

3. High Temperature Fuel Orifices

The program for developing a high temperature fuel orifice sealant should be completed. This development promises to substantially increase the temperature margin of the forward end of the core. The techniques required for brazing the orifices should be completely developed. The overflow of braze material into the channel will require flow testing to determine new loss coefficients over the range of Reynold's numbers expected in the reactor (10^4 to 10^8). Also, choking investigations will be needed for pressure ratios down to 0.5. The bending and tensile load capability of the brazed orifices should be found by testing.

Due to the uncertainty of the outlet flow characteristics, the film coefficients immediately aft of the orifices have always been uncertain. An attempt should be made to experimentally reduce this uncertainty.

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4. Core Periphery

The proper selection of a core periphery design for NR-1 requires additional structural, thermal and flow tests of the candidate designs under simulated reactor operating conditions. The simulation should include the lateral support spring loads, seal segments, peripheral pressure distribution with axial loads (up to 250 psi core pressure drop) on the filler strips, and heat input from the core. Leakage (radial inflow) rates, pressure and temperature patterns, axial load capacity of the filler strips effects on coatings and assembly clearance limits should be determined. Ultimately, an endurance test of the final design should be conducted to demonstrate a 60 minute operating capability.

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G. CLUSTER SUPPORT

1. Support Block

The counterflow tie tube will require modifications to the NRX-A6 support block. This modified block should be added to the present support block testing program. This program compares various block designs based on crack initiation at room temperature and during thermal cycling. Comparison is also made based on ultimate load capacity at both room temperature and at about 4500°R. Some reactor simulation is also included in the program. This includes subjecting the block to thermal shock and gradients as they are expected during startup and shutdown. It is particularly desirable to investigate the circumferential and radial temperature gradients in the irregular blocks. Furthermore, testing is required to develop an understanding of coating welding and the effects of temperature, pressure, and time in the reactor.

The support blocks used in the central element and full cluster corrosion tests mentioned above should provide additional test data. These blocks should be submitted to metallurgical inspection to find any evidence of the formation of eutectics in the graphite or carbiding of any metal components (such as metal protection cup).

(See also Section J, Vacuum Testing)

2. Blockless Cluster

Additional ultimate load-carrying capacity data is required for the blockless cluster components. In the case of the graphite components, the data should include crack and creep rupture information.

Early core bundling tests with the seven-cluster test rig should be conducted that include a determination of the load distribution on the forward end

keys when one or more of the forward end redundant cluster supports are operative. The test results can be confirmed later by the testing described in Section F-2 (core bundling) above.

Flow and corrosion tests are also required for these clusters. These tests should determine whether there is any alteration in the interstitial pressure distribution for cores containing these clusters.

3. Metallic and Carbide Support Block Components

The 60-minute life requirement may necessitate the use of refractory metals, carbides, and/or alloys and new insulation materials. These materials will see a wide range of transient thermal environments followed by steady-state operation and a transient cooldown period.

Current information on refractory materials, particularly tungsten and its alloys, indicate substantial property changes due to recrystallization when subjected to elevated temperatures even for a short time. Since these alloys were developed primarily for elevated temperature operation (above 3000°R), very little information is available at other temperatures. Also, little information is available on the properties after exposure to temperature that can cause recrystallization.

Tests are needed to establish or confirm the following:

- a. Tensile-elongation-reduction of area properties over the range 150°R to 5000°R.
- b. The effect of varying times of exposure to elevated temperatures on these properties (recrystallization).
- c. Resistance to thermally-induced stresses over the operating temperature spectrum.

- d. Low-cycle fatigue characteristics.
- e. Radiation effects*.

New type insulation materials such as zirconium oxide may be desirable in view of the corrosion problems with graphite. Since properties over the range of 150°R to 5000°R are not available, the following should be established by tests over this range:

- a. Strength
- b. Adherence to refractory alloys and/or carbon
- c. Expansion characteristics
- d. Conduction characteristics
- e. Compatibility to hydrogen and radiation

*This is part of a more general program aimed at obtaining radiation effects information on other reactor materials (i.e., aluminum, titanium, Inconel, etc.).

H. INSTRUMENTATION

Work is required in both development and qualification of instrument transducers suitable for NR-1 control and diagnostic measurements. Thermocouples are needed for measurement of both fueled and unfueled element in-core temperatures and for core exit gas temperatures. Calorimeters are required to measure irradiation induced heating rates in the flight-type shield. A suitable displacement transducer is needed to measure lateral and axial core motion. A position indicator is required for the control drums. Strain gages are required for the tie tubes, the beryllium components and the lateral support springs. Pressure transducers are required for measurement of acoustic pressures and for in-vessel mounting for control backup. Accelerometers are needed for measurement of core vibration.

Where possible, NRX qualified transducers must be requalified for the higher radiation levels (increased internal heating) of NR-1. Where requalification is not possible, new transducer designs must be procured and qualified.

Additional testing will be required to develop and qualify techniques for installing these instruments in the reactor.

1. Temperature Measurement

a. Core Material Temperatures: Thermocouples are needed for the measurement of core material temperatures up to 5000°R for both diagnostic and control purposes. Since installation may be in either fueled or non-fueled core elements, they should be suitable for installation in either type. The program will involve the evolution of an insulation system for the thermocouple capable of protecting the junction signal over the entire length of the system to the read-out equipment, together with the evolution

of a sheath system that will protect the thermocouple and its insulators from degradation by the core environment.

Space limitations in the central element with the larger tie tube channel and accuracy requirements may require the development of a fueled element thermocouple installation. Such a design will require testing to determine its thermal, structural, and corrosion characteristics.

b. Exit Gas Temperature: The core exit gas temperature is an important diagnostic measurement and may be considered as a candidate for a reactor control measurement. Present designs employ a cooled sheath routed through the tie rod. However, the gas temperatures in the NR-1 will be higher and counterflow tie tubes will be substituted for the tie rods. Thus, both a higher temperature thermocouple (to 5000°R) and a new installation design are required.

At present there are two candidate designs for the NR-1 exit gas thermocouple installation. The first is an extension of the hollow tie rod concept to the tie tube design. The second is similar to the LASL concept of replacing a fuel element with a specially designed unfueled element which would support the thermocouple. Both of these designs require some development testing. First, corrosion tests will be needed to check the effect of any leakage. Secondly, vibration tests may be needed if the overhang of the sensor at the aft end becomes large. An attempt should be made to determine whether flow induced vibrations result from such an unsupported overhang at the aft end. This latter requirement will require additional instrument development of an insulation system that will function properly to eliminate shunting errors over the length of the uncooled probe.

2. Calorimeters

Proper evaluation and development of the flight shield requires the development of a calorimeter to measure heating rates at various points in the shield region. An accuracy of 5 percent will be sufficient for this unit. Although the immediate need for this unit is in connection with the shield, once developed, the unit will be used at other points in the reactor.

3. Displacement Transducers

Displacement transducers are required to measure relative axial and radial motion of the core. The radial motion of the core must be measured to determine thermal expansion, core concentricity relative to the reflector, and also the dynamic stability of the core. Due to the enlarged core diameter of NR-1 and the higher flow rate, the dynamic stability during test may change and must be confirmed by measurement. The axial measurement monitors axial core motion due to thermal growth and cluster pressure load on the tie tubes. If suitable technique cannot be evolved for applying strain gages to the tie tubes, displacement transducers may be required to measure tie tube deflection and axial vibration.

All of these transducers must operate within 5 percent accuracy in a radiation-induced heating environment of 10 watts per gram. Displacement measurement capabilities required are: Lateral motion when connected to filler strips greater than ± 0.30 in.; lateral motion when connected to lateral support plungers greater than ± 0.50 in.; axial motion of ± 0.25 in.

The installation of the lateral motion transducer should undergo accuracy and endurance tests at both 140°R and 1000°R. If the motion of the filler strips is measured directly, a relatively long cantilevered plunger pin will be required. To obtain a suitable strength, the plunger pin will probably be made of

a refractory metal. This will require galling tests of the plunger in a beryllium guide hole.

4. Drum Position Indicator

The reduction of the number of actuators in conjunction with the ganged drum drive mechanism in NR-1 results in a loss of position indication for each control drum unless such an indicator is added at each drum. This position indicator would not be used for control but only for diagnostic purposes.

Such an indicator must be developed and qualified for the reactor. Tests should then be conducted simulating the reactor installation to check its accuracy and endurance.

5. Strain Transducers

The lack of data on beryllium properties makes it desirable to employ strain gages on the reflector, especially in the web and coolant hole areas. Techniques must be evolved to properly apply strain gages to beryllium. The thin-walled tie tubes will also require a new strain gage bonding technique. Reliable means of bonding and gage protection are not yet available for foil gages. The Microdot weldable gages exhibit severe "zero shift" due to irradiation. Both the mounting technique problems and the "zero shift" problem must be solved to have suitable strain gages available for NR-1. This should be followed by tensile and fatigue tests to assure that no degradation results in the structural properties of the beryllium or the tie tubes.

6. Pressure Transducers

Pressure transducers are required for in-vessel mounting for use as control back-up in the event that a nozzle must be replaced, and for measuring acoustic

pressures. The control back-up unit must operate in a radiation induced heating environment of 5 watts per gram. Frequency requirements will be relatively low, approximately 15 cps. Acoustic pressure transducers may be operable with short lines (up to 18 inches) for frequencies ranging from 100 to 1000 cps. If line response characteristics are such that the transducers must be mounted on the core side of the shield, it will be necessary to develop a unit capable of withstanding 10 watts per gram with good accuracy at high frequencies. The question of line response should be resolved before extensive development is conducted on acoustic units. If lines to the point of measurement from a shielded transducers can be used, the development problem will be lessened.

7. Vibration Transducers

Presently applied vibration transducers are used external to the reactor vessel. The presently developed **velocity pick-ups** should be qualifiable for similar applications in NR-1. If higher radiation resistance is required, the development program to obtain a more radiation resistant unit can be conducted concurrently with the work on displacement transducers.

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I. CORROSION

Corrosion studies are being performed on a continuing basis as part of the developmental test program in the Laboratory. This program is in addition to the specific component tests mentioned previously and is an attempt to determine the effects of all known variables on the phenomenon of corrosion. Where it is assumed that a particular variable or mode of control is more important for a specific reactor, increased emphasis will be placed there.

The more familiar variables are gas and material temperature, gas pressure, gas composition, material geometric, chemical, and thermophysical properties, material surface finish, flow channel geometry, and radiation effects. The more familiar controlling modes for chemical reactions are surface area, boundary layer diffusion, pore diffusion, and product or reactant absorption or desorption. These modes introduce additional variables which govern the simultaneous mass-heat-momentum transfer with chemical reaction. Chemical, thermal, or momentum boundary layer thickness and dynamic regime such as continuum, transition, or free-molecular are among these.

Changes in the flow geometry as well as in the values of the thermophysical properties will result in variation in the controlling mode. Analysis of the reactor conditions and configurations as well as those of the Laboratory tests will be necessary to validate any data trends uncovered. The final result of the program will be as complete an understanding of the corrosion phenomenon as possible.

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J. VACUUM TESTING

In addition to the vacuum testing of ganged drive and control drum components mentioned previously, some more basic vacuum testing should be initiated. This testing should investigate the changes of material properties at moderate vacuum (10^{-5} in Hg) and various surface effects at higher vacuum (10^{-9} or higher). The stress-strain relations and thermal conductivity of NERVA materials are of particular importance. The surface effects include friction and welding characteristics and coating adherence. Initially, the testing should cover fueled and unfueled graphites (including pyrolytic), beryllium, and the various refractory and carbide materials mentioned in connection with the cluster support development work.

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K. REACTOR CRITICAL EXPERIMENTS

1. Power Distribution

Tests are necessary for the specification of the channel - by - channel power factors for orificing the NR-1 core. Radial fission density distribution for at least three specified drum bank positions should be made. Also, at these drum positions, detailed (7 channels per element) measurements of the variations in fission density in the core peripheral region should be determined. Local fission density peaking around the tube elements containing hydrogen (polyethylene) should be measured as a function of hydrogen content. (See also Section K-5 below).

2. Control Drum Span

The integral drum span as a function of drum bank position is required with Cu-B₄C control vanes. The effect of material loss (carbon) from the core on the critical drum bank position should be determined.

3. Reactivity Coefficients

The calculated reactivity coefficients of the materials used in the NR-1 reactor differ considerably in magnitude from those used in NRX-A. Thus, a systematic and complete determination of the reactivity worths of the NR-1 materials should be made as a function of radial distance from the core centerline.

4. Fuel Reactivity Check (FRC)

Ultimately, production fuel elements should be used to check the reactivity (shim) effects of variations in composition from those used to obtain preliminary information. Approximately 126 fuel elements from each vendor should be systematically substituted for the preliminary elements at different radial locations. The resulting changes in reactivity should be measured.

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5. Worth of H_2 (Polyethylene vs. GH_2)

The difference between the worth of H_2 in the bound (polyethylene) state and free (gaseous) state is not known but is thought to be small. The primary reason is that the bulk of the worth of H_2 is due to its leakage reduction effect at relatively high energies where the two e.v. binding energy is inconsequential. However, this few percent effect may be significant in the NR-1 cold-to-hot reactivity due to the large amount of H_2 involved. Thus, the difference should be found experimentally to within one percent.

6. Dosimetry Measurements

Radial and axial measurements of the gamma doses and the fast and thermal neutron fluxes should be made in the core and the reflector. The gamma measurements in the reflector should be made both with and without the lateral support plunger pins in place. Fast neutron and gamma spectrum measurements should be made using threshold foils. For the neutron spectrum measurements, typical locations in the middle and at the edge of the core, within the reflector and at the nozzle and forward end should be included. Both neutron and gamma spectrum measurements should be obtained on the outer surfaces of the reactor mock-up. Also near the outer surface of the reactor, both the gamma and fast neutron axial distributions should be determined. Additional gamma and fast neutron measurements should be made with radial traverses at the core midplane covering a distance of twenty feet from the center of the core and axial traverses along the core centerline for the same distance from the center of the core.

During the period covered by this document, preparations should be made for measuring the heat rates of the various engine components including actuators

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and propellant tank. Facilities should be planned for measuring the coupling effects of clustered flight reactors.

7. H₂ Absorption in Graphite

To date, an accurate prediction of the worth of hydrogen in the core has been impossible since the amount of hydrogen absorbed by the graphite core components has been unknown. Early tests should be conducted on specimens of the various graphites used in the core to determine the amount of hydrogen that diffuses into the pores of the material. Later, confirming tests should be conducted on reactor-grade components. These tests can be conducted at room temperature and at various pressures up to reactor operating pressure.

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L. ACTUATOR DEVELOPMENT

Introduction:

Actuators for both the control drums and for tie tube flow control will be developed and/or qualified for NR-1. For the control drums, the effort will be directed away from high speed performance requiring wide bandwidth capability in the actuator. Thus, certain simplifications in the overall operation of the drums can be pursued. This has resulted in the ganged drum system being driven by a velocity type actuator at a relatively low speed (5 to 40 deg/sec). Working with this concept, the intent is to modify the existing XE-1 design to retain the well-qualified basic hardware but to replace the present servovalve with simpler and more reliable solenoid valving if possible.

In the case of the tie tube flow control, it has been proposed that tie tube flow be used as a means to control the reactor power level in conjunction with the control drum system. The two subsystems would then represent the shim and regulator modes originally envisioned for the control drum actuators. The eventual benefits to be derived from such a system depend upon many factors not yet assessed. However, reduced drum span requirements and drum speed requirements are possible by-products of the tie tube control. Thus, a program will be followed aimed at providing the NR-1 tie tube system with a variable flow capability controlled by a suitable valve and actuator. Preliminary investigations indicate that the Turbine Power Control Valve and Actuator (TPCV) might be capable of controlling the tie tube flow. If so, there will not be a need for new actuator development in this area.

1. Control Drum Actuator

The development testing will begin with a breadboard model with a nominal drum load. This will result in preliminary sizing of the actuator components

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(valves, solenoids, viscous damper, gears, etc.). The open loop performance will be examined during load variation tests and extreme temperature tests. The reactor closed loop response will be evaluated by simulated control loop testing using a computer tie-in and a breadboard controller, actuator and nominal load. This testing is intended to verify the prior analog model and aid in establishing a subsystem specification. Further breadboard tests will be conducted to determine the characteristics needed for component performance and environmental tests that will follow.

Next, proof test models will be constructed and tested individually to a specification. Then ganging tests will be conducted to verify the gang characteristics of the actuators (torque distribution, coupling effects, velocities, and key component performance). Both open loop and reactor closed loop tests with computer tie-in will be conducted as part of the ganging tests. After completion of these tests, the system integration tests described in Section B-1 (Ganged Drive Mechanism) above will be conducted.

2. Tie Tube Flow Control

Most of the detailed development testing depends upon the capability of existing hardware (TPCV, AG-21, butterfly valves, etc.) to meet the specific demands for this new subsystem. Based on some very preliminary estimates, it appears that one or two of the existing qualified butterfly valves can handle the maximum flow and the environmental extremes. However, the throttling range required of the valve is not well defined and this leaves the flow loading and internal construction in question. With regard to the actuator there appear to be two contenders, i.e. the Bendix TPCV, and the General Electric AG-21, however, both are complicated servos and potential simplifications could arise and should

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be pursued in the tie-tube flow control application. It is conceivable that with sufficiently low flow loads on the valve the existing AG-20 could be used. For purpose of development testing shown below, it is assumed that the specification can include existing hardware, and therefore, the first step is evaluation of hardware capability:

- a. Valve Tests: The valves will be tested with water with several cross checks on ambient and cryogenic helium or hydrogen. The tests will determine the flow coefficients as a function of Reynold's number, pressure drop, and valve angle. Also, the operating force will be found as a function of valve angle and pressure drop. Both throttling range and leakage flow will be determined as functions of pressure drop.
- b. Actuator Tests: These tests will determine amplitude, bandwidth, torque-offset including stall torque, open loop stability margins, actuator component stability, and load bandwidth requirements.
- c. Valve and Actuator Combination: On-line tests of the selected valve and actuator combination will include ambient to cryogenic temperature range tests to verify specification capability. Later proof models will undergo further temperature range tests and also pressure and vibration tests. Finally, qualification tests will be conducted to demonstrate on-line performance with tie tube flow impedance included.